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FOREWORD

This publication presents the results of a symposium held in Vienna during the week 8–12 November 1982, the fifth in a series which began in 1965, and which currently is planned for continuation at four-yearly intervals.

International safeguards for nuclear material is a broad field, and this symposium, like its predecessors, undertook to give a balanced consideration to all aspects of the subject. Symposia can only include papers where work is being performed, however, and the history of safeguards R & D efforts has not always been a balanced one. "Nuclear Materials Management" in 1965, dealt almost exclusively with accounting and measurement systems to be used, or in use, by plant operators to control their own nuclear material. "Safeguards Techniques", in 1970, recognized that material control systems were seriously hampered by a lack of accurate measurement methods, equipment or techniques, and placed emphasis on the development of measurement methods. "Safeguarding Nuclear Materials" in 1975 concentrated on the recognized need for inspector verification, and discussed both the statistics and the measurement practicalities of verification. "Nuclear Safeguards Technology 1978" occurred at a time when safeguards goals were much in discussion, and papers related to the possible achievement of relatively stringent goals were actively discussed.

Where then lay the emphasis in this symposium? First of all there was an emphasis on practicalities. Measurement methods which were ideas in 1970 or 1975 were described in terms of operating experience with working equipment. Verification was a much-discussed topic, but again the emphasis was on working verification equipment, or on authentication (a new word implying that the inspector has valid reason to trust automatic measurement equipment installed and used by the plant operator) of newly developed and usually highly automated NDA equipment. Even near-real-time material accountancy, the primary survivor of the advanced concepts discussed in 1978, was discussed not in terms of whether it would work, but rather in terms of the practicalities of various statistical tests, and how its remaining practical problems might be handled.

There was also an emphasis on matters relating to material accountancy as contrasted with containment and surveillance (C/S). Papers related to C/S were presented, of course, but such topics as quantitative destructive or non-destructive measurement methods, the statistics of MUF analysis, and safeguards approaches based on material accountancy, clearly predominated. It is probably unwise to read too much into this observation, but there is at least a suggestion that, at least for the near term, there is no known alternative to a good system of material accountancy as a basis for international nuclear material safeguards.
EDITORIAL NOTE

The papers and discussions have been edited by the editorial staff of the International Atomic Energy Agency to the extent considered necessary for the reader's assistance. The views expressed and the general style adopted remain, however, the responsibility of the named authors or participants. In addition, the views are not necessarily those of the governments of the nominating Member States or of the nominating organizations.

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AN ASSESSMENT OF EXISTING SAFEGUARDS

(Session 1)
Chairman

F. BROWN
IAEA SAFEGUARDS
Status and prospects

H. GRÜMM
International Atomic Energy Agency,
Vienna

Abstract

IAEA SAFEGUARDS: STATUS AND PROSPECTS.
The IAEA has just celebrated its 25th anniversary, and the first safeguards inspections were performed twenty years ago. Counting only since 1978, some 5100 inspections had been performed up to mid-1982, using a staff which now includes about 130 inspectors. Despite these impressive figures, and the fact that the IAEA has never detected any apparent diversion of nuclear materials, there are increasing public allegations that safeguards lack effectiveness. After briefly reviewing the nature of IAEA safeguards agreements, the paper examines the political and technical objectives of safeguards together with some of the criticisms which have been voiced. Allocation of limited safeguards resources is examined in terms of the sometimes conflicting allocation criteria which are contained in various safeguards documents. The paper argues that the credibility and deterrent effect of IAEA safeguards should not be underestimated. It should be of greater concern that a few States are known to be operating or constructing non-safeguarded nuclear facilities capable of producing weapons-grade nuclear materials. Thus the risk of safeguards would appear to be greatest at exactly the point where safeguards end.

INTRODUCTION

In 1982 the International Atomic Energy Agency celebrated its 25th anniversary. The first IAEA safeguards inspection was performed in 1962. The first safeguards agreements pursuant to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) entered into force ten years later in 1972. Counting only since 1978, some 5100 inspections had been performed up to mid-1982, and the IAEA staff had grown to about 130 inspectors. Today, about 98% of the nuclear activities in the non-nuclear-weapon States are under IAEA safeguards.

These are impressive figures. They have been counter-balanced, however, to a certain extent in the aftermath of the attack on the OSIRAK research reactor in the summer of 1981, by public allegations that international safeguards lack effectiveness and that the world is on the brink of widespread nuclear proliferation. However, we should
not forget that international non-proliferation efforts supported by IAEA safeguards have been and are being successful. In the last eighteen years no new nuclear-weapon State has emerged. Only one nuclear device has been exploded by a non-nuclear-weapon State, and that was in 1974. Nevertheless, safeguards effectiveness is our continuous concern. One of the primary objectives of safeguards is to assure the international community that States are complying with their international obligations regarding nuclear materials and nuclear facilities. The IAEA has never detected any apparent diversion of nuclear materials, but the degree of assurance provided has been less than completely satisfactory. However, major improvements have been achieved since our last symposium.

The concern about the destructive potential of nuclear fission goes back to the 1940s: the bomb preceded nuclear power; control efforts preceded promotion. Non-proliferation policies have oscillated between attempts at prevention through strict denial and liberal technology transfers linked to controls. Later on, and in particular since the International Nuclear Fuel Cycle Evaluation (INFCE), it became clear that restrictive measures alone were doomed to failure in the long run and that there were no "technical fixes" which would prevent the military use of nuclear fuel cycles.

The number of States technically capable of producing weapons-usable material without external support is already large and will increase further. What can be done to prevent this potential from being realized? Experience has shown that one of the main barriers against proliferation is the concerted political action of States who believe that it is in the best interests of their own security not to acquire nuclear weapons. In all, 114 non-nuclear-weapon States have demonstrated this conviction by their adherence to NPT. Those among these States having nuclear activities, and many of the others, entered into safeguards agreements with the IAEA, covering all of their nuclear activities. The IAEA acts thereby as an objective international auditor entrusted with the task of verifying that the States in question are faithfully abiding by their non-diversion commitments.

Twelve other States have so far submitted only specifically defined material, equipment and information to safeguards. Four of these States operate or are constructing unsafeguarded facilities capable of producing weapons-usable nuclear material. Nevertheless, as I have said, about 98% of the world's nuclear installations outside the nuclear-weapon countries are now under safeguards.
SAFEGUARDS AGREEMENTS

IAEA safeguards are unique in international relations. It is the first time in history that sovereign States have voluntarily agreed to the inspection of sensitive facilities on their territory by foreign nationals. Understandably, even NPT States will accept inspection only under clearly stated legal and technical constraints derived from international consensus and spelled out in safeguards agreements. The many agreements concluded between NPT signatories and the IAEA follow rather strictly the INFCIRC/153 (corr.) model agreement. Other safeguards agreements are to a certain extent standardized by the "Safeguards Document" INFCIRC/66/Rev.2, which contains a series of guidelines leaving room for variation within certain limits.

Experience has shown that in general the safeguards agreements concluded represent a reasonable framework for practical implementation. But the negotiation of INFCIRC/66-type agreements and subsidiary arrangements sometimes creates problems, depending on specific circumstances. Agreements concluded in the late 1970s differ from earlier agreements insofar as they take explicitly into account the progress achieved in the development of safeguards methodology, in particular the use of containment and surveillance measures. There are still occasional difficulties in the negotiation and up-dating of early agreements to conform to present-day requirements. Such up-dating is for instance necessary if a facility under consideration begins to be supplied with nuclear material from an unsafeguarded facility. In this case the loss of verifiability of one basic component of the material balance equation dictates an up-grading of the verification regime at the facility. Such a reconsideration due to developments in safeguards methodology may also become necessary in some cases of INFCIRC/153-type agreements.

On the other hand, taking into account the political problems which emerged at the Second NPT Review Conference in 1980, it would be unrealistic to expect substantial changes in the legal framework which underpins safeguards implementation. A possible exception to this general statement might be the system of International Plutonium Storage at present being studied and intended to implement Article XII A-5 of the Agency's Statute. This system is regarded as an extension of the IAEA safeguards system and may possibly become a reality on a limited scale in the not too distant future. Also, interest has been expressed by the Board of Governors in the possibilities and implications of
extending Agency safeguards to uranium concentrate. This has to be seen against a background of the spread of centrifuge enrichment technology which could lead without the use of a reactor fuel cycle to the direct production, starting from natural uranium, of highly enriched material.

The most important task before Member States and the IAEA is the strengthening of the existing safeguards system by extending the scope of agreements to all nuclear activities in non-nuclear-weapon States, by up-dating old agreements where necessary, and finally by improving their implementation.

However, following a parallel line of thought it must also be remembered that the philosophy behind the NPT was to bring about an immediate end to the further spread of nuclear weapons and to begin negotiations leading to nuclear disarmament. Both the nuclear haves and the have-nots were expected to shoulder responsibilities. In Article VI of the NPT the nuclear-weapon States undertook to make progress on nuclear disarmament. They have not been very successful. When the NPT was concluded they were estimated to have between them 5800 nuclear warheads. The 1981 figure has risen to about 16 000 nuclear warheads. Even more disturbing is the constant evolution of new types of nuclear weapons and their deployment, including those of small calibre. These discouraging developments in vertical proliferation do not make it any less vital to avoid the horizontal spread of nuclear weapons to additional countries but they do present new risks to the security of all States and could place further strains upon the NPT.

THE OBJECTIVES OF SAFEGUARDS

A paramount area in the development of safeguards is inevitably the definition and interpretation of the objectives of IAEA safeguards. These objectives are of course clearly spelled out in the safeguards agreements. Nevertheless, considerable misunderstanding over the role of IAEA safeguards, its potentials and its limitations has arisen during the last few years in the mind of some politicians and among the mass media.

Most of the misunderstandings are based on completely unrealistic expectations of what can be achieved by an international inspection system. For example, it was mistakenly assumed that the IAEA could unilaterally impose safeguards and determine their scope and methodology, that inspectors could roam around countries in search of
unsafeguarded nuclear activities, and finally, that the IAEA could impose effective sanctions in case of non-compliance. It was also not clearly understood that the IAEA cannot prevent a State from proliferating or from withdrawing from a safeguards agreement and that it obviously cannot give any guarantees about future actions of States. These limitations reflect the political realities of our world. Nevertheless this sudden realization has often led to harsh criticism of IAEA safeguards. It is to be hoped that the answers made by the IAEA in response to these criticisms will lead to an improved understanding of the Agency's role and of its safeguards system.

The objectives of IAEA safeguards are spelled out in the IAEA Statute and in some detail in paragraphs 2 and 28 of INFCIRC/153. The political purposes of the system can be seen in these paragraphs, i.e. the expected contribution to non-proliferation, and the technical objectives of the activities necessary to attain the political purposes. The main purposes of IAEA safeguards are:

- to assure the international community that each State is complying with its Treaty undertakings, by verifying its nuclear activities, chiefly by on-site inspection; and

- to deter any such State by the risk of early detection from diverting nuclear material for prohibited uses.

States conclude safeguards agreements voluntarily. It is therefore reasonable to expect that the normal result of applying safeguards will be to confirm that in fact there has been no diversion. By the same token, the IAEA's verification activities should deter any States, if indeed there are any, who might contemplate the diversion of nuclear material or misuse of facilities under safeguards. The deterrent effect is achieved by the perceived risk that IAEA safeguards will detect diversion. A State would weigh against other options the risk of being caught in breaking a solemn international commitment and of being exposed to the subsequent international reaction. The other options are: to refrain from diversion; to withdraw legally from the agreement; or to build an unsafeguarded fuel cycle.

In the case of detected diversion, or indeed the detection of any anomaly which cannot be satisfactorily explained, the IAEA would alert the international community. To date there has been no such report by the IAEA on any diversion of nuclear material or misuse of a facility under safeguards.
THE TECHNICAL OBJECTIVES OF IAEA SAFEGUARDS

The technical objective of IAEA safeguards as given in paragraph 28 of INFCIRC/153 is "the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or other nuclear explosive devices or for purposes unknown".

This definition contains terms which require quantification for the purpose of planning and evaluating implementation activities, in the first instance "significant quantity" and "timely detection". Widely varying opinions have been expressed regarding the present values of these terms, qualifying them from "uselessly large" to "unattainably narrow". To understand the values selected, a basic difference in meaning must be borne in mind between use of these terms in relation to national measures of physical protection as distinct from their use in the context of international safeguards. Protection of nuclear material and facilities against forcible seizure, theft, terrorism and other criminal activities is the responsibility of the State. A national physical protection system may have to be designed to detect unlawful actions within hours or even minutes. Moreover, the unlawful removal or other misuse of only a few grams of plutonium could be significant because of the potential health hazards which may be involved.

Since one of the objectives of an international safeguards system is to deter a government from making nuclear explosives, the detection of the absence of very small quantities very shortly after the act is not necessary and cannot reasonably be required of an international verification system. In this light it was determined with the support of outside experts that for IAEA safeguards purposes the significant quantity should be defined as the approximate quantity of nuclear material which could possibly be used to manufacture a nuclear explosive device. This quantity includes all kinds of conversion and fabrication losses and should not be confused with critical mass calculations. The significant quantities used in IAEA safeguards are of the order of magnitude of 8 kg of plutonium, 25 kg of highly enriched uranium, or about 10 tonnes of natural uranium. These figures are not fixed requirements but only provisional guidelines for the selection of inspection goals for individual facilities. These inspection goals may vary depending on the specific circumstances at a facility and available measurement capabilities.
In a similar way, in the context of international safeguards "timely detection" can be related to the time required to convert diverted material into the metallic nuclear components of an explosive device (the so-called conversion time). In practice the following timeliness guidelines have been selected:

- three weeks for separated plutonium and highly enriched uranium
- three months for irradiated reactor fuel containing plutonium, and
- one year for natural and low-enriched uranium.

It has been argued that these goals have been set too loosely and sophisticated examples have been adduced to that effect. For instance it has been argued that States with many nuclear facilities could escape detection by using a "partitioning" diversion strategy, diverting small amounts from many facilities, each amount below the significant quantity. To detect diversion in this example, the Agency's goal quantities would have to be set at the State level and not at the facility level. However, the strategy in question - an unattractive high-risk strategy - would involve different types and classes of material and presupposes a concerted effort involving a large number of installations, operators and their staff. If such highly hypothetical scenarios were to be covered the result would be a politically unacceptable inspection regime similar to a State's physical protection system. It must be remembered that the IAEA's goals have to be politically realistic and must take into account the capabilities of the available safeguards measures.

There is also an opposite argument: the safeguards goals in terms of quantities are too stringent; they cannot be attained, so the credibility of the system is undermined. This argument has to be examined in the light of experience. In the late 1970s, the quantitative goals set by the Agency could be attained only in the case of a rather small number of facilities. However, over the past four years the number of inspected facilities has increased by more than 50% and the cases where the IAEA has fully attained its inspection goals have increased from 17% to 45% for the facilities inspected and from 45% to 70% with respect to the direct-use nuclear material in these facilities. In many more cases the goals were partially attained, covering the more attractive diversion paths.

In the same period the number of available inspectors doubled. Experience shows therefore a clear correlation
between manpower available and goals attained. Hence it can be predicted that, other things remaining equal, the performance of the Inspectorate will further improve with the build-up of staff resources.

These considerations lead in the light of experience to the conclusion that the present detection goals seem to be reasonable for nuclear facilities currently under safeguards including small and medium bulk handling facilities. On the other hand, safeguards are still evolving. The guidelines on goals and related problems are continuously under discussion within the Standing Advisory Group on Safeguards Implementation (SAGSI), which has embarked on a study on considerations of fuel cycle characteristics in designing safeguards approaches. Also, the development of methods to safeguard the large bulk handling facilities expected in the 1990s may lead to different design goals and approaches.

**ALLOCATION OF EFFORT**

One of the basic factors determining safeguards effectiveness is the estimated Actual Routine Inspection Effort (ARIE) expressed in inspection man-days per year. The ARIEs considered to be necessary to safeguard a facility adequately have been agreed with States in most Facility Attachments. The Total ARIE for 1982 comes to about 10 000 man-days, about 25% of the Maximum Routine Inspection Effort (MRIE) spelled out in INFCIRC/153. However, in 1981 only about 54% of the total ARIE could be "produced" by the inspectors available. In view of this scarcity of labour, the proper allocation of manpower to the facilities to be inspected is of paramount importance. In the first instance the principle of non-discrimination between States has to be respected. At present this principle is implemented by applying equal inspection effort for similar facilities in all countries. However, this procedure can only be considered as a first approximation because according to paragraph 81 of INFCIRC/153 the ARIE should be established with due regard to criteria such as:

- form, composition, enrichment and accessibility of nuclear material;
- effectiveness of the State's system of accounting for and control of nuclear material (SSAC);
- the characteristics of the State's fuel cycle; and the international interdependence of receipts and shipments.
The first of the criteria, properties of the nuclear material, has been and is being taken into account in estimating the ARIE for each type of nuclear facility. The effectiveness of the SSAC has been anticipated in negotiating the agreements with EURATOM and Japan. However, it is difficult to assess and quantify in an objective way the effect on the ARIE of the characteristics of individual States' fuel cycles and their international interdependence. Therefore at present the IAEA approach is "facility-oriented", meaning that it depends only on the type of the facility, without taking specific fuel cycle contexts into account. This leads to equal effort at similar facilities, whereas the "fuel-cycle-oriented" approach suggests that different efforts should apply at similar facilities if they are embedded in different fuel cycles. This might be understood as discrimination by some States if the factors applied by the Secretariat are not agreed system-wide in advance. As I have said, SAGSI is dealing with the problem of developing such objective and acceptable criteria for a fuel-cycle-oriented approach.

THE CREDIBILITY OF SAFEGUARDS

The perception of IAEA safeguards effectiveness, i.e. safeguards credibility, is a fundamental ingredient in the attainment of the political goals of assurance and deterrence. In spite of the facts that at present only about 60% of the required inspection manpower is available and that the system still has growing pains, safeguards are now considered as an important confidence-building measure. Their deterrent effect should not be underestimated. Each year since the submission of the first Safeguards Implementation Report (SIR) in 1977 the Board of Governors has noted the Secretariat conclusions with satisfaction. At its 25th Session in 1981, the IAEA General Conference debating the OSIRAK incident reaffirmed "its confidence in the effectiveness of the Agency Safeguards system as a reliable means of verifying the peaceful use of a nuclear facility". Critics of IAEA safeguards often consider the system in a purely academic way, applying the standards of an industrial quality control system and forgetting the political realities of the world, in particular such unquantifiable factors as the deterrent effect of verification activities.

Nor should it be forgotten that the reports and conclusions of the IAEA are not the only source of information available to Member States. States may have their own national means of detecting unsafeguarded nuclear activities;
they may take into account the internal and external stability of other States and assess their political intentions and their technological capabilities. If in the light of all the information at its disposal any interested government studies the list of 114 non-nuclear-weapon States which are parties to the NPT and which have thereby voluntarily agreed to submit their nuclear activities to international full-scope safeguards, it will see that only 40 of these actually have significant nuclear activities, all of which are under Agency safeguards. It will find that for these States, as long as there are no basic changes in the internal and external political situation, incentives for diversion are limited to very few cases, if any. It should be of greater concern to observe that not all States with nuclear activities are at present prepared to enter into a full-scope safeguards agreement. In particular, serious concern may be felt about the fact that four States are already known to be operating or building non-safeguarded nuclear facilities capable of producing weapons-grade material and that the available stockpile of such material may be considerable.

Thus, from the perspective of a well informed government, the IAEA safeguards system should be seen as presenting a somewhat paradoxical picture. More than 90% of the effort of the Inspectorate is concentrated on facilities which operate in societies with a firm political commitment not to proliferate. The remaining effort is devoted to facilities in countries that have not made that commitment, and finally, the facilities representing a real proliferation risk are not at all accessible to inspection. Thus, the risk of proliferation is greatest where safeguards end.

It is well established that safeguards cannot prevent the diversion of nuclear material or the withdrawal by a State from a safeguards agreement. The system cannot predict future political intentions and decisions. It is an ex-post-facto warning system. It has been stated by some of the media that the IAEA is not living up to this task. This is incorrect. For years the Agency in its Annual Report and SIR has pointed to the non-safeguarded facilities operating or under construction in various countries. The international political reaction was and still is rather muted.

It has already been said that one of the purposes of safeguards is to deter proliferation by the risk of early detection. However, detection per se has only a very limited deterrent effect. Deterrence has to be provided by predictable, quick and effective international political reaction. Both elements, detection by the IAEA and political
reaction, are necessary. It is to be doubted whether the effectiveness of IAEA safeguards is really the weak link in this interaction.

In conclusion I can do no better than to quote from one of the few draft resolutions of the Second NPT Review Conference in 1980 in Geneva, which have been adopted by general consensus:

"The conference notes that the safeguards activities of the IAEA under Article III of the Treaty continue to respect the sovereign rights of States, that there are no indications that IAEA safeguards have hampered the economic, scientific or technological development of the Parties to the Treaty or international co-operation in peaceful nuclear activities, and that they contribute to the maintenance of confidence between States".
RECENT ADVANCES IN SAFEGUARDS OPERATIONS

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Abstract

RECENT ADVANCES IN SAFEGUARDS OPERATIONS.

The facilities and nuclear materials under IAEA safeguards have steadily increased in the past few years with consequent increases in the manpower and effort required for the implementation of effective international safeguards. To meet this challenge, various techniques and instruments have been developed with the assistance, support and cooperation of the Member States. Improved NDA equipment now permits accurate verification of plutonium and HEU bearing items; and optical and TV surveillance systems have improved remarkably. Experience in safeguarding nuclear facilities now includes fast-reactor fuel reprocessing and enrichment plants, even though the Hexapartite Safeguards Project is yet to define an agreed approach for safeguarding enrichment plants. The establishment of field offices now enables the IAEA to adequately implement safeguards at important facilities and also with more effective use of manpower. Closer cooperation with Member States via liaison or similar committees makes for effective safeguards implementation and the speedy solution of attendant problems. The technical support programmes from the Member States continue to provide the basis of the recent advances in safeguards techniques and instrumentation.

INTRODUCTION

A comprehensive presentation of the present status and prospects of IAEA safeguards has been given by the Deputy Director General for Safeguards (Paper No. IAEA-SM-260/131) and several papers are being presented at this Symposium which describe in detail the various techniques and methods developed for safeguards implementation activities. What the present authors attempt to do here is to provide an overview, without details, of what they consider the more important advances in the course of Agency safeguards operation activities over the past few years. Before going into the subject, a few observations are made about the Agency's safeguards implementation.

OPERATIONS

The total number of States party to the NPT is now 117, including three nuclear-weapon States; and the number of States with NPT or NPT-type safeguards agreements in force is 74.
Safeguards agreements are in force with 12 non-nuclear-weapon States which are not party to NPT and in eight of these all substantial nuclear activities known to the Agency are covered by the provisions of existing safeguards agreements. In September 1981 the agreement with France and Euratom entered into force, thus bringing to three the voluntary offer agreements negotiated by the Agency with France, the United Kingdom and the United States of America.

The number of nuclear installations under Agency safeguards or containing safeguarded nuclear material in the last four years is shown in Fig. 1.

The total number of nuclear installations increased from 559 in 1977 to 844 in 1981, and there has been a steady increase in the number of bulk handling facilities and power reactors since 1978 when a large number of nuclear facilities in Japan and Euratom came under Agency safeguards.

Figure 2 shows the quantities of nuclear material under Agency safeguards (except those covered by voluntary offer agreement). It is noticeable that the quantity of plutonium contained in spent fuel and of uranium with enrichment of less than 20% increased sharply in the past four years. The increasing number of complex facilities and the quantity of nuclear material now under IAEA safeguards have of necessity led to large increases in the staff of the inspectorate and the amount of the field effort required to apply effective safeguards.

The number of inspectors has increased in the meantime from 52 in 1977 to 135 at the end of 1981. About 1400 inspections were carried out at 475 facilities in 1981.

Table I shows the remarkable growth in the number of inspections and the number of nuclear installations inspected by the Agency. The increases in the total number of inspection mandays performed, the number of the inspecting staff as well as the increase in the other safeguards-related activities are also evident from the table.

SUPPORT TO OPERATIONS

The effectiveness of Agency safeguards depends not only on the availability of a sufficient number of qualified well-trained inspectors and a good system for data acquisition and treatment, but also on the availability of adequate equipment for in-field verification of nuclear material and for the application of containment and surveillance measures.
FIG. 1. *Number of nuclear installations under IAEA safeguards (except those under voluntary offer).*

![Graph showing the number of nuclear installations under IAEA safeguards from 1977 to 1981.](image)

FIG. 2. *Quantity of nuclear material under IAEA safeguards (except those under voluntary offer).*

![Graph showing the quantity of nuclear material from 1977 to 1981.](image)
### Table I. STATISTICAL DATA ON SAFEGUARDS IMPLEMENTATION, 1979 to 1981

<table>
<thead>
<tr>
<th></th>
<th>1979</th>
<th>1980</th>
<th>1981</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of inspectors(^a)</td>
<td>92</td>
<td>116</td>
<td>135(^c)</td>
</tr>
<tr>
<td>Inspector man-years(^b)</td>
<td>81.5</td>
<td>96.6</td>
<td>125.6</td>
</tr>
<tr>
<td>Number of inspections</td>
<td>~1000</td>
<td>~1100</td>
<td>~1400</td>
</tr>
<tr>
<td>No. of countries) NNWS</td>
<td>44</td>
<td>44</td>
<td>47</td>
</tr>
<tr>
<td>inspected ) NWS</td>
<td>3</td>
<td>3</td>
<td>3</td>
</tr>
<tr>
<td>Number of nuclear installations inspected</td>
<td>307</td>
<td>393</td>
<td>475</td>
</tr>
<tr>
<td>Total inspection man-days</td>
<td>3305</td>
<td>4004</td>
<td>5061</td>
</tr>
<tr>
<td>Number of surveillance pictures taken &amp; reviewed</td>
<td>~6 000 000</td>
<td>~8 000 000</td>
<td></td>
</tr>
<tr>
<td>Number of surveillance systems in use</td>
<td>~130</td>
<td>~140</td>
<td>~160</td>
</tr>
<tr>
<td>Number of seals verified</td>
<td>&gt;2000</td>
<td>&gt;3000</td>
<td>&gt;4000</td>
</tr>
<tr>
<td>Number of Pu &amp; U samples analysed</td>
<td>540</td>
<td>780</td>
<td>890</td>
</tr>
</tbody>
</table>

\(^a\) The figures for 1977 and 1978 were 52 and 79 respectively
\(^b\) The figures for 1977 and 1978 were 49 and 65 respectively
\(^c\) This figure includes 5 inspection assistants
A Safeguards Training Unit was established in February 1980 and upgraded to Safeguards Training Section in October 1981. The number of inspectors who participated in various training courses was 148 in 1980 and 265 in 1981, excluding the refresher courses on radiation protection. One of the major contributions to operations by the Training Section is that newly recruited inspectors, who joined the Agency in between normal Introductory Training Courses, no longer have to wait for weeks or months to get some training. With an excellent training system and material available, such recruits can now quickly cover some course programmes in the interim with self-instruction aids and thus be equipped to carry out inspections within a relatively short time.

The Agency's Safeguards Information Treatment Division provides invaluable support to operations by its well developed capability to sort, store, analyse and compare data received from the Member States and to provide inspectors with the accounting and other essential data for safeguards implementation.

Many speakers at this symposium deal in detail with the safeguards instruments that have been developed recently. Most of this development came as a result of the technical support programmes offered by the Member States via the IAEA's Division of Safeguards Development. A few of these instruments that have lead to recent advances in safeguards operations will now be mentioned.

**SAFEGUARDS EQUIPMENT**

Table II shows the number of non-destructive assay (NDA) devices now available to the Agency inspectors. Remarkable progress has taken place in the past three years in the availability and utilization of the principal NDA equipment.

The inspector's work-horse, SAM-II, has now given way in some applications to the SILENA MCA, which has more sophistication and accuracy. The neutron coincidence counter which had become routinely available by the end of 1980 led to a general improvement in the Agency's ability to analyse plutonium and its compounds. The HLNCC together with the SILENA MCA have now taken plutonium measurement to a degree never attempted with SAM-II or the earlier instruments. The quantitative verification of plutonium, even in fast-reactor fuel assemblies and fuel coupons of fast critical facilities with detector heads designed for specific geometries is now a routine safeguards activity with the HLNCC, of which over 11 units are located at various facilities throughout the world.
Table II. SAFEGUARDS NDA EQUIPMENT: Total number of devices for routine use

<table>
<thead>
<tr>
<th>Device</th>
<th>End of 1979</th>
<th>End of 1981</th>
<th>As of October 1982</th>
</tr>
</thead>
<tbody>
<tr>
<td>NIS - 322</td>
<td>9</td>
<td>9</td>
<td>9</td>
</tr>
<tr>
<td>SAM-II</td>
<td>36</td>
<td>36</td>
<td>36</td>
</tr>
<tr>
<td>SNAP</td>
<td>3</td>
<td>4</td>
<td>4</td>
</tr>
<tr>
<td>BSAM</td>
<td>5</td>
<td>6</td>
<td>5</td>
</tr>
<tr>
<td>SILENA</td>
<td>18</td>
<td>23</td>
<td>26</td>
</tr>
<tr>
<td>Ge DETECTOR</td>
<td>18</td>
<td>18</td>
<td>27</td>
</tr>
<tr>
<td>HLNCC</td>
<td>7</td>
<td>11</td>
<td>15</td>
</tr>
<tr>
<td>ULTRASONIC THICKNESS GAUGE</td>
<td>3</td>
<td>6</td>
<td>9</td>
</tr>
<tr>
<td>CERENKOV DEVICE</td>
<td>-</td>
<td>6</td>
<td>15</td>
</tr>
<tr>
<td>NEUTRON COLLARS</td>
<td>-</td>
<td>-</td>
<td>3</td>
</tr>
</tbody>
</table>

A recent development is the use of the neutron collar for verifying LWR fuel assemblies, which uses a Cf-252 neutron source and HLNCC-type detector and electronics. This instrument will soon become available for routine use.

Of equal importance with the development of the NDA equipment is the use of mini-computers such as the HP-85 in conjunction with these NDA devices. A specific computer program can now be
tailored to a specific measurement at a given facility so that the data are not only collected directly but also analysed and the results displayed at the time of measurement.

The verification of spent fuel by using instruments based on Cerenkov radiation given off by such irradiated fuel has now been made possible by recent developments. Very good results have been obtained with these "night-viewing devices" and every effort is being made to ensure the acceptability of their use to the facility operators and Member States.

Steady improvements have been made in recent times to surveillance devices. The number of film-camera units available for routine use increased from 137 at the end of 1979 to about 200 at present. The performance of the standard film-camera system has been so improved by very rigorous quality control and prior bench-testing procedures that complete failure of a twin-set film-camera surveillance system is comparatively rare these days. A more recent trend is the increasing use of closed circuit television (CCTV) systems and about 29 of these are now available for routine use. Trained technicians now regularly accompany the inspectors to the facilities for installing and maintaining these surveillance instruments.

EXPERIENCE

As shown in Fig. 1 the Agency is now implementing safeguards in a large number of nuclear facilities that constitute the various stages of the nuclear fuel cycle. Considerable experience and knowledge has been accumulated by the Agency in its application of safeguards to research reactors, LWRs, Candu reactors, fast critical assemblies, conversion and fuel fabrication plants.

By the time of the 1978 symposium the first reprocessing plant to come under Agency safeguards, other than as a demonstration exercise, had been inspected for only one year. Four years later the number of such plants has reached four, and includes the reprocessing plant at Dounreay for the prototype fast reactor fuel, which came under safeguards in December 1980. Under the UK voluntary offer agreement, this reactor and its associated reprocessing plant have provided the Agency with experience in safeguarding facilities of advanced design and thereby demonstrating their safeguardability.

Experience in safeguarding enrichment plants is at present limited to the UF₆ cylinder storage and UF₆ sealing areas of such plants. A recent occurrence is the field test at Almelo with
good results from a gas-phase enrichment monitor developed under the Australian support programme to the Agency. Currently and acceptable approach for safeguarding enrichment plants effectively is still under discussion. In this connection the Hexapartite Safeguards Project started in November 1980 to explore various practical safeguards approaches, including limited access to the cascade area, has yet to be completed; it is hoped that this will lead to an agreed safeguards approach or approaches for enrichment plants.

LIAISON WITH SSAC

It has been observed that the existence of an effective State System of Accounting for and Control of Nuclear Material (SSAC) is a prerequisite for applying effective international safeguards. The Agency welcomes the fact that the SSACs have been well established and maintained at highly developed level in some countries.

A growing realization in the course of Agency safeguards operations in recent times is the importance of liaison and coordination of activities with the appropriate bodies in the Member States. Liaison or joint committees have been established with the Agency to foster cooperation and improve safeguards implementation. Many operational problems and difficulties related to safeguards have been solved through such machinery.

The establishment of the joint team of inspectors by the IAEA and Euratom has provided a valuable co-operation and co-ordination mechanism in safeguards implementation. The joint team carries out inspections at important facilities, and the results of such inspection should permit both organizations to achieve the requisite safeguards objectives with minimum intrusion into facility operation. Thirteen facilities are at present under this inspection regime and this number will increase in the next few years with the addition of several other important facilities. The joint team inspection concept also applies to the inspections at Dounreay under the UK voluntary offer agreement.

FIELD OFFICES

It is now two years since the Agency set up its first field office in Toronto, even though the idea had been under consideration for over 10 years. The need to perform certain safeguards activities, such as inventory change verifications at short notice, has always been felt but was never easy to carry out for obvious reasons. In the cases where it is mandatory that the
inventory changes must be verified (e.g. at reprocessing, MOX, HEU fabrication plants) continuous or high-frequency inspection regimes had to be instituted. To operate this type of inspection regime from half-way across the globe poses considerable problems of expense and logistics.

The establishment of the IAEA field office in Canada has enabled the Agency to carry out in a convenient and efficient way the necessary inspection activities connected with the inter-bay transfer of spent fuel at Pickering and Bruce power stations, and the transfer to dry storage silos of spent fuel at Whiteshell. In addition to these, all the inventory changes involving the receipt and shipment of high strategic material are verified, often at short notice, thereby minimizing subsequent re-verification of the sealed items at the KMPs where these inventory changes have occurred.

The Agency inspectors outposted to Tokyo provide the essential coverage at the reprocessing, MOX and other important facilities in Japan. The overall experience indicates that a more efficient use of manpower has resulted since these inspectors can now get to their places of work without travelling great distances. This experience is encouraging and every effort is being made to further develop the functions and capabilities of these outstations.

CONCLUSION

Before closing this presentation, some changes occurring in the operations divisions at present deserve some mention. The use of General Service staff as "inspection assistants", and the rationalization of the management and the utilization of manpower through the recent reorganization in the operations divisions, are measures aimed at improving the work of safeguards operations.

There will still be room for improvement and there will be areas of difficulties, but the authors are confident that with the present trend of mutual co-operation between the Agency and the Member States the outstanding issues affecting safeguards implementation will be resolved. This presentation would not be complete without expressing sincere appreciation to the Member States for the co-operation and the generous support provided to the Agency and its staff. It is not an exaggeration to say that most of the recent advanced instrumentation and techniques used in safeguards operations owe their origin to the technical support programmes from the Member States.
RESEARCH AND DEVELOPMENT PROGRAMMES IN SUPPORT OF IAEA SAFEGUARDS

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Abstract

RESEARCH AND DEVELOPMENT PROGRAMMES IN SUPPORT OF IAEA SAFEGUARDS. Research and development for IAEA safeguards is an important but also expensive task which requires not only substantial financial means but also considerable expertise, manpower, facilities and materials. Most IAEA Member States have preferred to perform the work in their own facilities and to enter into national programmes in support of IAEA safeguards, rather than to provide the budgetary resources for IAEA-owned laboratories. National support programmes have officially been agreed upon with the IAEA by the following Member States: the USA, Canada, the Federal Republic of Germany, Japan, Australia, the United Kingdom and the USSR. In addition, a cooperative programme in the field of safeguards R & D has been concluded with EURATOM. Although some of the programmes focus on specific subjects, nearly all aspects of safeguards R & D are covered in these support programmes. Furthermore, several multinational programmes have been established in order to improve IAEA safeguards in specific situations. The coordination of the programmes and the utilization of the results rest with the IAEA.

INTRODUCTION

In 1970, when the IAEA was entrusted with the task of designing, implementing and successfully maintaining, in nearly all non-nuclear-weapon States, an international system for safeguarding nuclear materials in all peaceful applications, in accordance with article III of the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), only very limited experience in nuclear material safeguards according to INFCIRC/66 [1] was available. Furthermore, this experience was mainly limited to safeguards at nuclear power reactors and research reactors.

In 1970/1971 the Safeguards Committee established a basis for the new safeguards system, which satisfied the requirements of the NPT and which was defined in the document INFCIRC/153 [2]. This document was approved by the IAEA Board of Governors and has been used by the IAEA as the basis for all safeguards agreements negotiated pursuant to Article III of the NPT.
INFCIRC/153 describes the basic concepts underlying the IAEA safeguards system, and the main task for the following years was to develop procedures and techniques for applying the safeguards. Extensive research and development (R & D) activities were to be initiated covering, inter alia:

a) studies on safeguards concepts and approaches
b) the development, testing, implementation and maintenance of safeguards instruments, methods and techniques
c) the development of safeguards information and data treatment capabilities
d) the assessment and evaluation of safeguards results
e) the training of safeguards inspectors
f) the performance of safeguards inspections.

In principle, there were two possible ways of tackling the first three tasks:

To provide the IAEA with the required and very high budgetary resources to build its own R & D facilities and to employ a larger number of researchers; or

To conduct the necessary work in national R & D facilities in close cooperation with the IAEA and to provide the IAEA with the necessary limited resources and funds required for the coordination of the work and for the proper utilization of the results.

It is not surprising that, in a situation of financial shortage and a certain over-capacity in national nuclear R & D programmes and facilities, most Member States preferred to have the work done in their own facilities. Several formalized routes have been established and used for the coordination of the work and the transfer of results to the IAEA. These include:

- Conclusion of safeguards research agreements and contracts (inside or outside formalized coordinated research programmes)
- Participation in safeguards consultants' or advisory group meetings or working groups
- Establishment of national programmes in support of IAEA safeguards
- Participation in multinational programmes aimed at the improvement of IAEA safeguards.

In this paper the national and multinational programmes in support of IAEA safeguards and the experience gained until now
in coordinating these programmes and using their results are described.

THE UNITED STATES PROGRAMME

The United States programme of technical assistance to IAEA safeguards (POTAS) was the first formalized national research and development programme in support of IAEA safeguards, officially implemented in 1976. Since then more than 27 million US dollars have been obligated under the programme. Some 200 individual tasks have been completed and over 50 are currently under way. This is the most comprehensive of the national programmes, covering the development, testing and provision of prototypes of safeguards equipment, the elaboration of techniques and procedures for the usage of the equipment and their evaluation, the provision of information and system studies, planning and execution of inspector and other safeguards staff training (including the provision of training aids), studies and software in support of IAEA safeguards information treatment and the provision of expertise either through short-term consultations or by provision of long-term cost-free experts.

On the U.S. side the programme is controlled and guided by the Technical Support Coordination Committee (TSCC) with members from the Department of State, Department of Energy, the Arms Control and Disarmament Agency and the Nuclear Regulatory Commission. The DOE Office of Safeguards and Security, which has the responsibility for the implementation and technical management of the programme, has set up the International Safeguards Project (ISPO) at the Brookhaven National Laboratory for the day-to-day management of the programme activities. In order to be in direct contact with the IAEA, ISPO keeps its own permanent representative at the U.S. Mission in Vienna, a practice which has turned out to be very successful.

The U.S. programme concentrates most its efforts on equipment for inspector verification of nuclear materials and the application of containment and surveillance for safeguards purposes. More than 20 kinds of newly developed U.S. safeguards equipment are now in our hands for testing, demonstration or operational use. In addition, the development and implementation of verification methods and techniques has been supported through POTAS by making available, inter alia, test facilities, expertise, chemical-analytical services, development of special micro-chemistry techniques for uranium and plutonium and by
testing procedures for the use of the newly developed light-weight, airborne, accident-resistant container (LAARC or PAT-II) for air transport of safeguards samples containing plutonium.

System studies have included diversion path analyses, safeguards approaches and concepts and safeguards effectiveness assessment methodology for several types of nuclear facilities.

IAEA inspectors and other staff members are routinely trained in the USA in advanced verification techniques. Special support has been given in setting up the IAEA safeguards training facilities and the inspectors' training programme at the IAEA Headquarters.

A cumulative total of about 90 man-years of expert assistance has been provided cost-free to the IAEA Department of Safeguards under the U.S. support programme. They certainly have contributed significantly to the improvement of IAEA safeguards during the past six years.

In addition, the U.S. Government has also supported R & D for IAEA safeguards outside the POTAS programme. Besides major contributions to the multinational programmes, one should mention the development of the LAARC plutonium air transport container, the COBRA fibre optic seal and the Remote Continual Verification (RECOVER) project.

The activities performed under the U.S. support programme have recently been summarized in two reports [3,4].

THE CANADIAN PROGRAMME

The Canadian programme in support of IAEA safeguards, which was formally set up in 1977, covers the development, provision and installation of safeguards equipment for all 13 Candu reactors in operation or under construction world-wide as of 1977.

With the Candu 600-MW reactors, for the first time in the development of IAEA safeguards schemes a significant engineering effort has been made to include safeguards as an integral part of a reactor during the design and construction, rather than being a retro-fit requirement.

The programme is broken down into some 40 tasks, all concerned with the basic development of instrument prototypes,
followed by the construction of equipment and installation at designated reactor facilities. With the installation of complete sets of safeguards equipment at the Candu 600-MW stations in Quebec, New Brunswick, Korea and Argentina, the Canadian programme has now reached a phase in which the Candu reactor safeguards concept and the proper functioning of the individual types of equipment are being demonstrated. If this demonstration is successful the main emphasis of the programme will shift to concentrate on the large multi-unit power stations in operation in Canada.

The Canadian support programme is jointly administered and funded by the Atomic Energy Control Board and Atomic Energy of Canada Limited. Approximately 12 million Canadian dollars have been spent under this programme up till now. Beside the development, delivery and installation of the Candu safeguards equipment (CCTV systems, film cameras, core input monitors, irradiated fuel bundle counters, yes/no monitors, irradiated fuel verifiers and tamper-indicating containers and seals for spent fuel bundles), the programme has included the provision of five cost-free experts, system analytical studies and extensive training of IAEA staff members. More detailed information on the Canadian support programme can be found in two recent publications [5,6].

THE JOINT PROGRAMME ON R & D BETWEEN THE FEDERAL REPUBLIC OF GERMANY AND THE IAEA

The joint programme with the Federal Republic of Germany is similar to the U.S. support programme. It was initiated in 1978 and covers a broad spectrum of safeguards R & D activities. The programme, which is coordinated by the Ministry for Research and Technology in Bonn, is conducted in close cooperation with the German nuclear research establishments in Karlsruhe and Jülich, the German nuclear industry and Euratom. About 20 million Deutschmarks have been spent up till now in this programme. Individual tasks concentrate on the development of safeguards concepts and approaches for advanced nuclear facilities such as fast breeder reactors, high-temperature reactors and spent fuel reprocessing plants and other system analytical studies; on the collection, treatment and evaluation of safeguards data; on the development and testing of measurement techniques for nuclear materials (both NDA and DA) and on the development of containment and surveillance equipment. In addition, three cost-free experts have been sent to assist the IAEA Department of Safeguards, and substantial funds have been made available
to procure the ADABAS data base management system, NDA and C/S prototype equipment and IAEA staff travel. An overview of the scope and content of the FRG support programme is given in a recently published report [7].

THE AUSTRALIAN PROGRAMME OF ASSISTANCE TO IAEA SAFEGUARDS

The Australian support programme [8] was formally introduced in June 1980. It is managed by the Department of Foreign Affairs in consultation with the Department of Trade and Resources. Technical advice is given by the Australian Safeguards Office (ASO) and the Australian Atomic Energy Commission (AAEC), which is also responsible for the technical implementation of the research projects. For the first three years of the programme 541 000 Australian dollars have been obligated.

The programme mainly relates to enrichment plant safeguards and consists of a system analysis, including the assessment of available safeguards techniques and the development of measurement equipment for UF₆ (ruggedized assay meter and UF₆ gas phase enrichment meter). The latter instrument was recently successfully tested at the Almelo plant of Urenco. In addition, funds have been made available under the Australian support programme for a "once-only" contribution to the IAEA for the International Plutonium Storage study, as well as IAEA staff travel in connection with the support programme. Most recently, a cost-free expert from Australia has joined the Department of Safeguards.

THE EURATOM-IAEA SAFEGUARDS R & D COOPERATION PROGRAMME

In May 1981 the cooperative safeguards R & D programme [9] was formally established in the framework of the general cooperative programme between the European Atomic Energy Community and the IAEA, as signed in 1975. The programme, which is managed by the Joint Research Centre at Ispra, Italy, aims at an exchange of technical experience, particularly in the area of safeguards R & D and safeguards implementation in European nuclear facilities. It is expected that the programme will result in technical assistance to the IAEA, in harmonization of techniques and procedures and in the evaluation of priorities for R & D as a function of the requirements of the application of safeguards in the Euratom fuel cycle. A total of 27 tasks have been identified in the programme, mainly relating to containment and surveillance, measurement technology, inspectors and other safeguards staff
training, information and data treatment and evaluation. Much emphasis is placed on standardization and harmonization aspects of measurement techniques, data generation and evaluation, and the preparation of reference materials.

THE UNITED KINGDOM SAFEGUARDS R & D PROGRAMME

In July 1980 the United Kingdom Government offered to assist the IAEA with a safeguards R & D programme covering an annual cost of about £ 500 000. The programme [10], which was agreed upon in detail and formally implemented in July 1981, is carried out in the laboratories of the UKAEA at Harwell, Winfrith, Risley, Springfields and Dounreay. The programme's aim is to concentrate on safeguards for those parts of the nuclear power programme with which the U.K. is particularly experienced, such as the fast breeder reactor and its fuel cycle, uranium enrichment plants and nuclear material in storage.

In addition, service elements such as training, chemical analysis, and manual writing, are included in the programme. Both UKAEA and British Nuclear Fuels Limited (BNFL) have provided facilities for in-plant testing of newly developed safeguards equipment. One cost-free expert has been provided to work at the IAEA Headquarters in Vienna.

In parallel with the support programme, the U.K. has donated an Isomass 54E thermal ionization mass-spectrometer to the IAEA safeguards analytical laboratory and is providing continuous support for the maintenance and use of this instrument. Independently BNFL, which is a U.K. government-owned limited company, has held 2-week training courses for IAEA inspectors and carried out a three months' safeguards demonstration programme on the safeguarding of centrifuge enrichment plants at its Capenhurst facility.

JAPAN-IAEA SUPPORT PROGRAMME

The Japanese support programme for Agency safeguards (JASPAS) [11] was formally established in 1981. The programme is coordinated by the Nuclear Material Control Center (NMCC) in Tokyo under the guidance of the Science and Technology Agency (STA). Nine R & D activities have been identified under this programme and are being conducted at the facilities of the Power Reactor and Nuclear Fuel Development Corporation (PNC), the Japan Atomic Energy Research Institute (JAERI), the NMCC and other institutions. The total expenditure for JASPAS
projects and two cost-free experts who have been sent to Vienna amounted to 158 million yen in their fiscal year 1981; this amount excludes all personnel costs for PNC and JAERI staff members.

Six additional new projects have been added to the programme at the recent meeting of the JASPAS review committee, including a demonstration programme at the Ningyo ultracentrifuge $^{235}$U enrichment facility.

The activities in the programme are mainly related to safeguards system designs and safeguards approaches, safeguards data collection, treatment and evaluation, measurement methods and techniques, and containment and surveillance. The techniques developed under two of the JASPAS tasks have already been implemented at the PNC reprocessing plant at Tokai-Mura.

THE SOVIET UNION PROGRAMME

In June 1982 the Government of the Union of Soviet Socialist Republics formally offered to implement a USSR programme of technical support to IAEA safeguards and this offer was accepted by the Director General of the IAEA. The programme, which is now in an early stage, includes, inter alia, 14 tasks related to information processing systems for nuclear material accounting and control; development and improvement of methods and instruments for non-destructive assay of nuclear materials, nuclear material destructive assay methods, surveillance equipment, procedures and technical measures for implementing nuclear facility safeguards; organization of scientific visits, training courses, seminars and schools in the USSR; and provision of experts. One million roubles have been allotted to this programme. Additional resources will be made available at a later stage.

The programme is coordinated by the USSR State Committee on the Utilization of Atomic Energy, and the I.V. Kurchatov Atomic Energy Institute in Moscow has been designated as the technical organization responsible for the programme. A first meeting of the joint committee is expected in early 1983.

THE BELGIAN AND FRENCH PROGRAMME

The Governments of Belgium and France have also offered to engage in support programmes to IAEA safeguards and first discussions on the possible content of the programmes have
taken place during the summer of 1982. It is foreseen that both programmes will be formalized sometime in 1983.

OTHER NATIONAL SUPPORT

Before establishing formalized support programmes all the countries mentioned above provided extensive R & D support to IAEA safeguards. In addition, Bulgaria, the Germany Democratic Republic, Romania, Sweden, Czechoslovakia and several other countries have supported our work through studies, development of instruments, organization of tests for safeguards equipment, training and (cost-free) experts.

MULTINATIONAL COOPERATION PROGRAMMES

Several multinational cooperation programmes were established in recent years, the most important being:

1) The International Working Group on Reprocessing Plant Safeguards (IWG-RPS) [12],
2) The Tokai Advanced Safeguards Technology Exercise (TASTEX) [13],
3) The RECOVER project [14],
4) The hexapartite programme for improvement of safeguards at ultracentrifuge 235U enrichment plants [15].

IWG-RPS

The IWG-RPS was established in December 1978 for a period of two years. Belgium, the Federal Republic of Germany, France, the United Kingdom, India, Italy, Japan, the United States of America, Euratom and the IAEA cooperated in this group to produce a comprehensive study on safeguards concepts and techniques to be used at spent fuel reprocessing facilities. Most emphasis was placed on near-real-time material accountancy and on extended containment and surveillance concepts. Also, questions related to facility design features which could facilitate international safeguards were studied. The group completed its work and published a final report in 1981 [12].

TASTEX

Thirteen different tasks related to safeguards techniques for reprocessing plants have been undertaken in the TASTEX
programme. Japan, the United States of America, France and the IAEA all cooperated, from 1978 to 1981, in this programme to develop and test safeguards instruments, methods and techniques in the Japanese reprocessing plant at Tokai-Mura. The final report was published in 1982 [13]. Two of the tasks have already led to improved inspection procedures. Others are being pursued in the framework of the Japanese support programme.

RECOVER

Participating countries in the testing of remote continual verification techniques within the framework of the RECOVER project [14], which was established in 1979, are Australia, Bulgaria, Canada, the Federal Republic of Germany, Japan, the United Kingdom, the United States of America and the IAEA. The feasibility of the secured and authenticated transmission of safeguards information via public telephone lines has already been demonstrated. The project is now concentrating on the improvement of RECOVER instrument interfaces and on studies related to its cost-effectiveness.

THE HEXAPARTITE PROJECT

The improvement of international safeguards at uranium-235 enrichment facilities using ultracentrifuge technology is the goal of the Hexapartite Safeguards Project (HSP) [15]. Australia, the Federal Republic of Germany, Japan, the Netherlands, the United Kingdom, the United States of America, Euratom and the IAEA are cooperating in this project which was established in November 1980 and was originally planned for a duration of two years but is now expected to end in spring 1983.

COORDINATION OF SUPPORT PROGRAMMES

There are two important tasks which cannot be delegated to Member States: the coordination of the support programmes and the utilization of the results. All support programmes to IAEA safeguards are coordinated through the Division of Development and Technical Support in the IAEA Department of Safeguards. This task requires an ever-increasing amount of IAEA manpower and resources. The overall coordination of all safeguards research and development support programmes is in the hands of the Director, Division of Development and Technical Support, who represents the IAEA in all relevant
coordination bodies. The coordination on the working level is effected through the IAEA project officers, who are responsible for all sub-tasks in all support programmes related to a specific area. The project officers draw up the sub-tasks, ensure the required flow of information, direct the work into the proper direction as required by the IAEA, in particular by the safeguards operations divisions, contribute to the work, organize exercises, secure proper documentation and provide periodic reports on the results. The relatively high turnover rate of IAEA personnel is certainly counter-productive for the required continuity of the project officers' work. The researchers involved in the national laboratories and industry are in continual contact with the project officers and some of them occasionally come to Vienna to report on results.

Contact with the IAEA inspectors in the safeguards operations divisions is established through periodical IAEA internal working group meetings, direct cooperation, participation of inspectors in field demonstrations and exercises and training in new techniques. The technical results on individual sub-tasks are usually reported and discussed during coordinated research programme meetings, consultants' meetings and advisory group meetings. They are often presented to interested listeners at IAEA safeguards symposia, which we normally organize every four years, and at other scientific and technical meetings such as the INMM annual meeting or the ESARDA symposia. More than 250 technical reports have been published on support programme tasks. It is planned to improve the coordination between programmes through the release of an annual safeguards research and development report - the draft of the first issue covering the year 1981 has just been distributed to the support programme coordinators - and by organizing periodic support programme coordinators' meetings, the first of which was originally scheduled for the week following this year's symposium, but has now been postponed to spring 1983. Problems have repeatedly been mentioned in connection with parallel efforts in several support programmes. The IAEA certainly does not encourage unnecessary duplication of work, but, on the other hand, does not wish to establish monopolies. There are several examples where promising work on a specific sub-task in a particular country was terminated at an earlier stage in order to avoid duplication of on-going work in another country, but which then did not show the expected results. Every effort is being made to facilitate exchange of available information at an early date, in order to avoid the duplication of mistakes and to inform researchers and coordinators in different countries of on-going work in
other countries, but parallel efforts or healthy competition is not discouraged.

EXPERIENCE WITH SUPPORT PROGRAMMES AND UTILIZATION OF THEIR RESULTS

All R & D work in international safeguards aims at producing results to help improve the credibility and efficiency of the IAEA safeguards system, and to minimize its burden and intrusiveness on the operator and State. Success or failure of a specific sub-task is to be measured against this goal. It is our general observation that a high degree of willingness exists to start new projects and to perform basic research on new safeguards techniques. However, we experience difficulties when it comes to the stage where research results should be transferred to routine inspector activities and where field-usable instruments should be developed from laboratory prototypes. The required budget and manpower for this process is usually significantly underestimated and serious programme delays are characteristic of this stage of development; where the reliability of instruments is to be improved, the environmental conditions in the field have to be taken into account, simplification of procedures and provision of micro-processor controlled inspector prompting routines is to be considered, and such trivial, but very important, things as safe transport containers, and licences for radioactive sources, have to be obtained. Both the scientists and researchers who have usually performed the development up to this stage are, unfortunately, hardly qualified to do this work, so that contact with possible commercial suppliers is essential at this point. Close cooperation with and the involvement of experienced engineers and instrument technicians is required. Extensive field tests and determination of the working conditions are necessary before a new instrument or technique can be successfully implemented for routine inspector use.

Cost-free experts have contributed significantly in recent years to the improvement of IAEA safeguards. However, not every good researcher or expert is qualified to work for IAEA safeguards. Language problems, inflexibility, and insufficient capability to work independently have occasionally reduced the value of individual experts.

The same is sometimes true of information and system studies provided under the support programmes. Information is not always complete and correct to the extent necessary and some safeguards studies are based on wrong technical or
political assumptions. In some cases it has been very
difficult to ensure that the studies were based on a realistic
assessment of what IAEA safeguards were and what could be done
in the framework of existing safeguards agreements.

Whereas good communication on the programme management
level is usually established through annual or semi-annual
coordination meetings, earlier and more frequent direct
communication (visits) between the researchers and the project
officers could certainly improve the success of some support
programmes.

As a last criticism, I would like to mention the sometimes
insufficient degree of flexibility in the management of some
support programmes. Although the Agency usually gets a fast
response to urgent requests, this is not always the case and I
would like, in this connection, to remind you of the old Latin
saying, "bis dat qui cito dat", meaning: you give twice if you
give fast.

Although I am expressing some criticism and suggesting
improvements, I do not wish to diminish the worth of the
excellent support the Agency has received through the
different support programmes. There is no doubt that IAEA
safeguards have been significantly improved by the outcome of
these programmes. The concepts, information, techniques or
instruments which have been provided by or improved through
national or multinational support programmes are used in the
course of the preparation, performance and evaluation of
practically every inspection.

CONCLUSIONS

IAEA safeguards have profited significantly from national
and international support programmes in recent years.
Important improvements in safeguards training, information
treatment, development and implementation of methods,
instruments and techniques and in safeguards effectiveness
evaluation have been made as a result of this support. It is
not claimed that the work is now complete and that IAEA
safeguards are now perfect. It is clear that more has to be
done to improve safeguards credibility, and optimizing the
system to accomplish this will require continuing support from
Member States. Nowadays, a successful international
safeguards system is an important factor in the unhampered
utilization of nuclear energy in peaceful activities. To
maintain and improve this system requires and justifies the
commitment of significant resources.
REFERENCES


SOME PROBLEMS RELATING TO APPLICATION OF SAFEGUARDS IN THE FUTURE.

By the end of this century there will have been a considerable increase in the amount of nuclear material and the number of facilities subject to IAEA safeguards. The IAEA will therefore be faced with problems due to the increased volume of safeguards activity, the application of safeguards to new types of facility and to large facilities, the optimization of the existing IAEA safeguards system and so on. The authors analyse the potential growth in the IAEA’s safeguards activities up to the year 2000 and consider how to optimize methods for the applications of safeguards, taking into account a number of factors relating to a State’s nuclear activity, the application of full-scope IAEA safeguards etc. On the basis of a hypothetical model of the nuclear fuel cycle that allows for the factors considered as part of the International Nuclear Fuel Cycle Evaluation (INFCE), the authors assess the possible risk of diversion as a function of a full-scope safeguards effort. They also examine possible conceptual approaches to safeguarding large-scale facilities such as fuel reprocessing and uranium enrichment plants.

НЕКОТОРЫЕ ВОПРОСЫ ПРИМЕНЕНИЯ ГАРАНТИЙ В БУДУЩЕМ.

К концу текущего столетия количество ядерных материалов и число установок, к которым будут применяться гарантии МАГАТЭ, значительно увеличится. В связи с этим МАГАТЭ предстоит решать проблемы, связанные с увеличением объема контрольной деятельности, применением гарантий на новых типах установок и крупномасштабных установках, оптимизацией существующей системы гарантий МАГАТЭ и др. В докладе анализируется возможный рост деятельности МАГАТЭ по гарантиям до 2000 г. Рассматриваются пути оптимизации подходов к применению гарантий с учетом ряда факторов, имеющих отношение к ядерной деятельности государства, полноты применения гарантий МАГАТЭ и др. Авторами на основе гипотетической модели ядерного топливного цикла с учетом элементов, которые были рассмотрены в рамках Международной оценки ядерного топливного цикла (МОЯТЦ), оценивается возможный риск переключения в зависимости от условий полноты применения гарантий. Проводится рассмотрение возможных элементов концепции по применению гарантий на крупномасштабных установках — заводах по химпереработке топлива и изотопному обогащению урана.
Расширение деятельности МАГАТЭ по гарантиям в ближайшие 10-15 лет будет связано с развитием ядерной деятельности в основном в нескольких неядерных государствах.

Во-первых, увеличится количество ядерных материалов и число установок того же типа, что и те, к которым уже применяются гарантии в настоящее время. Это вряд ли потребует существенных изменений в существующих процедурах применения гарантий. Однако дальнейшее развитие и использование более совершенных приборов и методов измерений, улучшенных средств сохранения и наблюдения, введение в конструкцию установок элементов, облегчающих применение гарантий, позволит осуществлять более оптимальные и эффективные гарантии.

Во-вторых, оптимизация существующих подходов к применению гарантий на основе более глубокого рассмотрения и учета, в частности, таких факторов, как стратегия переключения, характеристики ядерного топливного цикла и международная взаимозависимость в ядерной деятельности, позволит более четко определить цели применения гарантий, приведя их в соответствие с реальными условиями наличия возможности создания ядерных взрывных устройств.

Далее, под гарантиями МАГАТЭ будет находиться ряд установок, в отношении которых Агентство имеет еще недостаточный опыт осуществления гарантий или вообще не имеет такого вот вида. К первой категории установок относятся быстрые реакторы-размножители, заводы по центробежному обогащению урана и химпереработке отработавшего топлива, ко второй — крупномасштабные заводы по диффузионному обогащению урана и химпереработке, заводы по производству тяжелой воды.

Задачи контроля таких установок, по нашему мнению, могут быть решены на основе существующих мер гарантий, а именно использования мер материального учета и сохранения/наблюдения. Оптимизация системы гарантий, о чем сказано выше, а также количественный учет эффективности применения мер сохранения и наблюдения позволят получить более высокий уровень уверенности в непереключении ядерных материалов.

1. ПЕРСПЕКТИВЫ ПРИМЕНЕНИЯ ГАРАНТИЙ

Современные прогнозы в области развития ядерной энергетики в неядерных странах показывают, что общая мощность энергетических реакторов составит в 1990 г. 228-305 ГВт(эл.), а в 2000 г. — 532-774 ГВт(эл.) [1-3]. Указаные мощности будут получены в основном за счет использования легководных и тяжеловодных энергетических реакторов [3]. Можно предположить, что каждые десять лет число реакторов, находящихся под контролем МАГАТЭ, будет удваиваться.

Существующие мощности заводов по изотопному обогащению урана позволят обеспечить производство топлива для легководных энергетических реакторов пример-

На основе методики, предложенной секретариатом МАГАТЭ [5], нами были сделаны грубые оценки предполагаемых инспекционных усилий МАГАТЭ вплоть до 2000 г. в неядерных странах с учетом вышеуказанных прогнозов в развитии ядерного топливного цикла. Так, согласно этим оценкам, основной рост усилий будет связан с проведением инспекций на установках по химпереработке, обогащению урана и изготовлению топлива. В 1990 г. всего потребуется от 11 000 до 16 000 человеко-дней инспекции МАГАТЭ, а в 2000 г. — от 18 000 до 24 000 человеко-дней. Это будет соответствовать штату в 160-230 инспекторов в 1990 г. и 260-350 — в 2000 г. При оценке инспекционных усилий в будущем следует учитывать также дальнейшее расширение сети региональных подразделений МАГАТЭ по гарантиям, расположенных в местах наиболее интенсивного инспектирования, и развитие прогрессивных методов контроля, что может привести к относительному уменьшению потребностей в штате инспекторов.

В настоящее время МАГАТЭ разработало и продолжает совершенствовать процедуры применения гарантий практически для всех установок ядерного топливного цикла [6]. Принципы применения гарантий, которые совершенствуются Агентством могут более эффективно применяться на практике, если в конструкции установок будут вводиться элементы, облегчающие их применение.

2. ОПТИМИЗАЦИЯ СИСТЕМЫ ГАРАНТИЙ

Система гарантий МАГАТЭ, как она изложена в Уставе МАГАТЭ и документах \textit{INFCIRC/66/Rev.2 и INFCIRC/153}, представляется нам достаточно гибкой и дает широкие возможности для дальнейшего технического совершенствования и оптимизации. Для оптимизации критериев и целей гарантий может быть использована их зависимость от следующих параметров, которые, в частности, оказывают влияние на сценарии переключения:
— степень полноты применения гарантий МАГАТЭ к ядерной деятельности государства;
— характеристики ядерного топливного цикла;
— реальные возможности государства по производству ядерных взрывных устройств;
— международная зависимость государства в ядерной деятельности и, в частнос-
ти, объем, в котором ядерный материал получается из других стран или направ-
ляется в них;
— эффективность государственной системы учета и контроля;
— влияние переключения ядерных материалов на функционирование ядерного
tопливного цикла;
— возможности по корреляции и взаимозависимость деятельности по проверке
на ядерных установках, включая импорт и экспорт.
Учет указанных параметров приводит к определенной градации основных крите-
риев и целей гарантий. Так, величины времени конверсии, определенные Секретариа-
tом МАГАТЭ, должны рассматриваться как минимальные при условии наличия всех
необходимых установок и проведения испытаний технологических процессов для пе-
ревода исходной формы ядерного материала в металлические компоненты ядерного
взрывного устройства.
Выбор оптимизированной величины времени конверсии, естественно, приведет
к соответствующим изменениям и величине времени обнаружения.
Для достижения оптимальных результатов в осуществлении гарантий МАГАТЭ
большое значение приобретает анализ стратегии и сценариев переключения ядерных
материалов, а также возможности их последующего использования для создания
взрывных устройств. В связи с этим было бы целесообразно введение фактора пере-
ключения (F), который включает два основных элемента — "возможность переклю-
чения" (d) и "возможность использования" (u) переключенного ядерного материала:

\[ F = d \cdot u \]

В работе [7] были рассмотрены вопросы практического использования этого фак-
тора для определения важности различных моделей ядерного топливного цикла с точ-
ки зрения применения гарантий. При этом также параметрически учитывался такой
фактор, как полнота охвата гарантиями ядерной деятельности государства.
В данной работе указанный метод использован в целях сравнения инспекционных
усилей для выбранной модели ядерного топливного цикла.

3. ОСНОВНЫЕ ЭЛЕМЕНТЫ ГАРАНТИЙ
ДЛЯ МОДЕЛИ ЯДЕРНОГО ТОПЛИВНОГО ЦИКЛА

С учетом изучения в рамках МОЯТЦ [1-4] нами рассматривается гипотетическая
модель ядерного топливного цикла (ЯТЦ) и его расширение в процессе 25-летнего пе-
риода функционирования.
Модель топливного цикла включает заводы по изотопному обогащению урана,
конверсии и изготовлению топлива из низкообогащенного урана; легководные реак-
торы (ЛВР); долговременное хранение отработавшего топлива; завод по химпера-
работке; быстрые реакторы-размножители (БРР); завод по изготовлению смешанно-
го уран-плутониевого топлива.
Исходный ЯТЦ включает 15 легководных реакторов ("начальная точка"). Каждые два года вводятся три ЛВР мощностью 800 МВт(эл.) каждый. Через пять лет после "начальной точки" в эксплуатацию вводится опытный быстрый реактор (300 МВт(эл.)), далее через 10 лет — промышленный быстрый реактор мощностью 600 МВт(эл.), еще через 5 лет — реактор мощностью 800 МВт(эл.) и далее каждые 5 лет — по два реактора мощностью 1000 МВт(эл.) каждый. Природный уран в виде гексафторида поступает из-за рубежа. Производительность балк-установок определяется потребностями в топливе для энергетических реакторов. Отработавшее топ-
ливо с ЛВР после двухлетней выдержки в бассейне реактора поступает в долговременное хранилище и на химпереработку. Отработавшее топливо быстрых реакторов не перерабатывается, а направляется на хранение после двухлетней выдержки в бассейне реактора.

На рис. 1 приведены потоки и инвентарные количества ядерных материалов в ядерном топливном цикле (ЯТЦ) в зависимости от времени его функционирования.
Как видно из рисунка, проверяемые потоки отработавшего топлива изменяются в течение 25 лет в значительно меньшей степени, чем инвентарные количества. Что касается свежего топлива и других форм ядерных материалов, то здесь увеличение потока и инвентарных количеств примерно пропорционально. Анализ показывает, что основное увеличение инспекционных усилий будет связано с проверками потоков ядерных материалов в ЯТЦ, т.к. инвентарные количества ядерных материалов будут находится в соответствующих зонах хранения в основном в виде отдельных единиц (ТВС, контейнеры), большая часть которых опечатана, что упрощает действия по проверке. Основная проблема при проверках инвентарных количеств будет связана с выделенным плутонием, который содержится в ТВС, контейнерах, рассыпной форме, топливных элементах и т.d.

При расчете инспекционных усилий нами были использованы данные по росту производительности балк-установок и числа энергетических реакторов. Как видно из рис.2, рост инспекционных усилий в основном определяется увеличением производительности заводов по химпереработке отработавшего топлива и изготовлению смешанного уран-плутониевого топлива.

В табл.1 приведены расчетные данные инспекционных усилий для установок модели ЯТЦ, полученные на основе методики МАГАТЭ [5].

Как уже упоминалось выше, введение фактора переключения дает возможность количественно определить важность ЯТЦ с точки зрения риска переключения ядерных материалов и тем самым определить степень концентрации инспекционных усилий.

Следуя работе [7], мы рассмотрели сценарии переключения для двух случаев: 1) помимо установок модели ЯТЦ, находящихся под контролем МАГАТЭ, имеются установки по химпереработке отработавшего топлива, на которые не распространяются гарантии Агентства; 2) учитывается возможность существования незаявленных установок по химпереработке отработавшего топлива.

Расчеты показывают, что отношение факторов переключения для указанных случаев составляет $F_1/F_2 = 1.81$. Как известно [7], фактор переключения зависит от таких параметров, как время обнаружения, значимые количества, вероятность обнаружения, которыми в свою очередь определяются инспекционные усилия. Вариация указанных параметров позволяет получить одинаковые величины факторов переключения для 1-го и 2-го случаев и тем самым оптимизировать инспекционные усилия. По нашим оценкам соотношение инспекционных усилий для указанных случаев будет составлять $\approx \left(\frac{F_1}{F_2}\right)^{1/3}$. В таблице приведены оценки суммарных инспекционных усилий, которые примерно на 20% меньше для 2-го случая, чем для 1-го случая.

В случае применения указанного выше подхода для модели ЯТЦ, находящегося под гарантиями МАГАТЭ и включающего легководные реакторы и долговременное хранилище, ожидается, что инспекционные усилия для 2-го случая будут примерно в 1,5-2 раза меньше, чем для 1-го случая, при условии возможности непрерывного функционирования систем сохранения и наблюдения в течение года.
4. ГАРАНТИИ НА КРУПНОМАШТАБНЫХ УСТАНОВКАХ

Наиболее важными с точки зрения применения гарантий в будущем являются крупномасштабные установки по химпереработке отработавшего топлива и изотопному обогащению урана.

4.1. Заводы по химпереработке отработавшего топлива

В некоторых странах планируется построить заводы по химпереработке отработавшего топлива производительностью 1500 т топлива в год [1]. Основной проблемой по контролю этих установок считаются ограниченные возможности измерительной техники, не позволяющие сделать удовлетворительное заключения о переключении на основе только мер учета. В последнее время был предложен ряд концепций по применению гарантий на крупномасштабных установках [8-14], элементы которых могут быть использованы для разработки систем контроля для таких установок.

При определении процедур гарантий для крупномасштабных заводов по химпереработке следует дифференцировать проверки ядерных материалов, находящихся в виде отдельных единиц (кассеты, контейнеры), и материалов в балк-форме.


Ядерные материалы в балк-форме обрабатываются в зоне технологического процесса. Анализ составляющих неопределенности баланса материалов [15,16] для модели завода по химпереработке отработавшего топлива производительностью 1500 т топлива в год показывает, что основной вклад в неопределенность полугодового закрытия баланса материалов обычными методами учета происходит от измерений количества плутония в баке учета (σ₅ ≈ 75 кг Pu) и в отходах (σ₅ ≈ 38 кг Pu — жидкие отходы, σ₅ ≈ 53 кг Pu — твердые отходы). Измерения на выходе зоны процесса завода производятся с более высокой точностью, что позволяет проводить достаточно точную проверку потока и инвентарного количества плутония (σ₅ ≈ 7 кг).

Как известно, в практике применения гарантий МАГАТЭ не всегда имеются технические возможности по измерениям ядерных материалов, обеспечивающие требуемые количественные заключения о переключении ядерных материалов. Поэтому возрастает роль применения мер сохранения и наблюдения. Одной из целей этих мер является обнаружение незаявленного перемещения ядерных материалов из зоны наблюдения. Наиболее успешно меры сохранения и наблюдения могут применяться, когда пути переключения ядерных материалов известны и сведены к минимуму, а сами материалы находятся в виде отдельных единиц.

Применение мер сохранения и наблюдения для ядерных материалов, находящихся в балк-форме и подвергающихся обработке, имеет свои особенности. Например, для заводов по химпереработке технологический процесс включает такие операции для отработавшего топлива, как передача кассет из зоны хранения в зону технологического процесса, резка и последующее растворение, отделение урана и
плутония от осколков деления и получение соединений урана и плутония в виде раствора или порошка.

Применение системы учета ядерных материалов во времени, близком к реальному, позволяет значительно улучшить результаты по неопределенностям закрытия материального баланса [17]. Для целей проверок данной системы потребуется практически постоянное присутствие инспекторов МАГАТЭ на установке, которые помимо независимых измерений будут проводить наблюдения за осуществлением операций, связанных с учетом материалов. Помимо наблюдений инспекторами МАГАТЭ должна осуществляться проверка применения мер сохранения с целью обнаружения незаявленного изъятия ядерного материала на каждой стадии зоны технологического процесса. Для эффективного применения мер сохранения и наблюдения необходимо в зоне процесса предусмотреть конструкционные элементы, которые облегчили бы применение этих мер.

Среди таких элементов необходимо предусмотреть специальную конструкцию помещений, в которых находится оборудование технологического процесса, и компоновку оборудования с тем, чтобы инспектор имел возможность удостовериться, что не существует какого-либо оборудования, применяемого в ходе технологического процесса, которое может быть использовано для незаявленного изъятия ядерных материалов.

Предположим, что на установках технологического процесса, начиная от резки кассет с отработавшим топливом до момента получения растворов продукта урана и плутония, применяются эффективные меры сохранения и наблюдения, позволяющие с высокой степенью достоверности обнаружить незаявленные изъятия ядерных материалов.

Согласно нашим расчетам, неопределенность измерений в баке учета смоделированного завода по химпереработке составляет $\sigma = 75$ кг плутония за полгода, а общая неопределенность закрытия баланса материалов обычными методами учета для всего завода равна $\sigma_{MUF} \approx 101$ кг Pu, что значительно выше значимого количества SQ для плутония ($8$ кг).

Чувствительность обнаружения переключения зависит от вероятности обнаружения при помощи всех применяемых мер гарантий, которая может быть выражена формулой $(1 - \beta_a \cdot \beta_c \cdot \beta_s) = (1 - \beta_a \cdot \beta_c \cdot \beta_s)$ [18], где $\beta_a$, $\beta_c$, $\beta_s$ — вероятность необнаружения переключения, соответственно при помощи мер учета, сохранения и наблюдения.

Задача количественного учета влияния мер сохранения и наблюдения на чувствительность обнаружения еще не решена, однако очевидно, что эти меры повышают чувствительность.

Предположим, что на стадиях технологического процесса применяется система учета материалов во времени, близком к реальному, дополненная системой сохранения и наблюдения, которая включает, в частности, пломбы, мониторы, радиационные детекторы. При помощи каждого такого инструмента можно обнаружить незаявленное изъятие определенного количества ($m$) ядерного материала с некоторой вероят-
номостью (р) в течение времени (Т). Согласно работе [19], в случае применения пломб изъятие ядерного материала (m > 0) может быть обнаружено с достаточно высокой вероятностью (р ~ 0,999) в течение времени, равного интервалам проверок целостности пломб. Мониторы и детекторы обеспечивают определенную вероятность обнаружения изъятия, которая возрастает с увеличением количества изъятого ядерного материала [19].

В зависимости от конструкции эти инструменты могут иметь очень высокую чувствительность (m < SQ) при высокой вероятности обнаружения (р ~ 0,99) не-заявленного изъятия плутония в течение длительного интервала времени (T1 ~ 1 год). Аналогичные количественные данные могут быть определены для других инструментов системы сохранения/наблюдения, а также для всей системы в целом. В случае, если суммарная чувствительность обнаружения системы высока (\( \sum m_i < SQ \)), то это позволяет инспектору сделать заключение о переключении количеств плутония, меньших, чем величина значимого количества, в течение длительного промежутка времени (T ~ 1 год), несмотря на значительную величину компоненты неопределенности в закрытии баланса материалов.

Концепция применения гарантий на крупномасштабных заводах по химпереработке предусматривает полную проверку инспекторами измерений, которые оператор проводит в баке учета (измерение объема и концентраций, взятие образцов, калибрование измерительной системы оператора), т.е. на входе в зону процесса. Важное значение имеет проверка продуктов разделянных урана и плутония на выходе из зоны. Это необходимо для оценок различия данных отправителя и получателя [6, 20, 21], для анализа потоков ядерных материалов и их инвентарных количеств в зоне процесса, что наряду с данными мер сохранения и наблюдения используется при получении заключения о возможности переключения ядерных материалов.

4.2. Заводы по изотопному обогащению урана

Крупномасштабные заводы по обогащению урана, также как и заводы по химпереработке, являются наиболее сложными с точки зрения применения гарантий. Ядерно-энергетические программы [4] ряда стран предусматривают в будущем сооружение заводов по изотопному обогащению мощностью до 10 000 т ЕРР/год* для производства низкообогащенного урана, и некоторые из них могут быть предметом гарантий.

Крупномасштабные заводы будут в основном использовать газодиффузионный и центробежный методы.

Кроме того, получат развитие такие методы обогащения, как аэродинамический, центробежный со стационарной стенкой, химического обмена, лазерный и некоторые другие.

Для диффузионного завода производительностью 10 000 т ЕРР/год [22], предназначенного для наработки низкообогащенного урана, потоки будут составлять

* т ЕРР — тонны единиц раздельительной работы.
для входа 23 100 т UF₆ с содержанием 0,71% ²³⁵U (~ 1850 цилиндров), для продукта — 4000 т UF₆ с обогащением 3,0% по ²³⁵U (~ 1700 цилиндров) и для отвала — 19 000 т UF₆ с содержанием 0,25% ²³⁵U (~ 1500 цилиндров), причем в зоне процесса будет находиться примерно 5000 т UF₆ (флуктуация — 100 т). Закрытие баланса материалов может быть произведено с неопределенностью ~ 240 кг ²³⁵U без учета инвентарного количества внутри процесса.

Производство высокообогащенного урана на таком диффузионном заводе требует существенного изменения конструкции (увеличение числа ступеней обогащения), что в принципе может быть легко обнаружено [4].

Использование принципа рециркуляции потоков для производства высокообогащенного урана связано с проблемами критичности и даже в случае гипотетической возможности такого использования эксплуатационные характеристики завода окажутся отличными от нормальных (уменьшение выхода продукта и степени его обогащения).

Завод по обогащению урана с использованием центробежного метода производительностью в 3000 т ЕРР/год [23] имеет следующие потоки урана: питание — 8500 т UF₆ с содержанием 0,71% ²³⁵U (680 баллонов), продукт — 1300 т UF₆ с обогащением 3,0% по ²³⁵U (570 баллонов), отвал — 7140 т UF₆ с содержанием 0,3% по ²³⁵U (570 баллонов). Неопределенность закрытия годового баланса материалов для такого завода может составлять 90 кг ²³⁵U, что по величине несколько больше одного значимого количества (75 кг ²³⁵U). В зоне процесса завода находится незначительное количество ²³⁵U (несколько килограммов).

По мнению экспертов [4], завод по обогащению урана с использованием центробежного метода может быть использован для производства урана с более высоким обогащением, чем предусмотрено, как путем перестройки соединений каскадных элементов, так и за счет рециркуляции; при этом учитываясь ограничения, накладываемые требованиями по критичности.

В настоящее время имеется ряд предложений по применению гарантий для газодиффузионных и центробежных обогатительных заводов. Проблемы контроля таких заводов, в том числе и крупномасштабных, в основном относятся к степени доступа инспекторов к обогатительным каскадам [23, 24]. Степень доступа при водит к различным требованиям по инспекционным усилиям, применению измерительной техники и средств сохранения и наблюдения. Анализ показывает, что эффективность систем контроля в этих случаях различна. Так, при отсутствии доступа инспекторов в каскадный зал центробежного завода процедуры контроля должны предусматривать не только проверки линий питания, продукта отвала и отходов, но и оборудования, средств транспортировки и персонала, покидающего каскадный зал, что потребует применения опечатывания и дополнительного мониторирования во всех местах проникновения в каскадный зал и приведет к увеличению инспекционных усилий.

Отсутствие какого-либо доступа инспектора в каскадный зал делает проблематичным достижение цели гарантий по обнаружению незаявленного производства
высокообогащенного урана на заводе по центробежному обогащению, который предназначен для наработки только низкообогащенного урана.

Обнаружение удаления произведенного значимого количества высокообогащенного урана из каскадного зала также является достаточно трудной задачей при существующей технике проверок. Кроме того, возникают серьезные организационные по проверке информации о конструкции.

До настоящего времени еще не решены проблемы применения гарантий для заводов по изотопному обогащению урана с использованием диффузионных и центробежных методов. По мнению авторов, одной из возможных концепций контроля заводов по изотопному обогащению урана является ограниченный доступ инспекторов в каскадный зал для целей проверок, при котором должны предусматриваться меры, обеспечивающие невозможность получения инспектором важной в коммерческом отношении информации.

Основой системы гарантий для обогатительных заводов должно быть использование мер учета, а также мер сохранения и наблюдения с проверкой закрытия баланса материалов, т. е. мер, аналогичных для заводов малой и средней мощности.

Целями гарантий для заводов по обогащению урана центробежным методом являются:

— своевременное обнаружение с высокой степенью достоверности переключения значимых количеств урана;
— обнаружение незаявленного производства высокообогащенного урана в течение времени, необходимого для производства значимого количества урана.

При разработке процедур гарантий целесообразно учитывать следующие элементы:
— организация зон баланса материалов может предусматривать создание специальной зоны, как это указано в п. 46 "в" документа INFCIRC/153;
— каждый каскадный зал представляет по крайней мере отдельную зону баланса материалов;
— инспектор должен иметь возможность проверки информации о конструкции как в период строительства завода, так и во время его работы, в том числе для целей получения достоверных сведений о полной схеме потоков ядерных материалов внутри каскадного зала;
— инспектора должны иметь в распоряжении измерительное оборудование, необходимое для анализа изотопного состава и содержания 235 U, портативные системы обработки данных;
— надежные системы оптического наблюдения, датчики потоков UF6, пломбы, системы мониторирования для использования в местах проникновения в каскадный зал.

Требуется разработка измерительных систем, использующих неразрушающие методы анализа, для обнаружения производства высокообогащенного урана, в частности путем детектирования нейтронов, индуцированных альфа-распадом 234 U, содержание которого увеличивается при более высоких степенях обогащения по 235 U.
ЛИТЕРАТУРА

RECENT DEVELOPMENTS IN THE IMPLEMENTATION OF EURATOM SAFEGUARDS

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Abstract

RECENT DEVELOPMENTS IN THE IMPLEMENTATION OF EURATOM SAFEGUARDS.

The EURATOM safeguards system is based legally on the 1958 Treaty of Rome establishing the original Community of six (now 10) countries. Under this safeguards system, the Commission has, inter alia, “to satisfy itself that any particular safeguarding obligations assumed by the Community under an agreement concluded with a third state or an international organisation are complied with” (art. 77b). The practical implementation of safeguards within the Community is significantly influenced by the requirements of: (a) the three different agreements between the Community, its Member States and the IAEA, concerning the application of IAEA safeguards to some or all of the civil nuclear materials in the Community, and (b) the various agreements between the Community and certain third countries, concerning inter alia the application of safeguards within the Community to nuclear materials supplied, directly or indirectly, by these third countries. Within the past four years significant developments have occurred in both groups of agreements. The NPT (Non-Nuclear-Weapon State) Agreement of course continues in force, with the inclusion since 1981 of Greece in the group of 8 NNWS Member States party to the Agreement; the EURATOM/UK/IAEA Agreement which has been negotiated following the British “Voluntary Offer” in the context of the NPT entered into force in 1978; the EURATOM/France/IAEA Agreement concerning the implementation of IAEA safeguards in France entered into force in 1981. An Agreement between Australia and the Community was signed and came into force in 1981, and a further amendment to the EURATOM/Canada Agreement also came into force in 1981. The EURATOM safeguards organisation is the only multinational safeguards organisation in the world, and currently has a staff of some 120 inspectors, with appropriate administrative support, and can draw for research and development work on the resources of the Community’s Joint Research Centre. The recent changes in inspection techniques, particularly in relation to non-destructive assay techniques, and the implementation of containment and surveillance measures, are discussed. A description is given of the experience gained in recent years in the operation of “Joint Teams” of EURATOM and IAEA inspectors in certain plants as well as the continuing experience gained under the normal regime, using the observation principle, as foreseen in the respective Agreement.

1. Introduction

The purpose of this paper, in the 25th year of existence of the European Atomic Energy Community (EURATOM), is to give an aperçu of the developments which have taken
place in the Euratom Safeguards system since the last major IAEA Safeguards Symposium four years ago in 1978.

The Euratom safeguards system applies to all non-military nuclear activities in the ten Member States of the Community (the same ten Member States being equally the partners in the European Iron and Steel Community, founded in 1951, and the European Economic Community founded in 1957). The multinational safeguards system is based on the Treaty establishing the Community and is quite unique, dealing as it does with ten member states, two of which pursue military nuclear programmes, eight of which are "Non-Nuclear-Weapon States" and have renounced nuclear weapons research or production, and nine of which are signatories to the Non-Proliferation Treaty.

The Euratom safeguards system is closely related to the IAEA safeguards system, both systems having similar objectives in the nonproliferation sphere and working together in a close and continuing cooperation. The technical aspects of the cooperation are reflected in the subsequent sections of this paper which deal with some of the developments which are apparent over the period under consideration.

This period has seen, as might be expected in view of the planned increase in installed nuclear generating capacity to over 100 GW by 1990, a considerable increase in the amount of nuclear material subject to Euratom safeguards. These currently amount, in rounded figures, to:

- Plutonium: 50 tonnes
- Highly-Enriched Uranium: 13 tonnes
- Low-Enriched Uranium: 14,000 tonnes
- Natural Uranium: 40,000 tonnes
- Depleted Uranium: 32,000 tonnes
- Thorium: 1,500 tonnes
- (Heavy Water: 440 tonnes).

The comparison with the world-wide totals for materials under IAEA safeguards is left as an exercise for the interested reader. There has not in recent years been a proportionate increase in the man-power resources made available to the Safeguards Directorate; certain economies of scale have worked in our favour, while the increase in the technical complexity of the safeguards implementation activities has operated in the opposite
sense, requiring additional effort to be applied. There is currently a staff of some 120 Euratom Safeguards Inspectors based in Luxembourg; they are career officials of the European Communities, and drawn from all 10 Member States (the IAEA has about 135 operational inspectors). The Directorate shares the appropriate administrative and logistic services available in Luxembourg and Brussels, and can draw for research and development work on the resources of the Community's Joint Research Centre, mainly the Ispra, Karlsruhe and Geel establishments. In addition, through the Commission's important role in ESARDA, the European Safeguards Research and Development Association, the Safeguards Directorate can draw on the coordinated research and development programmes of the laboratories within the Community.

2. Developments in the legal framework

At the time of the last major IAEA Safeguards Symposium in 1978 all three Euratom Community agreements with the IAEA had been signed, the NPT Agreement with the (then 7, now 8) Non-Nuclear-Weapon States, in 1973, the "Voluntary Offer" NPT Agreement with the United Kingdom in 1976 and the Agreement with France in July 1978. The Non-Weapon State Agreement was already in force and routine IAEA inspections were taking place, as foreseen by the Agreement, during some of the Euratom inspections. The intervening years have seen continuous progress toward the complete implementation of all three Agreements.

In 1978 certain "compromises" were reached between the services of the Commission and of the Agency in order to expedite the negotiation of Facility Attachments. The most significant of these related to the establishment of "Joint Teams" of Euratom and IAEA inspectors to perform the inspection activities in plants handling significant quantities of highly enriched uranium or plutonium or performing reprocessing or enrichment. By avoidance of unnecessary duplication and by providing mutual support each organisation expected to economise in its use of the restricted manpower available, while fulfilling the necessary safeguards tasks and allowing each organisation, from a common data base, to draw its own conclusions. In due course Facility Attachments were agreed and entered into force for all the installations in the Non-Nuclear-Weapon States with the
exception of about a dozen where it was agreed with the IAEA that the time was not yet ripe for the formalisation of the documents governing routine inspections (the IAEA, having rather wide inspection rights during the interregnum, by virtue of Article 71(a) of the Agreement).

In 1980 the Subsidiary Arrangements to the EURATOM/IAEA/UK Agreement came into force, and the full reporting procedures relating to all civil nuclear materials in the UK, already operative for Euratom safeguards, were implemented through onward transmission to the IAEA. Two facilities in the fast reactor fuel cycle, a fast prototype power reactor and its associated reprocessing plant, were designated by the IAEA for the application of routine inspection activities. Both these facilities, on the same site, have been inspected by Joint Teams.

The EURATOM/IAEA/FRANCE Agreement came into force on 12 September 1981 and the Subsidiary Arrangements have been agreed and are due to enter into force shortly. Reporting to the IAEA of the nuclear materials subject to this Agreement has been operative since December 1981.

The Community, since it has only limited indigenous resources of uranium, is substantially dependant upon outside suppliers for its requirements of nuclear fuel, and in the interest of assuring the long term reliability of its external supplies in a non-proliferation context has pursued a policy of entering into Agreements for cooperation and supply.

During the recent period the two most significant developments have related to the negotiation of the Community Agreement with Australia and of the most recent (1981) amendment of the Community's Agreement with Canada. The Agreement with Australia is somewhat unusual in that it deals only with transfers of nuclear material from Australia to the Community; it is not reciprocal. Furthermore it takes cognizance of the need for reliable and assured supplies for the Community's nuclear power programme, the widespread nature of the facilities involved in the nuclear power programme, and the need to reprocess irradiated fuel both for reasons of fuel management and of effective fuel utilisation. In the associated arrangements due heed is paid, by
acknowledgement of the principles of equivalence and proportionality, to the need to permit economic exploitation of the nuclear fuel and to the avoidance of a proliferation of potentially different and even conflicting safeguards obligations on materials which are required to be processed together.

The 1981 amendment of the EURATOM/Canada Agreement saw an end to a period of application of the temporary provisions envisaged by the 1978 amendment to the same Agreement, and allows the Community fuel cycle operators to plan ahead on a firmer basis for their long-term commitments in respect of fuel preparation, use and reprocessing over the life-span of the reactors.

These two Agreements now permit the Commission to authorise certain simplifications in the procedures for accounting "by obligation" for fuels while in certain process steps where the economic imperatives prevent an atom by atom segregation of the materials of different obligation.

In the present forum it should perhaps be emphasised that these "obligation accounting" questions are internal to the Euratom system, reflecting specific Community obligations to third Countries, the material accounting in the IAEA context relating to the totality of all nuclear materials, without knowledge of its "obligation", KMP by KMP and MBA by MBA.

3. Development in NDA techniques

The most interesting recent developments in NDA techniques for operational use are today related to the standardisation of measurement techniques.

The word "standardisation" here does not mean that a given method or procedure has to be adopted and used as the unique, only acceptable, method since it is commonly agreed that there are good reasons to enlarge the range of different radiometric measurements for nuclear material accountancy. The word standardisation is meaningful only where the fundamentals of any particular technique have been fully examined and are understood and considerable experience has been accumulated from its practical application. The outcome of a standardisation procedure is to allow the users, through experience and knowledge of the technique, to develop a uniform basis enabling them to judge the adequacy and applicability of a particular method for their measurement problem.
Radiometric assays in general are not absolute measurements and in consequence have to be calibrated by the use of reference samples. In the nuclear fuel cycle the variety of configurations of materials to be measured is however so large that the procurement and the availability of directly representative samples for calibration represent a continuing problem, not yet satisfactorily resolved. Efforts have been made by the safeguards authorities in recent years, in collaboration with the plant operators, in the preparation of the most representative reference materials. From this collaboration a new category of reference samples has been created, the so-called "plant specific reference materials".

Preparation schemes are currently underway for the fabrication of:

(1) light water reactor fuel rods,
(2) mixed oxide fuel rods,
(3) low enriched uranium powder,
(4) highly enriched uranium/thorium high temperature reactor fuel pebbles,
(5) platelets for fast critical assembly fuel, and
(6) plates for HEU MTR elements.

Studies are also under way for the reduction of the need for more reference materials through the analysis of the details of the detection processes and through more precise knowledge of the physical parameters.

The increase in the number of inspectors as a consequence of the constant increase of the verification activities in recent years, related to the major expenditure of effort in non-destructive measurements constituted a further reason for the review and possible revision of the previous methodology used by the Safeguards Authorities. An unlimited increase in the stock of instruments, bought on the commercial market, would have resulted in a proliferation of different standards for the instruments which could have led to serious problems of maintenance and spare-parts as well as an increased load in the training of inspectors.

The transfer of written measurement results to headquarters and their "digestion" by the central organisation would also have required an increase in personnel. As a result of the awareness of these problems, the development of the measurement instruments, made essentially by the JRC at Ispra for the Safeguards Directorate, was oriented towards the introduction of automation which on the one hand ensures reproducibility of the inspectors' operating mode
(through conversational procedures) and on the other hand allows the recording of all the data from a measurement campaign on an electronic support (magnetic cassette), simplifying the transfer and data evaluation at the headquarters.

The standardisation of instrumentation has obvious advantages on a local basis for interchangeability of subsystems maintenance and optimisation of the development effort.

The introduction of automated instrumentation which is to be integrated in a more general data acquisition, treatment and storage system has required the preparation of, and agreement on, standards for at least part of the existing instruments. A certain number of systems following this standard have already been built. These are: Phonid II, automated gamma spectroscopy systems with dedicated units for different applications (enrichment measurements and rod scanning system), IRIS standard terminals and IRIS tape duplicating device for inspections conducted by Joint Teams from the two different safeguards authorities, and a microcomputer for the SIGMA system.

Another application is under way in collaboration with the KFK at Karlsruhe for the specific data evaluation software for plutonium isotopic determination by gamma spectrometric measurements. Efforts are also being applied to the upgrading of the other existing instruments (mostly neutron instruments).

The data evaluation obtained via the dedicated microprocessor units allows the inspector to make a decision on the basis of an in-field data evaluation in certain cases of potential abrupt diversion. The central storage facility (or NDA data base) which collects the data from the magnetic cassette should allow retrieval, processing and the correlation from different measurement techniques. It is also important for the quality control of the instruments.

4. Developments in Containment and Surveillance

1. The main achievements in the last two years have resulted from the considerable experience gained in routine operation. Improved technical and administrative procedures yield better efficiency, both quantitatively and qualitatively. The main elements of C/S techniques are still general purpose seals and film camera surveillance. The required throughput has roughly doubled during the last two years.
2. Before the commercial metal cup seals are used as general purpose seals they each receive a special unique marking through solder stains and scratches. The verification after use consists of checks on integrity of wire and seal and on the identity of the markings. The latter check has been greatly simplified and improved by the adoption of a Video system for comparison of the returned seal and the record of its original state. Presently Euratom processes about 12,000 seals per year.

3. About half of the seals are used for strictly Euratom Safeguards activities. The other half, the COMMON Seals, are used jointly by EUR and IAEA. The procedures for production, application, removal and verification have been established in such a way that both authorities can draw full conclusions from the application of the COMMON Seals. Originally these procedures were developed for the "Joint Team" plants. But the positive experience gained has led to the extension of their use to other types of plants under the normal inspection scheme on a case by case basis. Euratom is trying to establish the use of Safeguards seals also during the transport of nuclear material outside the European Community. A better collaboration between national and international authorities would be required for this to be fully effective.

4. In order to draw full advantage from the sealing technique many particular problems relating to the tamper resistance of the method of the attachment of general purpose seals to items have had to be solved. Many specific accessories have been designed to seal UF₆ cylinders, "bird-cages", trays for fuel pins, fuel elements in a zero power reactor, the main shield of power reactors, etc. Technical responsibility for the verification of the containment structures and the accessories rests, of course, with the inspector(s) in the field.

5. A relatively small fraction of the seals applied are other than general purpose seals. Unfortunately development projects for the improvement of Paper Seals have not been successful yet. The MTR rivet seals are awaiting in-situ verification capability in order to become as fully effective for safeguards purposes as is desirable.

Improvements have been made in the field of tamper-resistant identification of reference materials.
through macrophotography of external marks (e.g. fuel pins) and through ultrasonic signature of internal marks (e.g. $^{3}$H reference samples produced by BCMN).

6. Optical surveillance is based on the Twin Minolta film-camera unit. The Euratom Safeguards Directorate operates these units jointly with the IAEA and has introduced very strict quality control into the technical procedure. This led to a very high rejection rate; therefore very recently technical revision and maintenance were also introduced into the technical procedures. These procedures help to eliminate camera handling faults. A technical failure of the unit, e.g. an incomplete coverage of the surveillance period (loss of more than a few frames by both cameras) presently occurs in 4% of the surveillance periods. With these periods being normally 3 months (for LWPRs) the mean interval between failures of a Twin unit is about 6 years. Our technical services are quite proud of this result, achieved on what is basically a camera designed for the amateur market.

7. Many experiments have been carried out in connection with the routine camera work aiming for an improved film quality and for a reduced failure rate. The first aspect includes optimisation of the film processing with respect to the lighting contrast and the irradiation dose in particular plants. The latter aspect included experiments on the effect of low humidity for MFX films, which appears to have led to a blockage of the film in a number of cases. We are confident that further improvement will be possible. Currently the Directorate is preparing to introduce colour film for routine use because of its contribution to improved visual perception. Even though the colour deteriorates under high irradiation, the contrast of the colour film is considerably more radiation-resistant than for black and white film.

5. Experience of inspection with IAEA

Inspections carried out by joint teams of inspectors are a compromise to provide a working solution to a practical problem. The problem was recognised by Euratom and IAEA by the fact that international apprehension, concerning the dangers of diversions from "sensitive" installations, seemed considerable. On the other hand, as provided for in the Agreement, the IAEA had to profit by the multinational character of the Euratom Safeguards system.
The Council of Ministers of the European Community has endorsed the compromise on the joint teams for a certain period of time and extended its approval until autumn 1983.

The primary objective of joint team arrangements is that inspectors of each organisation work together as equal members of a team in mutual support - and mutual reliance - making efficient use of the inspection efforts and each having access to all inspection results.

Planning of joint team operations requires careful handling. This relates to the programming, the preparation of working papers and the sharing of work between the inspectors.

Pursuant to the terms of the safeguards agreements the normal work of inspection performed by both safeguards organisations follows the principle of observation. Under this operational scheme the IAEA participates at certain Community inspections in order to satisfy itself that the EURATOM inspections are carried out in an appropriate and agreed manner.

For those inspections carried out in the absence of IAEA inspectors the provision of timely and accurate inspection reports has been established. On the basis of these reports as well as the results of the inspection activities observed as carried out by EURATOM the IAEA may perform verifications on a statistical basis of those results in order to draw its safeguards conclusions.

In 1981 this mode of operation was carried out at some 200 installations of the Community under both IAEA and EURATOM Safeguards including, for example, some 30 power reactors. EURATOM spent about 800 man-days and IAEA about 500.

One result of the implementation of both the joint team and the observation principle is that the inspection practices of both safeguards organisations need to be made more transparent so that full use be made of the data generated and the other findings obtained.

Further effort will still have to be spent on the standardisation of methods, use of the agreed chemical and physical standards for calibration, the evaluation procedures, and safeguards conclusions and statements.

Planning of inspections involving the two safeguards organisations operating from different centres and under
different administrative rules needs to be given more attention and care. Based on the experience gained it is clear that this task of proper planning is of particular complexity since the operation of the nuclear facilities in the European fuel cycle involves quite naturally unexpected changes of the respective programmes. Therefore, the frequency of the planning and review meetings has been doubled in 1982 and the other communication links have been improved constantly.

6. CONCLUSIONS

Since the previous Safeguards Symposium in 1978 considerable progress has been achieved in the implementation of the safeguards agreements concluded between the European Community and the IAEA.

The experience gained during this phase of implementation, involving the conclusion of more than 200 facility attachments and establishing the fact that the Community receives by far the largest share of IAEA inspection effort (just under 50%), permits a careful review of the implementation practices in a spirit of cooperation.

During this time period the safeguards technology has progressed with reasonable speed and it seems necessary to devote considerable, if not increased, effort on the crucial questions of standardisation and harmonisation of instruments, techniques, evaluation and conclusions.

One of the most challenging issues in safeguards methodology is the establishment of acceptable goals and criteria for non-proliferation safeguards. EURATOM, on the basis of its experience gained over the years, will contribute to these deliberations in the appropriate fashion.

Finally, in view of its dual function, to operate with the IAEA in the field of non-proliferation safeguards and to fulfill its role as safeguards authority in the European Community, the EURATOM safeguards system will participate at all levels in international cooperation in safeguards and will ensure that the Community continues to be that area in the world providing maximum non-proliferation assurance.
APPROACHES TO SAFEGUARDS

(Session 2)
Chairman

D. GUPTA
RECENT ADVANCES IN
IAEA SAFEGUARDS SYSTEMS ANALYSIS

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Abstract

RECENT ADVANCES IN IAEA SAFEGUARDS SYSTEMS ANALYSIS.

Efficient implementation of effective safeguards, the objective of the IAEA's Department of Safeguards, would be unthinkable without carrying out systematic studies on many different problems related to technical and other aspects of safeguards. The System Studies Section of the Department concentrates its efforts on such studies with the purpose of elaborating concepts, criteria, approaches and rules for the implementation of safeguards. In particular, the Section elaborates concepts and approaches for applying safeguards at the complex facilities that are expected to enter under safeguards in the future, develops approaches and rules in the areas where the Agency is still gaining experience, and assists in the implementation of safeguards whenever problems requiring non-routine solutions arise. This paper presents examples of the present activities of the System Studies Section: development of guidelines for use by facility designers in order to make safeguards easier and more effective, studies on near-real-time material accountancy, preparation of safeguards approaches for specific facility types, preparation of model inspection activity lists for different facility types and alternative safeguards approaches and preparation of safeguards policy papers containing the rules and regulations to be followed in the design and implementation of safeguards.

1. INTRODUCTION

The System Studies Section within the Department of Safeguards was established in 1967. Its role during this fifteen-year period has gradually changed. Some early functions still remain, e.g. the forecasting of nuclear fuel cycles and inspection requirements in future years. Others, such as the development of an information system, or the development of statistical procedures for the verification and evaluation of material accountancy data, have been assigned to new Divisions or Sections.

Today, in the American idiom, the System Studies Section is the "think tank" for the Department of Safeguards. Summarized very broadly, the Section's assignment is the development, evaluation and improvement of safeguards.
approaches for all types of nuclear facilities and materials and for fuel cycles.

For some types of facility, e.g. the light water reactor, the development of a safeguards approach is essentially complete. Current activities are related primarily to the development of model inspection activity lists, the application of a safeguards effectiveness assessment methodology, and the documentation of design features important to the implementation of the accepted safeguards approach. For other types of facilities, such as the large reprocessing plant, the centrifuge enrichment plant, or the heavy water production plant, the development of an effective and generally acceptable safeguards approach is still being studied, and detailed inspection activity lists would be premature. There are also a number of areas not specifically related to a single facility-type in which system studies work is in progress. Examples include policy papers on safeguards for non-nuclear materials, equipment, or technology, the development of written design guidelines for future nuclear facilities, and the development of a safeguards effectiveness assessment methodology for comparative studies of safeguards effectiveness.

2. DESIGN CONSIDERATIONS

The Agency is developing guidelines for use by facility designers in order to make safeguards easier and more effective. The importance of this activity has been expressed many times at various groups and discussions. The first consultants' meeting was held in June 1980; subsequent meetings occurred in December 1981 and September 1982. Design guidelines have been drafted for light water reactors, and work is in progress on guidelines for reprocessing facilities. It is hoped that work can begin soon on several other types of facilities.

It should be emphasized that recommended design features are intended only as guidelines and not as requirements for the implementation of safeguards. However, it is believed that the consideration of these guidelines would result in facility designs which are not only in the IAEA's interest, but are also in the best interest of both the State and the facility operator.

The guidelines are intended to contribute to the following objectives:
- Improvement and preservation of the quality of safeguards data
- Improvement of the Agency's ability to obtain safeguards data
- Improvement of the conditions under which inspections are carried out
- Reduction of the consumption of IAEA inspection resources without adverse effect on the effectiveness of Agency safeguards
- Reduction of the burden on facility operators.

3. NEAR-REAL-TIME MATERIAL ACCOUNTANCY

The interest of the System Studies Section in near-real-time (n.r.t.) material accountancy began in 1977, more or less coincident with the development of significant quantity and timeliness criteria which could not be satisfied by conventional material accountancy for large-scale facilities [1]. As with nearly all safeguards developments, the study of n.r.t. material accountancy has been a joint effort between the IAEA and various Member States, the Section's role being primarily one of coordination and evaluation.

In January 1982 the IAEA invited a group of consultants to meet in Vienna to review the current technical status of n.r.t. accountancy. Emphasis was placed on the word "technical"; the consultants were asked to consider what n.r.t. accountancy potentially could do, not what its role should be, in IAEA safeguards for current or future reprocessing facilities [2]. In summary, the consultants concluded that n.r.t. material accountancy could potentially improve both the sensitivity and the timeliness of material balance information. This was true, the consultants agreed, even in the limiting case in which observed anomalies could not otherwise be resolved, necessitating a clean-out physical inventory. The exact extent of the improvement in sensitivity and timeliness cannot easily be specified in advance, however, since it depends on the nature of the statistical tests used, the nature of the anomalies observed (if any), and the means used to resolve observed anomalies.

The consultants also agreed on a number of recommendations for further study, which emphasize the need to develop procedures for IAEA verification of operator-generated material balance data and the need for a full-scale demonstration under realistic operating conditions. The recommendations are being given serious consideration, and it
is hoped that a demonstration which includes IAEA verification procedures can be initiated during 1983.

4. MODEL INSPECTION ACTIVITY LISTS

Agency safeguards are applied according to safeguards agreements, subsidiary arrangements and facility attachments, documents which provide the necessary formal basis for implementing safeguards activities. To ensure that safeguards are implemented effectively, practical safeguards procedures have to be developed for each facility. These procedures should specify:

- what should be done during inspections
- when, or at what time intervals, this should be done
- how it should be done.

The first two points are unique to each facility under safeguards and their specification may be put the in the form of a list of inspection activities for the facility. The third point is addressed by means of instructions which generally need not be specific to a given facility (e.g. instructions for use of typical NDA equipment).

For practical reasons lists of inspection activities being prepared by the facility officers normally take into account only routine activities. Such Routine Inspection Activity Lists (RIALs) should reflect the safeguards approaches adopted within the Department of Safeguards as appropriate for facilities of given type and size and at a given time. This means that the inspection activities should be standardized for each facility type and size. The easiest way towards such standardization seems to be through the development of model RIALs, which might serve as guidance for preparation of individual RIALs [3]. This task has been allocated to the Section for System Studies. Once the model RIALs are developed, they also serve two other important purposes, namely as a reference for the evaluation of safeguards implementation, and as a training aid for new inspectors.

Model RIALs are being developed on the basis of analytical studies, in particular analyses of possible diversion paths. Diversion analyses have been made available for certain facility types through technical support programmes and are included in more general form in the model safeguards approaches developed in the Section for System Studies.
Unlike these model safeguards approaches, however, model RIALs have to reflect current inspection practices. They must also recognize the limitations imposed in the Facility Attachments which are actually in force for facilities of a given type.

The important elements of RIALs are the activity description, the type of inspection during which the activity should be carried out, the priority of the inspection activity, an indication of the anomaly which can be disclosed, and follow-up actions required if an anomaly is observed, together with their timing. Three levels of inspection priority are distinguished:

- **Mandatory**, which requires the activity to be carried out every time the appropriate inspection is performed.
- **High**, which requires the activity to be carried out if technically feasible and if there is no acute shortage of the manpower.
- **Intermediate**, which requires the activity to be carried out, if technically feasible, at some randomly selected inspections.

Indications of priorities, anomalies and follow-up actions may contribute to the optimization of the inspection effort and to further improvement of safeguards effectiveness.

5. SAFEGUARDS POLICY PAPERS

The Section for System Studies participates in the preparation of departmental policy on different issues related to the implementation of safeguards. Policy papers concerning the safeguards approach or specific implementation procedures are drafted in the Section on the request of the Regional Sections or other units within the Department of Safeguards where problems requiring policy definition are identified. Policy papers, subject to approval by the Deputy Director General for Safeguards, present rules which are to be followed in the implementation of safeguards. Concepts laid down in the policy papers are widely discussed with the participation of inspectors before they are submitted for final approval.

One problem resolved via a policy paper was the interpretation of the role the Material Balance Area (MBA) structure of a facility plays in establishing the material balance for Agency safeguards purposes. The policy on the MBA structure solved in a pragmatic way the question of the number
of MBAs required in large facilities. Considerations of the impact which the MBA structure may have on the accuracy of closing of the material balance on one hand and on the effort related to the reporting and subsequent verification of reports on the other, as well as exploration of the possibilities which are offered by appropriate determination of KMPs and other strategic points, have led to the formulation of flexible rules with respect to the selection of optimal MBA structures.

The policy on undeclared plutonium or $^{233}\text{U}$ production in research and power reactors has established requirements with respect to the adequate verification of design information. There is also a need for periodic re-verification, as well as implementation of safeguards measures to verify that the reactor is not being used for the production of undeclared plutonium or $^{233}\text{U}$.

A policy paper on the use of metal seals tackles, inter alia, the problem of periodic verification of sealed units. It recognizes that only removal and subsequent destructive verification of seals can be considered appropriate for verifying that the seal was not tampered with. The re-verification of sealed nuclear material is also envisaged whenever external examination of the containment structures cannot be fully conclusive.

Another policy area in which the System Studies Section has prepared guidance concerns the application of safeguards to non-nuclear material and equipment. The approach adopted was to specify the data and rights of inspection which must be provided to the Agency in order to assure that non-nuclear material and equipment subject to safeguards are "not used in such a way as to further any military purpose". While the precise types of material and equipment which are subject to safeguards are specified in INFCIRC/66-type safeguards agreements, for purposes of analysis it was found convenient to categorize items and materials in terms of their flow and inventory characteristics. An item or substance was placed in one group, called the "flow-item" group, if it goes through a nuclear or non-nuclear facility either alone or together with nuclear material. It was placed in another group, called an "inventory-item" group, if it is present in a facility for a long time (e.g. in relation to the life of the facility). As examples, zirconium pressure tubes would be categorized as inventory-items, but zirconium fuel cladding would be a flow-item. Graphite moderator would be an inventory-item, while graphite sleeves (a structural component of fuel elements) would be a flow-item. Pumps, reactor internals,
welders, etc. would be inventory-items, while heavy water would be a flow-item.

Broadly speaking, the data and inspection activities which were considered necessary are similar to those which are required for nuclear material. This is not surprising, since to verify that an item is not used for any military purpose the Agency must assure itself that the item is used for a safeguarded peaceful purpose, and, to this end, must have data on the location, form, quantity, composition and throughput of the item or other relevant information. As a consequence the inspection activities are also similar to those for nuclear materials.

With respect to termination of safeguards on these items, this was considered to be feasible when it was permitted by the agreement and when the item was no longer usable for any nuclear activity relevant from the viewpoint of safeguards or was practicably irrecoverable.

In addition, it was also considered that the Agency verification procedures that were otherwise required could be omitted in special circumstances - e.g., when because of the integrity of the physical combination of a flow-item with safeguarded nuclear material the flow-item is automatically accounted for, or when for an inventory-item the very fact of the operation of a safeguarded facility can be taken as conclusive evidence of the presence of the safeguarded inventory-item (so that, for example, no undeclared substitution of items could have occurred).

The ideas formulated by the System Studies Section on the application of safeguards to non-nuclear material and equipment appear to form the basis for an effective and efficient set of activities which would satisfy the obligations of the Agency in this regard. In applying safeguards, of course, the Agency has to take into account that non-nuclear material and equipment are "indirect use" items with respect to the manufacture of nuclear explosives.

6. SAFEGUARDS APPROACHES FOR SPECIFIC FACILITY TYPES

6.1. On-Load Reactors

On-load reactors differ from other reactor types in that refuelling is performed on a continuous basis during operation, whereas light water reactors are shut down for refuelling about once a year. As a consequence unique
safeguards problems arise requiring specific solutions. Safeguards system analyses for several types of on-load reactors have been performed by the System Studies Section with considerable support from the Canadian Support Programme to the IAEA. Results have been presented in various documents [4 - 7].

On the basis of these analyses, safeguards approaches have been developed for some reactor stations and others are being developed. The basic idea of these approaches is to derive the core inventory, which is from a practical point of view not routinely verifiable, and the spent fuel pond inventory, from the readings of a bundle counter monitoring spent fuel bundles passing through the transfer channel. Supplementary containment and surveillance measures are applied in order to preserve the recorded data. The implementation of the approaches has been completed in some cases and the first practical experience has been gained.

Based on these approaches and implementation experience, analyses are being made of possible improvements in safeguards systems for operating reactor stations. The goal is to reach a uniform level of safeguards performance in all on-load reactor stations.

6.2. Mixed Oxide Fuel-Element Fabrication Plants

In mixed oxide (MOX) fuel-element fabrication plants direct-use nuclear material is handled in bulk form. The safeguards approach for such facilities has to cover both abrupt and protracted diversion possibilities. A first outline of a safeguards approach was presented to the IAEA Safeguards Symposium of 1978. Thereafter a more detailed analysis was performed by the System Studies Section, focused on capabilities of nuclear material verification in the process line as a function of various parameters such as measurement errors [8]. The recommendations of an Advisory Group on this subject have also been taken into account, and the basic elements of the suggested approach are being implemented in facilities where the IAEA is applying safeguards. It is planned to discuss practical problems and limitations in due course in a further Advisory Group Meeting.

From the point of view of systems analysis one important problem remains, namely the evaluation of sequential interim inspection data. In this context, some work is being performed regarding the application of n.r.t. material accountancy methods to MOX plants in order to derive some estimate of detection capabilities.
6.3. Heavy Water Production Plants

In anticipation of the application of safeguards to a large, water-fed heavy water production plant, the System Studies Section has begun to analyse various safeguards approaches. The goal of these approaches would be to provide the Agency with an effective measurement and verification capability for the reactor-grade heavy water that is produced. This capability would be needed to permit the Agency to assure itself that there were no significant unrecorded withdrawals or removals of heavy water (or enriched, deuterated compounds) with concentrations above some target figure, and to permit the implementation of the approach with an effort that is consistent with that devoted to other safeguarded bulk handling facilities which process indirect use material.

Alternative approaches which are being examined include those related to both material balance and containment/surveillance (C/S). As an example of a material balance approach which is under consideration, the Agency would measure the flow and concentration of the feed and discharge water streams. The flow and concentration of the heavy water product stream would also be measured, and the plant inventory change and losses would be estimated from process data supplemented by measurements of those parts of the inventory which are accessible. Preliminary estimates indicate that a material balance based on these measurements could achieve a detection sensitivity of better than 20 tonnes D₂O equivalent through the use of commercially available instrumentation. Alternatives to this approach which involve measurements of the upflow and downflow of different enrichment stages appear to have similar sensitivities.

The C/S approach uses the same output measurement, plus a measurement of the feed to the deuterium burner. While the material balance approach would detect removals of deuterium through a discrepancy in MUF, the C/S approach would rely on seals and surveillance, as well as the intrinsic containment of the plant process vessels, to detect unrecorded removals of enriched deuterium or deuterium compounds from points where it might be possible to withdraw significant quantities.

6.4. Away-From-Reactor Storage Facilities

The IAEA has always given considerable attention to the development of safeguards procedures for irradiated fuel storage, but this problem may become more important in the future. Member States are accumulating irradiated fuel
assemblies at a rapid rate, and some of these States will have to store them separately in so-called "Away from Reactor" (AFR) storage facilities.

The design of an irradiated fuel storage facility plays a vital role with respect to international safeguards for the spent fuel stored in it. The construction elements of such storage, e.g. building walls, penetrations, fuel containers, handling equipment, may serve as safeguards containment measures and determine the possibility of using surveillance measures.

From the safeguards point of view, the AFR storage facility ideally should have several redundant C/S barriers, e.g. building structure, containers of the irradiated fuel, monitors of transfer mechanisms, optical surveillance.

In a large AFR storage facility the receipt of the irradiated fuel and its handling can be a very intensive activity, at least for the first several years. Since all fuel received should be verified before being placed into storage, there is a need for automation of such verification. The collection of information from the interrogation instrumentation in question might be done by an IAEA-owned computer.

The findings of the C/S devices might also be collected and compared with reference values by the same computer.

6.5. Enrichment Facilities

The Agency has been developing safeguards procedures for enrichment plants for a long time. In the last two years this activity took the form of participation in a Hexapartite Safeguards Project sponsored jointly by several Member States directly affected by the application of safeguards to uranium enrichment plants of the gas centrifuge type.

A report on this subject is being presented later in this Symposium. The goal of the project is to gain enough technical experience and acquire enough information for developing and implementing an effective and efficient safeguards system for specific facilities. The project has successfully gone through the first stage - acquisition of relevant information, development of methodology for evaluating and comparing alternative approaches and for considering fundamental problems, such as safeguards strategies. It has also approached the second stage - development of practical and reliable safeguards procedures
for specific facilities that are now under Agency safeguards or are planned to be in the near future. As usual this practical part of the work creates many more problems than general studies. For this reason the original project schedule is slightly delayed. However, a very important part of the programme, the demonstration of safeguards methods and techniques under actual plant conditions, has been started and has yielded some promising results.

7. CONCLUSION

The significant experience gained by the IAEA in the area of practical safeguards implementation at a growing number of increasingly complex nuclear facilities demonstrates certain key points, namely that the system of international safeguards is (1) effective enough to assure the reliable control of nuclear materials in accordance with safeguards agreements, and (2) flexible enough to accommodate the increasing technological developments in the field of safeguards, including the expansion of the available set of inspection equipment, and the greater complexity of modern nuclear facilities. Several basic areas in which the Secretariat is conducting studies for the further improvement of safeguards systems were mentioned in this paper. Studies discussed in other papers presented at the Symposium, e.g. the development of safeguards effectiveness assessment methodology [9], complete the picture. One such study of particular importance is that related to fuel cycles. The design of safeguards approaches up to now has been oriented toward individual nuclear facilities. The acquisition of experience in safeguards implementation, the increase in the number of facilities under safeguards, and financial and technical limitations which hardly will be eliminated in the near future, require that the Section give increased attention to studies aiming at the optimal allocation of existing resources. One means is through the development of optimized safeguards approaches which take into account factors which are characteristic of the State and its fuel cycle structure. At present such studies are the object of very intensive investigations by SAGSI. Similar work is conducted by the Secretariat. The problem is extremely complex, but we hope that System Studies Section will be able to report on this subject at the next Safeguards Symposium.
Record of the 511th Meeting (of the IAEA Board of Governors) held at Headquarters, Vienna, on Tuesday 21 February 1978 at 3:15 p.m. IAEA internal document GOV/OR.511 (May 1978). (Strictly speaking, the Board of Governors took note of a Special Implementation Report which contained and was based on the criteria, not of the criteria themselves).


M. Honami, D. Jung, General Considerations In Safeguarding CANDU Reactors By Item Accounting And Containment/Surveillance, IAEA-STR-83 (June 1979).


APPLICATIONS OF IAEA SAFEGUARDS EFFECTIVENESS ASSESSMENT METHODOLOGY

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Abstract

APPLICATIONS OF IAEA SAFEGUARDS EFFECTIVENESS ASSESSMENT METHODOLOGY.

For a number of years, the International Atomic Energy Agency (IAEA) has been developing a methodology for evaluating the effectiveness of international safeguards systems. In late 1981, an IAEA Advisory Group endorsed the basic principles of this safeguards effectiveness assessment methodology (SEAM), as well as its trial application to LWRs using actual inspection data and its continued development to bulk handling facilities. During this development process a number of case study applications of the methodology were also undertaken as part of the U.S. support programme to IAEA. Specifically, these case studies dealt with (a) a mixed oxide fabrication facility, (b) a reprocessing facility, and (c) a number of special cases of LWR safeguards. This paper describes experience with the application of the methodology in these studies, preliminary experience in the trial application to LWRs, and plans for possible future utilization of the methodology. A number of general lessons have been drawn from the process of applying the assessment methodology to the various types of facilities that were considered: (1) Safeguards problems differ within a facility type, so that an assessment must be made using a fairly specific set of assumptions. (2) The question of what follow-up activities would be necessary to resolve the initial anomalies discovered by the safeguards system is an important one, whose answer may determine the acceptability of the system to the Agency or the operator. (3) The number of diversion paths analysed grows rapidly with the number of combinations of concealment methods considered, and a simplified means of presentation and manipulation is important. (4) A major problem remaining is the estimation of detection probabilities in complex situations involving "human factors", such as the process of surveillance film review. (5) Although an IAEA Consultants' Group set up before 1981 defined a set of three "technical complexity" levels categorizing the difficulty of diversion paths, a more elaborate system may be useful in other contexts. (6) If SEAM is to be used routinely by the IAEA it would be necessary to facilitate the transfer of information from inspection reports to the SEAM process.
I. INTRODUCTION

This paper describes the development of and initial experience with a methodology for assessing international safeguards effectiveness. This methodology has been developed by the International Atomic Energy Agency (IAEA) Secretariat with the help of member states over the last few years.

The Safeguards Effectiveness Assessment Methodology (SEAM) originated in a series of meetings of an IAEA Consultants' Group which occurred between 1979 and 1981. With the help of Member States and the European Community, two documents outlining and providing an example application of the methodology were drafted: a methodology overview and a pressurized water reactor (PWR) case study [1]. The documents, as well as a report by the Safeguards Evaluation Section (SES) on the applicability of the methodology to the evaluation of inspection data, were considered in late 1981 by an IAEA Advisory Group. This group found that the basic principles of the methodology were sound, and recommended further testing of the methodology on inspection report data. This trial application is now underway, and preliminary results will be discussed in subsequent sections, as will experience with further case study applications conducted under the U.S. support programme.

The methodology takes as its basis the safeguards objective as stated in INF CIRC/153 (Corrected): "the timely detection of diversion of significant quantities of nuclear material... and deterrence of such diversion by risk of early detection." For its role in the application of safeguards, the Agency concentrates on that element of "the risk of early detection" related to its intended function, the detection of diversion. Thus the methodology takes as the principle parameter characterizing effectiveness the probability of detection of diversion, should diversion occur, given that the prescribed safeguards activities are performed.

Safeguards activities result in the detection of diversion by making possible the observation of anomalies that would result should diversion occur. Thus the methodology requires an assessment of the probability of anomaly detection in the event of diversion. To reduce the probability of incorrectly concluding that a diversion may have occurred, the methodology also requires the identification of follow-up actions to determine if such anomalies were due to innocent causes.

In brief, the methodology may be described as having the following components:
- A description of the facility, including material types and amounts and other safeguards-relevant data.
- A specification of the technical safeguards objectives, including detection goal quantity and timeliness goals.
- A definition of the safeguards approach. This includes a description of the different types of inspections planned and the timing of each type of inspection. It includes the set of safeguards activities to be performed, both in the field and at headquarters. It also includes a description of the set of anomalies that might be observed as a result of these activities, and the follow-up activities that would be needed to resolve those anomalies.
- A diversion path analysis, which systematically identifies the possible methods by which nuclear material might be diverted and the logical combinations of concealment methods that might be used to conceal the diversion.
- An assessment, for each diversion path, of the probability that an anomaly will be detected by the defined safeguards system, should diversion occur in the manner described by the path.
- A categorization of the set of diversion paths (and hence detection probabilities) into three levels of "technical complexity" depending on the degree of difficulty involved in carrying out the diversion and/or concealment strategies. It is clear that some diversion paths are more readily implemented than others and are more important to protect against. It is desirable to be able to allocate Agency resources so as to cover the more important paths more effectively.
- A summary of results. Basic results are bar charts giving the set of detection probabilities, categorized by technical complexity level (TCL). It is important to reduce the results to a more transparent and manageable form. One method of doing this is to present a histogram of the detection probabilities of the paths for each complexity level. Further data reduction, by means of a formula (or "aggregate measure") combining detection probabilities into numerical indices, is under consideration, although no particular formulations have been agreed upon.

Three modes of application are envisioned for the assessment methodology: (1) the design assessment, in which a safeguards approach is evaluated a priori, based on realistic
but general assumptions with respect to a representative facility, in order to evaluate safeguards design concepts; (2) the implementation assessment, in which a safeguards approach to a specific facility is evaluated; and (3) the performance evaluation, in which an assessment is made on the basis of inspection activities as carried out in the field over a specified period.

A more detailed description of the methodology can be found in Refs [2] and [3].

II. MODEL ASSESSMENTS

A number of "case studies" are being undertaken under the U.S. support programme in which the methodology is applied to safeguards systems for various facility types: light water reactors (LWRs), mixed-oxide (MOX) fuel fabrication facilities, and reprocessing plants. These studies have two purposes: to demonstrate the manner in which the methodology can be applied to various safeguards techniques, and to use the methodology to provide a detailed analysis of the safeguards schemes under consideration. It is hoped that these analyses, which are basically in the nature of "design" assessments, will help provide a framework within which the safeguards problems of the various facility types can be discussed, and help form the basis for more specific ("implementation") assessments for particular facilities. In the case of on-going work on MOX and reprocessing plants, this is also the first application of SEAM to bulk handling facilities, a step encouraged by the Advisory Group.

An initial case study for a PWR was considered by the Consultants' Group. The safeguards design analysed included (1) verification of the consistency and completeness of records and reports, (2) item counting, item identification, surveillance, and NDA techniques applied to the spent fuel pool, (3) sealing of the reactor vessel during power operation, (4) sealing of spent fuel shipping casks declared to be partially full, (5) item counting, identification, and NDA applied to fresh fuel. Concealment techniques considered in developing the list of diversion paths included records and reports falsification, camera and seal tampering, and material substitution. Three types of models were used in estimating detection probabilities: (1) a simple, direct estimate of the likelihood of detection, given that the appropriate inspection activity was performed (this was used in the case of the detection of records and reports falsification), (2) a sampling model, which considered the probability of a substituted item being sampled, and the probability that, once
sampled, an anomaly would be detected (by an NDA technique, for example), (3) a model estimating detection probability for surveillance anomalies, based on the time duration of the observed activity, the camera frequency, and human factors. Methods of estimating some of these latter factors in practical circumstances remain to be formulated.

Case studies dealing with safeguards for other LWR cases (e.g. where fuel elements are disassembled), MOX fuel fabrication facilities, and reprocessing plants are currently under way. By contrast with reactor safeguards, which rely on containment and surveillance in addition to material verification measures, the MOX study is based on a safeguards approach which implements the IAEA attributes/variables material accountancy verification approach, so that most anomalies (aside from those dealing with records and reports) involve measurement discrepancies or discrepancies in material accountancy statistics. For such anomalies, detection probability formulas using fairly standard mathematics are available, based on sample sizes and measurement uncertainties. In the case of material accountancy verification, however, the diversion path analysis must attempt to "cover" the continuum of falsification possibilities (falsification by large defects, small defects, diversion into MUF, etc.) by a discrete set of scenarios or paths for which detection probabilities will be calculated. The analysis must also include concealment methods designed to mislead the inspector's measurements. Despite the difference between the MOX bulk handling facility and the reactor item handling facilities on which SEAM was first tested, the general framework of SEAM as outlined above appears to be applicable.

In addition to the facility-specific results of the studies, there were a number of general lessons that can be drawn from the process of applying the assessment methodology to the various types of facilities that are being considered:

- Safeguards problems differ within a facility type, so that an assessment must be made using a fairly specific set of assumptions. Put differently, the assessment of a "generic" safeguards approach applied to a "generic" facility may have to be considerably modified when analysing effectiveness of safeguards in an actual situation. This is especially true of containment/surveillance (C/S) systems, but this also relates to facility-specific record keeping systems. A considerable amount of facility-specific detail is needed for a useful facility-specific result.
The question of what follow-up activities would be necessary to resolve the initial anomalies discovered by the safeguards system is an important one whose answer may determine the acceptability of the system to the Agency or the operator. This again is particularly true of C/S devices.

Since the number of diversion paths analysed grows rapidly with the number of combinations of concealment methods considered, and can be quite large, a simplified means of presentation and manipulation will be important.

A major problem remaining is the estimation of detection probabilities in complex situations involving "human factors", such as the process of surveillance film review. Because so many parameters affect the effectiveness of a surveillance system, it is suggested that certain standardized procedures be developed whose effectiveness can be documented.

Although the Consultants' Group defined a set of "technical complexity" levels categorizing the difficulty of diversion paths (in relation to the PWR case study), a more elaborate system or an extended set of definitions may be useful in other contexts.

III. TRIAL APPLICATION OF SEAM

Trial applications of the methodology to actual LWR inspection reports should refine the understanding of the methodology's basic rationale on the basis of practical experience. Such trials should also provide the Agency with a wider and firmer basis on which to consider further the uses to which the developed methodology might be applied.

Although many potential uses of the methodology were identified in the report to the Advisory Group, certain obvious applications fall within the Division of Development and the SES. For the Division of Development, the methodology can play a role in developing and assessing alternative safeguards approaches and in directing research and development activities into promising areas (i.e. promising to offer the largest improvements in safeguards effectiveness).

SES has, among others, a responsibility for carrying out a review of the effectiveness of the safeguards as implemented in Member States. This involves the processing of large amounts of information and requires summarizing it in a way useful to management understanding of the effectiveness of safeguards. It also involves the identification of
deficiencies, the means for correcting those deficiencies, and the communication of the technical basis for policy, funding, and R & D requirements to non-technical segments of management.

For SES to fulfil its mission in a more effective manner, a tool is needed with the following characteristics:

1. A systematic basis and methodology which can be applied uniformly to the evaluation.
2. A standardized analytical model which can be used to analyse and synthesize the data reported in inspection reports, State reports and other pertinent Agency documents.
3. A process that can be carried out efficiently making optimum use of computers in order to minimize manual work and possible introduction of errors.

The methodology described in the preceding sections has all the desirable features.

Because of the perceived utility of the SEAM to both units, the trial application recommended by the Advisory Group Meeting is being carried out jointly between the System Studies Section (in the Division of Development) and SES. Close cooperation with the regional sections is, of course, a necessary part of the trial.

The facilities selected for the trial consist of 16 reactors of the PWR type drawn from all Regional Sections in the Department of Safeguards. The following describes results at the mid-point of this trial application.

III.1 Programme Plan for the Trial Implementation of SEAM in SES

The specific objectives of the Trial Application were as follows:

1. Assess the practicability of applying the methodology to evaluate PWR inspections using actual field data.
2. Assess its applicability to PWRs of all types as safeguarded by all regional sections of the Department of Safeguards.
3. Assess the uses of the outputs of the analysis, such as bar charts, representing degrees of coverage of individual diversion paths and aggregate measures being considered for the assessment of annual performance.
III.2 Implementation of the Trial Application

The initial phase of the task consisting of preparatory work and the analysis of a number of reactor types has now been completed. Some significant results are already apparent. These will be discussed in section III.3. The main elements of this preparatory phase are as follows:

1. Reactor facility designs differ within the group of PWRs being studied in ways which are relevant to the methodology. There are four major functional variants of the PWR which affect the choice of diversion paths. These variants are as follows:
   a) A two-building class where the reactor is located in a containment not accessible during reactor power production operations and the spent fuel pond is located in a separate building which is accessible.
   b) A single building class where the reactor and spent fuel pond are located in the same building. Reactor vessel shielding as well as spent fuel pond are accessible during power operations.
   c) A single building class containing the reactor core and spent fuel pool and neither the core nor the spent fuel pond are accessible during power production operation.
   d) A single building class containing two reactor cores and two spent fuel ponds, both cores and spent fuel ponds being accessible for inspection during power production operation.

Case (b) has been analysed in some detail in earlier work, and model inspection activity lists and diversion path listings are available. Model inspection activity lists and diversion path listings for the other cases are in the process of being constructed.

2. The evaluation requires as an input a determination of the activities that were performed, not only during the course of the inspection, but also at Agency headquarters. These must be input in a format similar to the one illustrated in Fig. 1. This is completed for about half of the selected reactors for the year 1981.

3. The analysis can be performed using the EVAL computer program. This has been completed with the assistance of the U.S. support program and is now available for use by Agency personnel. An analysis of inspections at several reactors has been completed with satisfactory results.
| ABC.3 | Compare S/R ICR's |
| ABC.4 | Compare facility records to reports |
| ABC.5 | Compare total book inventory |
| ABC.6 | Internal consistency of accounting vs. operating records |
| ABC.8 | Examine power operating record |
| ABC.9 | Compare CMR to IC's for SFP |

| A.1 | Count FFS and compare to records |
| A.2 | FFS item identification |
| A.3 | FFS attributes NDA |
| A.4 | FFS variables NDA |

| B.1 | Count RC and compare to records |
| B.2 | RC item identification |
| B.3 | RC attributes NDA |
| B.4 | RC variables NDA |
| B.5 | Examine RC guide tubes |

| C.1 | Count SFP and compare to records |
| C.2 | SFP item identification |
| C.3 | SFP attributes NDA |
| C.4 | SFP variables NDA |

| BC.2 | Review film for RC/SFP, compare to records |
| BC.3 | Verify integrity of seals on cameras |
| BC.4 | Verify integrity of seals on reactor missile shield |
| BC.5 | Attach seals on partly filled cask after verifying its content |
| BC.6 | Verify seals at Headquarters |

**FIG.1. Model Inspection Activity List (Summary Format) for Reference LWR.** Input data to EVAL for a particular inspection consists of this list together with indication that activity has been performed or not. Additional data input shows total population of items and sample size when Activities A.2, A.3, A.4, B.2, B.3, B.4, B.5, C.2, C.3 and C.4 are applied.
DIVERSION PATH DETECTION PROBABILITIES

MATERIAL TYPE: SPENT FUEL

TECHNICAL COMPLEXITY LEVEL: A

<table>
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<tr>
<th>Path</th>
<th>Probability Level</th>
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<th>L</th>
<th>I</th>
<th>H</th>
<th>VH</th>
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FIG. 2. Example of ex post facto assessment for a particular inspection of the probability of detecting a diversion path of technical complexity level A. The results of an ex post facto assessment of the effectiveness of a single inspection are displayed as charts showing for each diversion path the probability of having detected one or more anomalies if a diversion had occurred using that path. The diversion paths are grouped by technical complexity level and material type. The initial runs of EVAL used detection probabilities of "0" or "1", an assumption that will be improved as the trial application proceeds.
Because of their availability, the diversion paths and model inspection activity lists that were used to date represent case (b). Additional modifications will be made to represent cases (a), (c) and (d).

III.3 Preliminary Results

1. For each reactor type which is functionally different in a way which affects safeguards activities, an appropriate set of diversion paths must be constructed. The set must be relevant not only to the class of PWR but also to the particular circumstances of the actual facility. For each set, the number of paths is considerable (some hundreds) and as a consequence a simplified means of listing and manipulating the sets needs to be found. Some progress has been made in this area, and a computerized means of path generation and subsequent manipulation (including the selection of appropriate paths in the light of case data) is under development.

2. Inspection activity performance data are processed by the appropriate algorithms to estimate the anomaly detection probability if there were a diversion along each path. The output is printed in the form of bar charts and summary tables. A typical bar chart is shown in Fig. 2. The example shows the result of a physical inventory verification (PIV) of the nuclear material in the core and spent fuel pond. The bar chart provides a very useful representation of the degree to which each diversion path is covered by the inspection activities. It provides a valuable tool for a rapid assessment of safeguards effectiveness. In addition, the results are summarized for each material, location and TCL in terms of average probability of detection and listing of diversion paths for which probability of detection is low. This is useful in making comparisons between different inspections and inspections at different facilities.

3. The progress of the trial application to date indicates that it holds a potential as a useful tool for the review of effectiveness. However, a number of steps have been identified that must be taken before SEAM can be used without causing an undue time-consuming burden. To complete the steps for routine use will require cooperation and intimate understanding of SEAM among operational, development and review activity staffs. The principal steps now foreseen are:
1. Develop mechanistic means for selecting diversion paths appropriate to specific facility conditions.

2. Arrange for efficient information transfer from inspection reports and other Agency data files to the SEAM computer program EVAL.

3. Develop improved approaches for the estimation of probabilities of detection of anomalies based on actual inspection data.

The first step would facilitate the routine use of SEAM by conveniently selecting diversion paths appropriate to specific facility conditions (e.g. excluding diversion paths from a storage area because fuel was not present in that area).

With regard to the third step, the computer program EVAL calculates anomaly detection probabilities for safeguards activities which use sampling methods (e.g. the probability of detecting an assembly with an erroneous identification number as a result of checking a limited number of randomly selected assemblies). In all other cases, the probabilities are based on the assumption that activities are executed properly in conformance with Agency practice. There is no reason to believe that high standards of performance set by the Agency's directives are not met in the execution of inspections.

IV. FUTURE PLANS

The results of the trial application of SEAM to LWRs obtained to date appear promising to Agency staff. The trial application will be extended so that full results are available covering each of the reactor types which is included in the sample. It is anticipated that, following the completion of this analysis, the results of the trial application and their evaluation will be presented to a Consultants Group. The Agency would look to the group's guidance as to any further steps which are needed to make use of the methodology. It is anticipated that such a Consultants' meeting would be convened during the first half of 1983.

With respect to the development of the methodology for bulk handling facilities, the Agency is encouraged by the early results of design assessments. It appears that the methodology can be applied in a consistent and meaningful manner to this type of facility despite the fact that the type of diversion path is, in general, quite different from that seen in the case of LWRs. This work is, of course, far more
preliminary than that of the LWR work, and any definitive judgments would need to await later results and further consultation with experts from Member States.

In both cases the Agency will be guided by the lessons which have been learned to date and will need to continue, and in some cases initiate, corollary activities which can proceed in parallel as described above.

References


BELGIAN EXPERIENCE IN SAFEGUARDS IMPLEMENTATION AND RELATED R&D ACTIVITIES

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Franco-Belge de Fabrication de Combustible, Dessel

H. DE CANCK, R. INGELS  
Belgonucléaire, Dessel

J. SATINET  
Centrale Nucléaire de Tihange (Tihange Power Plant), Belgium

Abstract

BELGIAN EXPERIENCE IN SAFEGUARDS IMPLEMENTATION AND RELATED R&D ACTIVITIES.

As a non-nuclear-weapon State party to the Non-Proliferation Treaty, Belgium considers the Agency's safeguards system as one of the basic components of the Treaty. Therefore, in this particular case, the effectiveness of the Agency's safeguards system as a result of the cooperation between the Agency, Euratom and the State has to be clearly stated and demonstrated. This paper will follow this idea as a guideline, considering the implementation of international safeguards in Belgium's main representative nuclear facilities. It should be borne in mind that nuclear energy contributes a significant part of the production of electrical energy (about 25% in the year 1980) so that the impact of safeguards on the Belgian nuclear fuel cycle constitutes important information for the Agency's safeguards system. This impact has led Belgium to propose a Research Support Programme to the Agency, the three parties involved being IAEA, Euratom and Belgium. The first two studies in this programme concern: (i) the optimization of safeguards measures in mixed oxide fabrication plants, and (ii) the impact of improvement in verification capabilities on the safeguards approach in LEU fabrication plants.

1. INTRODUCTION

The first principle of nuclear material control is the need for a strict correspondence between records and actual transactions; the second principle is to obtain a material balance in which all known material movements or inventories are measured ('closed material balance'). Where these
principles are applied the 'material unaccounted for' (MUF) is
the primary means of showing any unrecorded loss or gain of
material in a specific MBA and during a specific time period.

The technical framework of the IAEA safeguards system has
been defined in the safeguards document INFCIRC-153 (1970).
The Agreement INFCIRC-193 between non-nuclear-weapon States of
the European Community and the Agency was agreed in 1973. For
the implementation of this Agreement the Commission Regulation
No. 3227/76 is the reference. The Belgian operators have to
implement a material accounting system in accordance with this
regulation. It should be noted that all the Facility
Attachments concerning Belgian facilities were drawn up in
1978/1979, including the mixed oxide fabrication plant
(sensitive facility).

2. POWER REACTORS

2.1 Introduction

If 1979 is set as the reference year [1], Belgium's
energy supplies and consumption break down as follows:

(i) Nature of the fuel
      Oil  51.4%
      Coal 22.5% (8.8% Domestic)
      Natural gas 21%
      Nuclear power 5.1%

(ii) Consumption
      Electricity production 23.8%
      Industrial uses 26%
      Transport 11.5%
      Household uses 25.9%
      Losses and non energetic uses 12.8%

Belgium relies on foreign countries for approximately 90% of
its energy imports.

In order to reduce such dependence and also to diversify
the origin of their supplies, most industrialized countries
build and operate nuclear power stations.

It seems reasonable that each country should rely on
nuclear energy in proportion to its degree of dependence \(D\)
regarding fuel and gas imports. This parameter \(D\) is defined as
the total fuel and gas imports divided by the total energy
consumption.
Fig. 1A shows the estimated degree of dependence of several industrialized countries in 1985; the share of nuclear energy in the total consumption is indicated in Fig. 1B. In this respect, Belgium's position is comparable to that of France, Japan, the Federal Republic of Germany, and the United States of America.

The evolution in time of the equipment policy followed by the Belgian electricity producers is given in Fig. 2. This policy should reach its objectives in ten years' time, thanks to the combined use of coal and nuclear power. In 1986, the quota of nuclear power stations will thus reach 60%. The remainder of the requirements will be supplied by classical means:

- Oil and gas: 16.5%
- Coal: 22.5%
- Hydraulic power: 1%

The Belgian power reactors are located in DOEL and TIHANGE and have the following installed capacities (in MW(e)):

All these reactors are PWRs. Typically a 925 MW(e) PWR reactor core contains 72.50 tonnes of LEU in 157 fuel assemblies.

2.2 Description of the safeguards approach

The safeguards approach to LWRs and its effectiveness versus practical diversion paths is described in pages 15-20 of Ref. [2]. This approach is applied in the Belgian power reactor stations.

2.3 Experience in safeguards implementation

The safeguarding authorities spend about 10-15 man-days per year at the facility but this figure only takes into account the presence of inspectors. The operator's total effort for nuclear material accountancy has been estimated for the unit TIHANGE-1 (900 MW(e)) and is distributed as follows:

- Attendance at inspections: 10-15 man-days **
- Bookkeeping (files, logbook): 30-40 man-days *
- Drawing up physical inventory listing: 5-10 man-days *
- Nuclear transformation calculations: 10 man-days
- Drawing up book inventory: 5-10 man-days
- Drawing up special report (fuel assembly modification and repairs): 30-70 man-days **

Total: 90-155 man-days
The items denoted with (**) or (*) are totally or mainly imposed by the safeguards activities.

2.4 R & D activities

Safeguarding of PWR facilities is a comparatively easy task, the nuclear material accountancy being based on item counting and identification. However, considering these facilities as part of the fuel cycle, one has to take into account not only $U_{eq}$ and $^{235}U_{eq}$, initial contents in the fresh fuel assemblies, but also the estimates of $U_f$, $^{235}U_f$, Pu contents at the time of discharge. Realistic estimates of these final values are required for fuel management purpose; they are also very important in the application of isotopic correlation techniques as a safeguards tool [3,4].

In most reactor fuels, the changes in the amounts of the various fuel isotopes are related to each other in a rather simple manner if one pays attention only to the total quantities present in the fuel assembly or in a group of fuel assemblies. These relationships are called 'isotopic correlations' [5]. Procedures for their safeguards application have been summarized in Ref. [6].

For the Belgian fuel cycle, the demand for reprocessing will be of the order of 120 tonnes U per year in steady state conditions, corresponding to about 1200 kg Pu. The following up of each reprocessing campaign by the reactor operator and the comparison between the estimated values (deduced from his reactor code updated on the basis of past campaigns) and the measured values at the input of the reprocessing plant will significantly enhance and maintain the confidence one can place in this technique.

In this way, it will be possible for the safeguarding authorities to draw up semi-independent U, $^{235}U$ and Pu balances of the head-end with a minimum of additional analytical effort [7].

3. LEU FABRICATION PLANT: FBFC, DESSEL

3.1 Introduction

The low enriched uranium (LEU) fabrication plant produces PWR fuel assemblies of 1.5% to 4% enrichment at an annual throughput of 400 t of U. The input material is $UO_2$ powder. The PWR fuel assemblies contain uranium of a single level of

<table>
<thead>
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<th>INPUT</th>
<th>OUTPUT</th>
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<tr>
<td></td>
<td></td>
<td>kg U</td>
<td>%</td>
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<tr>
<td>1</td>
<td>Fresh powders (shippers' data)</td>
<td>441 700</td>
<td>99.9</td>
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<tr>
<td>2</td>
<td>Fuel assemblies</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>Others</td>
<td>237</td>
<td>0.1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>441 937</td>
<td>100</td>
</tr>
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</table>

enrichment. The production follows a strict 'path per enrichment rule', and this characteristic is fully taken into account in the present safeguards approach.

The statement of all inputs and outputs is given in Table I for the inventory period 1981–1982; the fresh powders in standard drums contribute 99.9% of the inputs and the finished fuel assemblies 99.7% of the outputs.

The 1982 inventory is given in Table II. Finished fuel assemblies and standard drums of fresh powder contribute 64% and 11% of the total, respectively.

From Tables I and II, it appears that fresh powders and finished fuel assemblies are the main components of the yearly material balance (MUF equation).

The MUF equations corresponding to recent yearly periods have been carefully studied by the operator, identifying their various components and assessing their error structures.

The main conclusions are as follows:

- The estimated measurement uncertainty for closing the material balance over the last inventory periods ($\sigma_{MUF}$) is smaller than 0.2% of throughput, even though pessimistic statistical approaches are applied.
- More than half the variance originates from inputs (fresh powders) and outputs (fuel assemblies)
TABLE II. NUCLEAR MATERIAL INVENTORY AT THE LEU FABRICATION PLANT: 1982 INVENTORY

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<th>KMP</th>
<th>DESCRIPTION</th>
<th>kg (U)</th>
<th>% total U</th>
<th>batches</th>
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<td>A</td>
<td>Fresh Powder Store</td>
<td>39 370</td>
<td>11</td>
<td>23</td>
<td>1 668</td>
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<tr>
<td>B</td>
<td>Purified residue Store</td>
<td>697</td>
<td>0.2</td>
<td>49</td>
<td>49</td>
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<tr>
<td>C</td>
<td>Pelletizing Area</td>
<td>17 309</td>
<td>5</td>
<td>311</td>
<td>1 361</td>
</tr>
<tr>
<td>D (*)</td>
<td>Pellet Store</td>
<td>25 625</td>
<td>7</td>
<td>48</td>
<td>202</td>
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<tr>
<td>E (**)</td>
<td>Canning Area</td>
<td>3 667</td>
<td>1</td>
<td>460</td>
<td>4 885</td>
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<tr>
<td>F (*)</td>
<td>Rod Store</td>
<td>22 829</td>
<td>7</td>
<td>83</td>
<td>12 975</td>
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<td>Assembling Area</td>
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<td>64</td>
<td>195</td>
<td>195</td>
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<td>H (')</td>
<td>Assembly Store</td>
<td>122 381</td>
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<td>265</td>
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<td>I</td>
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(*) Strategic points (quantity + quality) for monthly verification
(**) Strategic Points for monthly verification

- $\text{MUF} \ll \text{MUF}_\text{MBA}$, validating the hypothesis of zero diversion.

3.2 Material accountancy

The plant is considered as a single material balance area (MBA). The overall safeguards approach is described in Ref. [8] and the relevant key measurement points (KMPs) and strategic points (SPs) are listed in Tables I and II.

3.2.1 Physical inventory

The yearly physical inventory is based on item identification and on the operator's relevant measurements.

The verification of the physical inventory is based on the physical inventory listing drawn up by the operator. Nuclear materials are stratified and labelled item by item. The man-days required for this labelling and its verification (100%) are given in Table III.

The Book Inventory has been verified by the inspectors and the presentation of the results has taken 34 man-days for
TABLE III. MAN-DAYS REQUIRED FOR LABELLING OF THE PHYSICAL INVENTORY LISTING (1980) AND ITS VERIFICATION BY THE SAFEGUARDING AUTHORITIES

<table>
<thead>
<tr>
<th>DESCRIPTION</th>
<th>t(U)</th>
<th>Number of batches</th>
<th>Number of items</th>
<th>Operator man-days (1)</th>
<th>Safeguards man-days (2)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pins</td>
<td>24.8</td>
<td>129</td>
<td>14 249</td>
<td>13</td>
<td>1</td>
</tr>
<tr>
<td>Bottles</td>
<td>0.9</td>
<td>117</td>
<td>279</td>
<td>1.5</td>
<td>0.5</td>
</tr>
<tr>
<td>Bottles</td>
<td>0.1</td>
<td>76</td>
<td>241</td>
<td>2</td>
<td>0.5</td>
</tr>
<tr>
<td>Bottles-Drums</td>
<td>5.5</td>
<td>164</td>
<td>407</td>
<td>4.5</td>
<td>0.5</td>
</tr>
<tr>
<td>Bottles</td>
<td>0.3</td>
<td>378</td>
<td>2 421</td>
<td>8</td>
<td>1.5</td>
</tr>
<tr>
<td>Assemblies</td>
<td>311.3</td>
<td>676</td>
<td>700</td>
<td>3</td>
<td>1.5</td>
</tr>
<tr>
<td>Bottles-Drums</td>
<td>43.2</td>
<td>1 788</td>
<td>2 115</td>
<td>25</td>
<td>9</td>
</tr>
<tr>
<td>Drums</td>
<td>43.0</td>
<td>191</td>
<td>1 880</td>
<td>1.5</td>
<td>1</td>
</tr>
<tr>
<td>Trays</td>
<td>37.1</td>
<td>47</td>
<td>5 467</td>
<td>5</td>
<td>1</td>
</tr>
<tr>
<td>F.B.F.C.</td>
<td>465.9</td>
<td>3 652</td>
<td>27 764</td>
<td>63.5</td>
<td>16.5</td>
</tr>
</tbody>
</table>

(1) Presentation of results to the authorities (34 man-days) not included

(2) NDA, sampling for DA and accounting verification (19.5) not included. The verification of the P.I. has been carried out by four Euratom and four IAEA inspectors (36 man-days).

The operator. Furthermore, nuclear materials have been sampled and measured by non-destructive assay (NDA) techniques (80 measurements) and destructive analyses (20 measurements).

It has to be said that at present the physical inventory is organized not only for the safeguarding authorities, but also for inner material management, and on the request of certain customers. So the time for preparing the inventory (63.5 man-days) is not completely lost for the operator. Nevertheless the customers are expected to attach less and less importance to physical inventories in the future. For this reason, the operator's interest in a yearly physical inventory may decrease until this effort becomes in the first place a requirement of the authorities and thus a real extra task for the facility.

In any case the presentation of the results of the inventory to the safeguarding authorities (34 man-days) is of no profit for the operator and must be considered as an extra cost for the plant.

3.2.2 Flow verification

When a new enrichment enters a store or a production stage, these are cleaned out to avoid any possible mixing of
different enrichments. The safeguards approach provides that, at intervals of about one month, the inspectors verify:

- the inventories 'per enrichment' of the main four stores
- the book values 'per enrichment' (audit)

and count all the assemblies produced since the last inspection, the data being updated by relevant accounts.

The analysis of the data provides the evidence for the flow through the production steps of a sort of 'wave' of uranium of a particular enrichment i, which finally has to be present in the assemblies of this enrichment.

The verification of the flow of the nuclear materials is effected monthly by two Euratom and two IAEA inspectors (88 man-days/year); the operator's effort amounts to 77 man-days/year.

3.2.3 Inspection effort estimates

The total inspection effort is summarized in Table IV; the man-days for the safeguarding authorities are estimated taking into account only the presence of inspectors at the facility.

3.3 Observations on safeguards effectiveness

The following observations are relevant to safeguards effectiveness:
(i) The estimated operator measurement uncertainty (standard deviation) associated with closing a material balance expressed as a percentage of the larger inventory or throughput is 0.3% in the case of LEU fabrication plants. This value is currently 0.2% at the FBFC plant. The minimum quantity of nuclear material which could be detected by the operator with a probability \((1- \beta_a)\) by application of accounting measures alone is connected with \(a_a\), the false alarm probability.

When \(\text{MUF} \ll \sigma_{\text{MUF}}\), as was the case at FBFC for the previous material balances, this minimum quantity amounts to

\[
80 \text{ kg } ^{235}\text{U} \text{ with } a_a = 5\%, \beta_a = 5\%, \text{ for a } 400 \text{ t } \text{U} \text{ throughput,}
\]

and this quantity is very close to the significant quantity of 75 kg \(^{235}\text{U}\).

(ii) A very significant gain in the sensitivity of loss detection can be obtained by changing from the traditional emphasis on \(^{235}\text{U}\) MUF to U-element MUF [9]. A suggested way of changing the loss detection emphasis to U element and still retaining the same control would be to have two \(\sigma_{\text{MUF}}\), one for U and one for enrichment on that part which arises solely from the enrichment measurement errors. The gain in loss detection sensitivity by this change amounts to 30%.

(iii) For the verification, the safeguarding authorities have the declared data from two main shippers (or MBAs) outside Belgium with the accuracies of destructive analyses. Furthermore, the rod scanner associated with reference standard rods constitutes a very powerful tool for output measurement verification, based on destructive analyses of randomly chosen pellets. The verification capabilities are thus very close to the capabilities of the operator.

(iv) A greater emphasis should be given to the role of LEU fabrication plants in furnishing safeguards cross-checks in the safeguarded fuel cycle rather than the over-emphasis currently placed on detecting the diversion of LEU [9].
3.4 R & D activities

The following observations are relevant to R & D activities:

(i) The quality of the source data $U_0$ and $235U_0$ for fresh fuel assemblies is of fundamental importance for getting the best estimates of the Pu content at the head-end of the reprocessing plant based on isotopic correlation techniques.

These source data may be complemented by the abundances of the uranium isotopes $234U$ and $236U$:

- The relationship Pu/U versus $\Delta 236U$ gives one of the most accurate estimates of the Pu/U ratio
- The $234U$ content may be used as a tracer for the $235U$ content in passive NDA techniques applied for example to the complete fuel assembly.

(ii) The present overall approach has notably to be reviewed as soon as the technology is able to offer an effective NDA system for the verification of $235U$ and U contents of the complete fuel assembly.

The active neutron collar as tested in Refs [10] and [11] seems very promising in this specific area. This portable device has the advantage of being practical enough to be operated by a skilled inspector in a reasonable time without undue interference with the activities of the facility. Furthermore, a transportable photoneutron active interrogation device (PHONID-3) designed by JRC-ISPRA, to verify the complete fuel assembly offers another possibility [12].

The capabilities of these NDA techniques for flow and inventory verification have to be fully investigated in plant conditions and their capabilities have to be taken into account in a new safeguards approach which in the FBFC case would only involve input and output flow measurement points (monthly verification) and about ten inventory verification measurement points (yearly verification). This new approach, reducing drastically the amount of source data, will have the advantage of being efficient and less intrusive. This study will form part of the Belgian support programme.
4. MIXED OXIDE FUEL FABRICATION PLANT (BELGONUCLEAIRE)

4.1 Introduction

The dry-blend mixed oxide fuel fabrication facility has a nominal throughput of 500 kg Pu per year [13].

For criticality control reasons, the whole plant is subdivided into a number of work units, a work unit being the smallest entity in which the nuclear materials can be moved without any restriction within fixed boundaries. Every movement of nuclear material between two work units is submitted to a preliminary computerized control. When the transfer of the nuclear material is authorized, the records of the work units involved are immediately updated. This 'real-time nuclear material accountancy system' can produce at any moment a complete inventory of all nuclear materials in the facility with their localization, quantity, quality, material description, etc. A detailed description of the system is given in Ref. [14].

4.2 Description of the safeguards approach

The facility is considered as a single material balance area (MBA) subdivided into three areas: the input area, the process area and the output area.

All the external safeguards reports, namely: inventory change report (ICR), physical inventory listing (PIL) and material balance report (MBR) are transmitted to Euratom on magnetic tape and hard copy as required by Euratom regulations 3227/76.

The overall safeguards approach implies two physical inventory takings (PITs) per year and the verification of all inputs and outputs for the protracted diversion supplemented by a current inspection rate of once every fortnight for abrupt diversion.

4.2.1 Protracted diversion

(i) Flow measurement verification

The verification of the quantities in the input and output storage area (called Strategic Point 1 (SP1)) can easily be carried out because:

a) there is a permanent access to the stores

b) the material is present in well-defined forms (cans, rods, assemblies) and is easy to check
c) since all the nuclear material is itemized, sampling plans are easily set up on the basis of computer print-outs
d) most of the material is kept under seal, which diminishes the inspection effort considerably. Furthermore, the facility permanently stores a six months' supply of feed materials and accumulates the production for at least the same length of time, so that the verification of source data by destructive and non-destructive analyses is not precluded by the time available. The presence of the declared items in the storage is verified by sealing techniques during the fortnightly inspection.

(ii) Physical inventory verification
The physical inventory verification follows the same rules as described in sub-section 3.2.1, the items being in this case characterized by their weight and the Pu factor and isotopic composition.

(iii) Material balance
Long-term operation of the facility reveals that some fraction of the processed material inevitably accumulates in the process line and cannot be retrieved by routine cleaning techniques. This material, which is often called the 'hidden inventory', constitutes a part of the observed MUF with the result that in the absence of a real diversion the observed MUF is no longer distributed about zero, but about some positive value. The operator's experience indicates an upper limit of about 1.5 kg of plutonium for this expected value of MUF. The current attitude in dealing with an ending hidden inventory HI is to remove it from both the MUF and the set goal quantity: GQ - HI, so that the MUF* is again distributed about DL, the diversion level.

4.2.2 Abrupt diversion
Since the end of 1978 the facility has been submitted to the 'short detection time' verification done by a joint Euratom-IAEA team. Every fortnight an inspection is carried out in the storage area (SP1) and in the process area (SP2) [15]. While the verification is rather easy for SP1, the situation is different for SP2 because:
   a) there is only limited access to this area
b) the material is located in glove boxes and is found in continuously changing forms (powders, pellets, stacks, etc.)
c) the material is not kept under seal.

To find a practical solution, the process area (SP2) has been divided into a number of 'process steps'. For each 'process step' a 'strategic point location' is chosen in such a way that at this particular point all the nuclear material present in the process step can be made accessible to the inspectors.

To prepare the necessary documents for the inspectors of the joint Euratom–IAEA team every second Friday afternoon, the operators responsible for one or more process steps carry out a quick inventory of all unsealed material covered by the strategic point locations. The entities are compared and checked with the data on the operating records. All useful information is copied on a 'strategic point verification list'. For the verification the joint team inspectors receive in advance during the weekend all the required information for preparing a sampling plant on a statistical basis. To minimize disturbance to the fabrication activities during the week-end the verification starts immediately in the process area on the Monday morning.

4.3 Inspection effort estimates

The operator's effort in preparing for and taking the physical inventory (reference year: 1981) amounts to 140 and 35 man-days respectively. The second physical inventory is not needed for prudent management practices, but is solely required by the safeguards authorities. The surplus of effort on the part of the operator amounts to 35 man-days (attendance on the first inventory verification) + 175 man-days (second inventory preparation and verification) = 210 man-days.

The effort of the joint team in verifying the physical inventory amounts to 40 man-days taking into account only the presence of inspectors at the facility. Furthermore, about 40 PuO₂ cans and 100 rods have been measured by independent NDA techniques. The joint team's effort for the fortnightly inspections amounts to 40 - 50 man-days, while the operator's effort is estimated to be 80 - 100 man-days. The inspection effort estimates are summarized in Table V.

In addition, the disturbance of fabrication activities constitutes a very important penalty, about 8% of the fuel fabrication cost for inspection, preparation and performance.
TABLE V. INSPECTION EFFORT ESTIMATES (man-days): MIXED OXIDE FABRICATION PLANT (BELGONUCLEAIRE), REFERENCE YEAR 1981

<table>
<thead>
<tr>
<th></th>
<th>Operator</th>
<th>Safeguards</th>
</tr>
</thead>
<tbody>
<tr>
<td>Physical inventory</td>
<td>210</td>
<td>80&lt;sup&gt;a&lt;/sup&gt;</td>
</tr>
<tr>
<td>Flow verification and timely detection</td>
<td>80 - 100</td>
<td>40 - 50</td>
</tr>
<tr>
<td></td>
<td>290 - 310</td>
<td>120 - 130</td>
</tr>
</tbody>
</table>

<sup>a</sup> There are two physical inventories, each of 40 man-days.

New improvements have been documented, and it seems that their application could reduce this contribution by a factor of 2 [16].

4.4 Considerations on safeguards effectiveness

A parametric study has been undertaken to investigate the sensitivity of detection of the operator over a material balance period of one year in a representative mixed oxide fuel fabrication process [17]. A detailed error analysis of the plutonium balance based on a realistic structure of errors has yielded the following main observations.

a) Protracted diversion
   (i) Even at the relatively low throughput of less than 300 kg of Pu, the systematic error components, on the whole, appear to dominate (0.175% as opposed to 0.2% for the combined error); as the throughput is increased, the random error components tend to vanish, in relative terms, due to the increasing number of measurements, and therefore the combined relative error approaches the constant systematic component of 0.175%.
   (ii) A good balance has been achieved by the operator between the error components associated with the main input and output terms.
   (iii) The detection capability of the operator predicated on a confidence level of 95% and a false alarm rate of 5% has been estimated on the
basis of a material balance period of 1 year between two successive physical inventory takings. With a hidden inventory of 1.5 kg Pu, the throughput can be increased to 1000 kg Pu.

b) Abrupt diversion

(i) To maintain his detection capability, the operator must keep the hidden inventory under control. It has been observed that a level of about 2.5 kg Pu is maintained by taking appropriate cleaning actions when this level is exceeded.

(ii) In these conditions, the minimum detectable level is

\[ 8 - 2.5 = 5.5 \text{ kg Pu} \]

and the throughput of the plant could be raised to 800 kg Pu per year.

(iii) At the higher throughputs, the detection criterion is no longer strictly fulfilled, unless higher false alarm rate is tolerated. However, owing to the repetition of the operations every two weeks, the risk of later detection is probably so high as to deter such a diversion attempt.

These estimated throughputs constitute upper limits for the safeguards authorities when material accountancy is used as the only means of verification.

However, because of the long transit time of the PuO2 cans and rods or fuel assemblies in the input and output storage areas, results from destructive analyses will be in the hands of the safeguards authorities long before the fuel assemblies are shipped, so that the capabilities of the safeguarding authorities will approach the operator's capabilities.

4.5 Research and development activities

One of the basic tools of the safeguards system is the independent verification of the operator's data (source data).

In study No.1 of the Belgian support programme, proven means and techniques of achieving independent verification will be combined. Overall capabilities of the proposed approaches will be compared taking into account several scenarios ranging from protracted to abrupt diversion. The study will be carried out step by step, considering the reference plant firstly as a separate unit and later on as an integrated part of a fuel cycle fully safeguarded as understood by the NPT.
TABLE VI. CEN/SCK (7 MBAs): 1981 OPERATORS' EFFORT (man-days)

<table>
<thead>
<tr>
<th></th>
<th>Preparation</th>
<th>Inspection</th>
</tr>
</thead>
<tbody>
<tr>
<td>Routine inspections (57)</td>
<td>102</td>
<td>120</td>
</tr>
<tr>
<td>Physical inventories (8)</td>
<td>67</td>
<td>52</td>
</tr>
<tr>
<td></td>
<td><strong>169</strong>*</td>
<td><strong>172</strong>*</td>
</tr>
<tr>
<td>Daily book-keeping</td>
<td>811</td>
<td></td>
</tr>
<tr>
<td>Administration</td>
<td>128</td>
<td></td>
</tr>
</tbody>
</table>

*Mainly due to international safeguards

The study will be conducted by CEN–SCK, JRC (Ispra), Belgonucléaire, IAEA and Euratom.

5. CEN/SCK RESEARCH CENTRE

5.1 Introduction

The main installation of safeguards relevance at CEN–SCK are the materials testing reactor BR2 (HEU fuel), the VENUS subcritical assembly and the laboratories (Pu laboratories and post-irradiation examinations).

The site has been divided into 7 MBAs. Six of them are controlled by a near-real-time accounting system and the Pu laboratory is equipped with a real-time computerized system mainly for criticality reasons.

The number of inspections has continually increased up to 57 for all the MBAs in 1982. In addition to the item counting and identification, NDA measurements and weighings of fresh fuel assemblies and Pu cans have been effected since 1981. The inspection effort is summarized in Table VI.

5.2 R & D Activities

The following R & D work has been undertaken:

(i) A computerized information system for the management of nuclear materials has been developed. The experience gained by the use of this system is planned to be published in 1983 [18].
Research activities are conducted in the frame of the programme "Reduced Enrichment for Research and Testing Reactors" (RERTR) under the direction of A. Travelli of the Argonne National Laboratory, USA [19]. This programme aims at replacing the current 90% HEU fuel by a 20% enriched fuel.

6. CONCLUSIONS

(1) Some observations have been expressed on the safeguards effectiveness at the main Belgian nuclear facilities. The effectiveness of the nuclear material accounting by the operators has been assessed in realistic industrial conditions for the UO$_2$ and the mixed oxide fuel fabrication plants, with throughputs of 450 t of uranium and 500 kg of plutonium, respectively. It appears clear that the Agency's tentative technical objectives have been reached through the implementation of nuclear material accountancy together with the described inspection effort.

(2) Present and future R & D activities relating to the facilities in operation have been described; they concern the following matters: source data quality, source data verification by NDA and by isotopic correlation techniques, safeguards approaches to be applied to specific facilities at higher throughputs. The case of the reprocessing facility will be treated in conformity to the recommendations of the International Working Group on Reprocessing Plant Safeguards. Regarding the application of NDA to source data verification, it is essential that these techniques be tested extensively in field conditions. Most of the safeguards instruments (SAM-2, HLNCC, neutron collar) have in fact been tested in Belgian facilities, and are reported in another paper in these Proceedings [20].

REFERENCES


SAFEGUARDS AT AN AWAY-FROM-REACTOR STORAGE FACILITY

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Abstract

SAFEGUARDS AT AN AWAY-FROM-REACTOR STORAGE FACILITY.

The Swedish Nuclear Fuel Supply Company (SKBF) is at present constructing an away-from-reactor (AFR) facility for the temporary storage of spent fuel. The facility is planned to commence operation in 1985 and will serve all reactor units in Sweden. The storage pools are located 25 m deep in a rock cavern and have a capacity of 3000 tonnes of uranium. After the temporary storage, the fuel assemblies will either be encapsulated for direct terminal disposal as waste, or be sent for reprocessing. The alternative to be applied is not yet decided. A safeguards system designed to be applicable in both cases has been worked out by the Swedish Nuclear Power Inspectorate and SKBF. The system is founded on precise and complete information about the nuclear material, identification and NDA measurements with advanced technical surveillance systems as means to reduce inspection effort. The presentation of the system is the main purpose of this paper. An AFR storage facility like the CLAB is argued to be of the same safeguards category as an at-reactor (AR) storage facility.

1. INTRODUCTION

The outcome of the national referendum on nuclear power in 1980 called for the operation of a nuclear power system consisting of a maximum of twelve reactor units through to the year 2010. At present there are ten reactors in operation. Decommissioning of the nuclear reactors is planned to start around the year 2010.

Compared with previous plans, international reprocessing capacity has decreased, and as a result the number of irradiated fuel assemblies in storage at the reactor stations (AR storage) has increased significantly. High density racks have been installed
FIG. 1. Cutaway view of the CLAB facility.

- FUEL ELEVATOR: Transport of spent fuel between reception and storage.
- FUEL STORAGE: The spent fuel is stored in water-filled pools.
- AUXILIARIES: Cooling and clean-up systems, ventilation, waste handling.
- OFFICE BUILDING: Offices, canteen, washing, and dressing rooms.
- ELECTRICAL BUILDING: Electrical power and control systems.
- GARAGE AND STORES: Garage and maintenance shop for transport vehicles.

Note: The diagram illustrates various components and processes involved in the CLAB facility.
in the AR storages, but their capacities are nevertheless expected to be inadequate. The Swedish Nuclear Fuel Supply Company (SKBF), jointly owned by the Swedish power utilities, bears technical responsibility for the back end of the nuclear fuel cycle. In May, 1980, SKBF started the construction of an Away-From-Reactor (AFR) storage facility called CLAB at the Oskarshamn power plant site. Spent fuel from all the Swedish power reactors will be intermediately stored here. A specially designed ship will transport the fuel from the reactors to the CLAB facility, which will be put into operation in 1985.

It is a Swedish tradition that regulatory matters be discussed between the facility operator and the national supervisory authority before regulations are issued. It is our experience that this ensures an effective fulfilment of both safeguards and safety objectives. Such early discussions have been particularly important in the case of the CLAB facility, since it represents a new type of facility in the nuclear fuel cycle. Moreover, safeguards principles for an AFR storage facility have not really been discussed internationally. The present paper discusses how these matters have been dealt with for the CLAB facility.

2. CLAB - STORAGE AND TRANSPORTATION

2.1 Design

The design of the AFR-CLAB facility (fig 1) is based on a temporary storage capacity of 3 000 tonnes of uranium with the possibility of extending the capacity to 9 000 tonnes, which would exceed the total amount of uranium that will be generated in the Swedish nuclear programme stipulated by Parliament. In terms of functions, the facility can basically be divided into three main parts: fuel reception area, storage section and auxiliary systems. The fuel reception area and auxiliary buildings are located at ground level and the storage pools in a rock cavern. Storage of the spent fuel in an underground rock cavern ensures that the environment will be well protected by the bedrock in the event of external impact. The pools in the rock cavern consist of a free-standing concrete monolith divided into four pools, each with a capacity of 750 tonnes of uranium, plus a
central pool (reserve) connected to a transport channel from the fuel elevator. The rock cavern has a length of about 120 m, a width of 21 m and a height of 27 m. The depth from ground level to the ceiling of the cavern is 25 m. The facility's receiving capacity is 300 tonnes/year. Fig 3B shows the construction site as of September 1982.

2.2 Transportation

As the Swedish nuclear power plants and the CLAB facility are located on the coast, SKBF started in 1978 a project for the development of a transportation system based on transports by sea. The system will be able to transport the spent fuel and all types of radioactive material stemming from the reactors and the back end of the fuel cycle.

The transportation system consists of a roll-on roll-off vessel, fig 2, transport casks for spent fuel, several types of other transport containers for reactor waste, and a terminal transport system. All components included in the transportation system are currently being manufactured. The system is planned to be put into operation during the latter part of 1982.

The vessel is being built to comply with international conventions, Swedish and French national regulations for unrestricted worldwide service including the Baltic Sea, and with the highest requirements of the classification societies. The vessel is designed in accordance with the most restrictive IMCO (Inter-Governmental Maritime Consultative Organization) rules concerning "damage floatability" for ships carrying dangerous chemicals in bulk.

All transports of spent fuel casks within the nuclear power plants and the CLAB facility and between the sites and the vessel will be performed by a self-propelled vehicle. The vehicle with its cargo drives to the harbour and over the aft ramp directly onto the ship. The transport frame including the cask is placed in position, hydraulically lowered and then lashed down to the cargo deck. This procedure guarantees a simple and safe method of loading and unloading the casks. No lifting devices are required and the turn-around cycle for each cask is kept to a minimum,
which also results in a very low dose commitment to the personnel. The main components in the transportation system are shown in fig 3.

2.3 Mode of operation

The CLAB facility is being built for a receiving capacity corresponding to 300 tonnes of uranium per year. 15-20 deliveries per year, each with 5 fuel casks, are foreseen. This receiving capacity will be achieved by operating the facility on a semi-continuous basis. The transport cask that will be used weighs about 80 tonnes and can hold 17 BWR or 6 PWR fuel assemblies. Each transport cask will thus carry 3 tonnes of uranium. After unloading from the transport cask, the fuel will be placed in storage canisters. 16 BWR or 5 PWR assemblies can be stored in each canister. After loading, a handling machine will move the canister to the fuel elevator. The elevator cage, which is filled with water, will be lowered down to the main storage chamber. Canisters with fuel assemblies from several deliveries can be stored temporarily in the receiving section. The flow of nuclear material at the CLAB facility is shown in fig 4.
TRANSPORT VESSEL

A. TRANSPORT VESSEL

B. CLAB

1. FUEL ARRIVAL
2. AUXILIARIES
   - Cooling and clean up systems
   - Ventilation, waste handling
3. OFFICE BUILDING
   - Offices, canteen, washing, and dressing rooms
4. ELECTRICAL BUILDING
   - Electrical power and control systems
5. GARAGE AND STORES
   - Garage and maintenance shop for transport vehicles
6. FUEL RECEPTION
   - Equipment for receiving, cooling, cleaning and unloading of transport casks
7. ACCESS TUNNEL
   - To fuel storage

C. TRANSPORT EQUIPMENT FOR TRANSPORT CASK TN17 MK2

D. TRANSPORT CASK TN17 MK2

**TECHNICAL DATA**

<table>
<thead>
<tr>
<th>ORIG.</th>
<th>87 1/2 m</th>
<th>18 00 m</th>
<th>6 05 m</th>
<th>4 00 m</th>
<th>1 900 tons</th>
<th>1 200 tons</th>
<th>2 x 1550 HP</th>
<th>11 km/h</th>
</tr>
</thead>
</table>

**TECHNICAL DATA**

<table>
<thead>
<tr>
<th>TYPE</th>
<th>PMR</th>
<th>PMR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel assemblies weight (max)</td>
<td>6 900 kg</td>
<td>3 900 kg</td>
</tr>
<tr>
<td>Basket + fuel support weight (max)</td>
<td>7 000 kg</td>
<td>5 000 kg</td>
</tr>
<tr>
<td>Total weight (loaded packaging) (max)</td>
<td>78 000 kg</td>
<td>78 000 kg</td>
</tr>
<tr>
<td>Total heat output</td>
<td>43 5 kW</td>
<td>43 5 kW</td>
</tr>
<tr>
<td>Number of fuel assemblies</td>
<td>17</td>
<td>6</td>
</tr>
</tbody>
</table>

FIG. 3. A. Aft ramp of the transport vessel. B. Picture of the construction site, as of September 1982. C. Site transport vehicle for the spent

NILSSON and GUSTAFSSON
FIG. 4. Block diagram of the nuclear material flow in the CLAB facility.
2.4 Destiny of the fuel after intermediate storage

The CLAB facility is a temporary storage facility which is being built in order to bridge the time gap until a terminal storage system has been developed and put into operation. An important factor here is uncertainty regarding the future handling route for the terminal disposal of waste. Spent fuel assemblies may either be encapsulated for direct terminal disposal as waste, or be sent for reprocessing, whereby the nuclear material will be separated.

3. LEGISLATIVE BACKGROUND OF SWEDISH SAFEGUARDS

As a member of the Treaty on the Non-Proliferation of Nuclear Weapons, Sweden has committed itself to a comprehensive international control of all nuclear material processed, used, stored or otherwise kept in Sweden. Since 1975, when the NPT safeguards agreement [1] between Sweden and the International Atomic Energy Agency was signed, all nuclear material in Sweden has been under the Agency's control. However, Sweden had decided long before not to produce nuclear weapons. A commitment to the non-military use of nuclear material is a well-known condition in the bilateral agreements that regulate the supply of nuclear materials, facilities etc.

Legally, the use of nuclear energy is regulated by the Atomic Energy Act. Major decisions under this Act, such as the granting of licences to construct nuclear facilities or to process large quantities of nuclear material, are made by the Government. All licenses are granted conditionally, i.e. the licence is valid only if certain rules and regulations are followed. The Swedish Nuclear Power Inspectorate issues these regulations and has been appointed by the Government as the national supervisory authority. Its responsibilities include nuclear safety, nuclear waste management and safeguards, including physical protection.

In the field of safeguards, the Inspectorate has issued regulations for accountancy for and control of nuclear materials. The regulations are composed of
two parts, one administrative and one technical, for, e.g., reactors, bulk handling facilities or storage facilities. The administrative part is of a general nature and applies in whole or in part to all facilities. Together, the two parts make up the Swedish "State System for Accountancy and Control" (the SSAC). For further details, see Nilsson, 1980 [2].

4. SAFEGUARDS CONSIDERATIONS

Only a few years ago, the peaceful nuclear fuel cycle included no AFR storage. Spent fuel derived from nuclear reactors was planned to be shipped directly for reprocessing after a short "cooling period" in a spent fuel pool. This situation has changed considerably in recent years. The amount of spent fuel at the reactors is increasing each year, as is the storage time. Reprocessing is no longer taken for granted as the next step in the nuclear fuel cycle.

This change of status for the spent fuel has also been reflected in the safeguards systems applied internationally. Originally, and in many safeguards systems currently in use, all nuclear material at a reactor has been accounted for at its unirradiated value, i.e. according to the nuclear material specifications stated by the fuel fabrication plant. Burn-up, plutonium production, etc. have been kept as source data and reported to the safeguards authority only upon shipment. In new safeguards systems, e.g. those documented in new IAEA model facility attachments, the reactor facility has been divided into two functional units, although these have not been given separate MBA labels; the reactor itself and the spent fuel storage system. Fresh fuel and fuel inside the reactor will still be accounted for with pre-irradiation data. But as soon as fuel is withdrawn from the reactor and placed in the spent fuel pool, post-irradiation data will be calculated and reported. Such internal changes within an MBA are normally not reported. When introduced they mark an internal boundary, a change from one status to another. Shipment from the reactor will later be made upon actual, already recorded nuclear material content. However, after a long storage time, the content of plutonium should be adjusted due to decay of 241Pu.
An AFR storage is, in principle, not different from an AR storage. No external changes are imposed upon the fuel. The system of safeguards applied should therefore have much in common with that applied for an AR storage facility. However, after storage in the CLAB facility, the fuel will either be reprocessed or deposited as waste. The fact that this is an open question for this fuel is of great importance when the safeguards system is to be designed.

For any fuel that is sent for reprocessing to another country, both the owner of the fuel and the responsible safeguards authority ought to have an obligation to know, as exactly as possible, the content of nuclear material in each fuel assembly. The authority should also be able, independently, to verify quantities of nuclear material stated by the owner and operator.

For the fuel that is to be deposited directly as waste, the safeguards measures could be simpler. Itemized control, i.e. control measures to ensure that fuel assemblies received are the same as those shipped for disposal, can be used. In this situation, precise, updated information on the content of nuclear material in each fuel assembly is of less importance.

The system must provide a safeguards level adequate for the reprocessing alternative but so designed that the inspection effort required is about the same as for a strict itemized control.

5. SAFEGUARDS REQUIREMENTS

The system of safeguards for the AFR-CLAB storage facility, should fulfil the following objectives:

- The basic design of the facility should be conducive to the application of safeguards

- Identification of fuel assemblies should be possible at any time both in the receiving area and in the main storage section
- Effective control of separate fuel rods should be provided for
- Design and equipment should allow NDA measurement
- Follow-up calculations of isotopic composition should be possible
- The accountancy system should comply with the principles laid down in the SSAC.

Some of these objectives are discussed below.

5.1 Basic design features

Discussions between the Inspectorate and SKBF on how to fulfil these objectives started as early as 1978, when the operator presented a preliminary design to the Inspectorate. The main purpose was to find out if the basic design had to be modified in order to allow for effective safeguards. It was found that two adjustments had to be made.

In order to fulfil the objective of identification at any time, the canisters used to store fuel assemblies had to be designed with an uncovered opening. Unless they were open on the top, fuel assemblies in the main storage section could not be identified. The final design of the canister met this requirement. Secondly, in order to permit NDA measurements of fuel assemblies, one wall of one of the pools in the receiving area had to be made with a hole covered with a steel tube, in which the collimator used in the measurements could be placed.

Since these changes were introduced before the designs were finalized, additional costs due to safety requirements were minor.

Discussions concerning basic safeguards principles were also started at that time. This allowed a consensus to be reached between the Inspectorate and SKBF on all major parts of the safeguards system to be applied.

We would have appreciated the participation of the International Atomic Energy Agency in these
discussions. However, according to the routines presently followed within the Agency, the Agency is first introduced to a new facility as late as six months before the start of operation, in this case before the initial receipt of fuel, when the Facility Design Information is to be submitted to the Agency. By that time it is no longer possible for the Agency to influence the design of the facility or its equipment in favour of international safeguards. Accordingly, the application of the Agency's safeguards system has to be based upon a fixed design.

5.2 Accountancy and documentation

In the Swedish SSAC, great emphasis is placed on integrated measures for all facilities in the nuclear fuel cycle. The AFR storage facility is, in our view, only an intermediate station for the fuel, and it is important that the same basic control principles be applied here as well.

All data on the fuel must be on record at the facility or otherwise available. The following information should be included for each item, i.e. fuel assembly or fuel rod.

The accountancy records will include:

- content of total uranium and total plutonium, stated with measurement accuracy
- isotopic composition
- irradiation data from the reactor facility
- receiving and shipping documents.

The operating records will include:

- precise location at each time; identification No. of canister or capsule in which the fuel assembly or fuel rod is stored and its storage position
- working documents for all handling procedures such as leakage test, dismantling operation, NDA measurements.

Separate records will be kept for the continuous recording of receipts, shipments and other
inventory changes. Furthermore, the records shall show the bilateral obligation of the nuclear material and other control labels, if any. Handling of fuel rods in an assembly, e.g., exchanges, redistribution etc., is considered to be an inventory change.

It is up to the facility to decide about computerizing the safeguards information. For the CLAB facility, a preliminary decision has been taken to computerize as much of the accountancy records as possible. Nevertheless, receiving, shipping and working documents have to be handled manually.

From the above it is clear that the operator of the CLAB facility will be required to have access to previous data on the fuel received. The isotopic composition of nuclear material in spent fuel has hitherto been calculated at the reactor station, but has not been reported to the authority or otherwise used for safeguards purposes. So far there has been no practical and valuable safeguards application for isotopic composition. We believe, however, that the information on isotopic composition will be of major importance in the safeguards measures that are foreseen for spent fuel in the CLAB facility.

The isotopic composition of plutonium will change over a time period of 30 years. The amount of plutonium-241 will decrease significantly for example. These changes have to be monitored and the facility will have to be equipped to permit recalculation of isotopic composition. This is particularly important prior to shipment for reprocessing.

Reports to the Agency will be prepared by the Inspectorate as for other facilities. The reports will be based upon continuous reporting of inventory changes from each facility to the Inspectorate in accordance with the Swedish SSAC routines.

5.3 Fuel rod handling

At the reactor station, the number of fuel rod replacements is continuously increasing, and thereby also the number of fuel rods stored in the spent fuel pool. Even though some of the fuel rods will be reused in new fuel assemblies, an increasing number of fuel rods will eventually have to be stored
in the CLAB facility. Leaking or otherwise damaged rods will be put into special capsules. The capsules will then be placed in a canister for storage in a pool in the main storage section. The capsules will be closed and it will not be possible to check the number of rods in each capsule that has been closed.

Fuel rods of Swedish design used in BWR fuel assemblies have unique identification numbers. However, this identification number cannot be read under water. The only possibility to obtain complete itemized control of fuel rods is therefore to check the content of each capsule when it is loaded. At present, there is no way to fulfill this objective of verification of fuel rods. Since fuel rods contain nuclear material in quantities that cannot be neglected, this problem will have to be considered seriously within the near future and in any case before the CLAB facility is taken into operation.

It is assumed that dismantling of fuel assemblies will take place in the CLAB facility. It may be necessary to withdraw rods after leakage tests or other checks performed by the operator.

The IAEA Facility Attachments for reactors contain the requirement that advance notice be given to the Agency 30 days before any fuel assembly is dismantled. Such a requirement could be most difficult to fulfill for the CLAB facility, since dismantling may have to be done in connection with fuel receipts. Discussions are needed here in order to arrive at a solution satisfactory to all parties concerned.

5.4 NDA measurements

In order to be able independently to verify burn-up and nuclear material content in spent fuel, the Inspectorate started a project in which the Agency has participated, to develop a non-destructive method for this purpose. The method has been tested at the Oskarshamn nuclear power station, where nine assemblies were tested [3]. The results obtained have been encouraging for future application, particularly at the CLAB facility, but much more development work is needed.

All NDA measurements will take place in a separate pool in the receiving area. The instrument
and the detector will be placed on the outside of the wall while the fuel assembly to be measured will be placed in a special rig in the pool. The rig will be able to lift the assembly as well as rotate it in order to expose it in all ways. The principles of the arrangement for measurements are shown in fig 5.

No measurements will be allowed in the main storage section. The amount of equipment placed in the storage section, as well as the time for which personnel are exposed to radiation, has to be kept to an absolute minimum.

FIG. 5. Cutaway view of the arrangement for non-destructive measurements, in the component pool in the receipt area.
5.5 **Identification of fuel assemblies**

The identification of nuclear material at the CLAB facility is based on use of TV cameras and monitors. The handling machines in both the receiving area and the storage section will be equipped with such cameras.

Normally the identification number of a fuel assembly is easily legible on a TV monitor. In some cases, however, after several years of radiation, the identification number might be illegible. Additional safeguards may therefore have to compensate for the lack of an identification number on these assemblies. These measures will have to be determined separately in each case.

6. **VERIFICATION ACTIVITIES**

An independent verification of nuclear material in spent fuel at the CLAB facility should include the following measures:

A) identification  
B) check of shipping document  
C) NDA measurement.

As mentioned, 15-20 deliveries are expected to be received each year. The fuel assemblies will be loaded into storage canisters. There is a pool for the temporary storage of 30 loaded canisters in the receiving area. Fuel assemblies from several deliveries could be stored temporarily in this pool awaiting transfer to the main storage section. The transfer of fuel from the receiving area down to the storage section could therefore be done in campaigns comprising several deliveries. This arrangement makes it quite possible to identify all fuel assemblies received while they are still in the receiving area. If the fuel has already been transferred at the time of inspection, it is still possible to perform identification in the main storage section. However, any problem with identification could not be compensated by measurement, since this can only be done in the receiving area.
A) and B) above should apply to all fuel received. C), verification by NDA measurement, will be applicable only to fuel that is to be reprocessed.

For fuel originating from Swedish reactors, NDA measurements will be postponed until a decision is taken concerning shipment for reprocessing. If fuel is to be received from other countries, there may be reason to perform check measurements at the time of receipt.

With this arrangement, the verification activities at the CLAB facility will equal itemized control. If the fuel assemblies are to be deposited as waste, this is all that is needed. On the other hand, if reprocessing is planned, the verification activities could easily be extended to include independent verification of the nuclear material.

At the CLAB facility, all movements, handling operations, etc. are performed in the receiving area. The situation in the main storage section is much more stable, since only internal transportation could take place there. We therefore believe that safeguards efforts should be concentrated on the receiving area. The main storage section seems to be well suited for reliable systems of technical surveillance, which in itself provides a high level of safeguards.

7. TECHNICAL SURVEILLANCE

Technical surveillance measures play an important role in the itemized control administered by the Agency. They are introduced in order to maintain a defined control level but with less inspection effort. Cameras are used to record all movements and provide unquestionable evidence of unannounced shipments of spent fuel. However, it must be a reliable camera system that requires only a very limited amount of service. Such cameras do not seem to be available at present. The development of reliable advanced surveillance cameras is therefore an urgent task in safeguards development.

At present, the Agency performs reverification of the inventory of nuclear material in case the
surveillance system fails. A reliable reverification may include verification of nuclear material content as well. However, the failure of a surveillance camera will not be considered reason enough to start internal movements for the purpose of measuring significant quantities of spent fuel. If, however, reverification is judged to be necessary due to inspection results, etc., no effort will be spared in order to assist the Agency.

Seals used as surveillance equipment should, apart from being tamper-resistant, be easy to apply and verify. If these criteria are met, there are three possibilities for sealing in the storage facility; the cassette, the handling machine or all or part of a pool. To apply and check seals under water is very difficult. The movement of the handling machine could be blocked in various ways, but its upper part is equipped with additional tools that still could be used to lift and redistribute cassettes or single fuel assemblies. To seal a storage pool will most probably cause considerable inconvenience for the operator.

To sum up the possibilities of technical surveillance, it must be concluded that major difficulties may be associated with seals. An advanced camera system seems to be a better alternative for the CLAB facility, but will have to carry a heavier surveillance burden than if seals could also be used.

8. ROUTINE INSPECTION EFFORT

For each facility subject to the Agency's system of safeguards, a maximum routine inspection effort is determined. The formula used for this purpose is defined in § 80 of INFCIRC/153. In this section, facilities are grouped into three categories depending upon their strategic value.

Spent fuel in an AR storage belongs to the reactor and the maximum routine inspection effort will be that defined for the reactor, namely one sixth of a man-year of inspection. An AFR storage facility, however, does not come under this category, although in principle it is the same type of storage facility. It could be argued that the maximum routine inspection
effort for an AFR storage facility is determined by § 80b, i.e. \(30 \cdot (\text{inventory or throughput (eff. kg.)})\), due to its content of plutonium.

This would mean that nuclear material in the same physical form and with the same availability is regarded to be of quite different significance from the viewpoint of safeguards depending upon its storage place. Since the AFR storage facility is a new type of facility, it is questionable as to whether it was considered when § 80 was written. Normally, § 80b is used to determine the maximum routine inspection effort for facilities for the handling and storage of readily available plutonium or highly enriched uranium.

Considering the nature of nuclear material stored in an AFR storage facility, it would seem more appropriate to include the AFR storage facility in § 80a.

9. FINAL REMARKS

We believe that the open cooperation between SKBF and the Inspectorate regarding safeguards at the AFR storage, the CLAB facility, has been fruitful for both parties. We have reached agreement upon all major points in the safeguards system. Our basic, common interest in safeguards has been a great advantage especially since there has been no complete safeguards system for an AFR storage to compare with.

However, as mentioned previously, there are still some points and problems to be solved, among which we would like to emphasise the development work needed in the fields of technical surveillance and non-destructive measurement. For all cases, we are confident that the final safeguards system will be applied in harmony with the operation of the facility.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, The Structure and Content of Agreements Between the Agency and States Required
in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons, INFCIRC 153 (1972).


NEAR-TERM AND FUTURE DEVELOPMENTS OF LMFBR SAFEGUARDS*

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Abstract

NEAR-TERM AND FUTURE DEVELOPMENTS OF LMFBR SAFEGUARDS.

The objective of the study is to perform a diversion analysis and an assessment of the available safeguards methods and systems for verifying inventory and flow of nuclear material in accessible and inaccessible areas of liquid-metal fast breeder reactor (LMFBR) systems. The study focuses primarily on the assembly-handling operations, assembly storage facilities, and reactor operations facilities relating to experimental, demonstration and prototypical reactor plants. The safeguards systems and methods presented are considered to be feasible for development and implementation within the resource limitation of the IAEA and are considered to be consistent with the objectives, requirements, and constraints of the IAEA documents INFCIRC/153 and INFCIRC/66/Rev.2.

1. INTRODUCTION

Safeguards for nuclear fuel cycles should be considered in the context of a world deployment of various reactor types and a varying availability of fuel cycle services. This is particularly true of the current experimental, demonstration, and prototypal fast breeders, which must be fuelled and safeguarded in the transition period. The quantities of fissile materials, the in-air and in-sodium fuel handling systems, and the storage methods for the fuel assemblies will have a significant impact on safeguards techniques. The total quantity of nuclear material in inventory and in flow for each demonstration and commercial breeder reactor system would be 3 000–10 000 kg of plutonium,

* Work performed under the auspices of the US Department of Energy, Office of Safeguards and Security and the International Safeguards Program Office.
<table>
<thead>
<tr>
<th></th>
<th>Experimental Systems</th>
<th>Demonstration Systems</th>
<th>Commercial Systems</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>&lt;100 MWt</td>
<td>&gt;100 MWt</td>
<td></td>
</tr>
<tr>
<td><strong>Number of Assemblies</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Inner Core</td>
<td>60</td>
<td>30-80</td>
<td>60-110</td>
</tr>
<tr>
<td>Outer Core</td>
<td>200</td>
<td>50</td>
<td>50-90</td>
</tr>
<tr>
<td>Blanket</td>
<td>90</td>
<td>200-325</td>
<td>90-170</td>
</tr>
<tr>
<td><strong>Fuel Elements/Ass.</strong></td>
<td>90</td>
<td>200-325</td>
<td>160-220</td>
</tr>
<tr>
<td><strong>Uranium Elements/Blanket Ass.</strong></td>
<td>30</td>
<td>80-90</td>
<td>80-90</td>
</tr>
<tr>
<td><strong>Nuclear Fuel Content (Fresh/Irradiated)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Material Form</td>
<td>PuO₂/UO₂</td>
<td>PuO₂/UO₂</td>
<td>PuO₂/UO₂</td>
</tr>
<tr>
<td>PuO₂/UO₂</td>
<td>2-14</td>
<td>30-50</td>
<td>30-55</td>
</tr>
<tr>
<td>Pu₂3⁵U, kg/Ass.</td>
<td>1-3 Pu, 2-3 23⁵U</td>
<td>8-13 Pu, 10-12 23⁵U</td>
<td>6-14</td>
</tr>
<tr>
<td>Pu₂3⁵U, kg/Element</td>
<td>0.03-0.07</td>
<td>0.03-0.04</td>
<td>0.03-0.06</td>
</tr>
<tr>
<td>Irradiated Blanket</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pu, kg/Ass.</td>
<td>1</td>
<td>1-3</td>
<td>1-3</td>
</tr>
<tr>
<td>Pu, kg/Element</td>
<td>0.03</td>
<td>0.01-0.04</td>
<td>0.01-0.04</td>
</tr>
</tbody>
</table>
which is in the range of the annual flow rates of 6 000-12 000 kg for large LWR reprocessing plants. The content of the fissile material, 5-15 kg of plutonium, in each of the fresh and spent fuel assemblies is on the order of the IAEA significant-quantity guideline of 8 kg.

A timely perspective for developing LMFBR safeguards is obtained by considering that the existing experimental reactors have plutonium flow rates and inventory quantities (ranging from 1/2 to 2 tonnes of plutonium) which are comparable to some current reprocessing and mixed oxide fabrication plants having flow rates of about 200 tonnes of heavy metal (1 to 2 tonnes of plutonium) per year.

2. REFERENCE LMFBRs AND FUEL HANDLING SYSTEMS

The design features of LMFBR systems are categorized into two general classes: (1) demonstration/commercial and (2) experimental. The basic units of one fuel assembly and one blanket assembly have been adopted for item accountability with reactors in the normal operating mode. However, for the experimental reactors, experimental programs and experimental-assembly designs could necessitate the more restrictive item accountability by fuel element (pin). The core and blanket design details for the demonstration/commercial and the experimental reactors are summarized in Table I.

3. DIVERSION ANALYSIS

The significant safeguards concerns for LMFBR systems are the accountability, surveillance, and assembly identification of fuel and blanket assemblies during the following fuel cycle activities: unloading the fresh fuel and blanket assemblies from the transport casks into storage cells with an air atmosphere, transfer into other cells for storage under sodium, loading the assemblies into the reactor vessel and reactor core, unloading the spent fuel and blanket assemblies from the reactor into temporary storage pools for planned decay-heat cooling periods, and finally removing the assemblies from the storage pools, and transferring the assemblies to the shipping cask [1-3]. Both the fresh and the irradiated fuel assemblies, each containing about equal masses of plutonium, are submerged in sodium throughout most of the handling and reactor operations facilities. The total time interval from receiving fresh fuel and blanket assemblies to shipping the spent fuel and blanket assemblies may be on the order of 1-2 yr for experimental reactors and 5-7 yr for demonstration/commercial reactors.
<table>
<thead>
<tr>
<th>Nuclear Material</th>
<th>Significant Quantity (SQ)</th>
<th>Number of Assemblies</th>
<th>Number of Elements</th>
<th>Conversion Time</th>
</tr>
</thead>
<tbody>
<tr>
<td>Non-Irradiated Fuel PuO₂/UO₂</td>
<td>8 kg Pu</td>
<td>1</td>
<td></td>
<td>110-320</td>
</tr>
<tr>
<td>Non-Irradiated Fuel ²³⁵UO₂/UO₂</td>
<td>25 kg ²³⁵U</td>
<td>2-5</td>
<td></td>
<td>order of 1-3 weeks</td>
</tr>
<tr>
<td>Irradiated Fuel PuO₂/UO₂</td>
<td>8 kg Pu</td>
<td>1</td>
<td></td>
<td>110-320</td>
</tr>
<tr>
<td>Irradiated Fuel ²³⁵UO₂/UO₂/PuO₂</td>
<td>8 kg Pu + (0/25) ²³⁵U</td>
<td>2-5</td>
<td></td>
<td>220-1600</td>
</tr>
<tr>
<td>Irradiated Blanket PuO₂/UO₂</td>
<td>8 kg Pu</td>
<td>2-8</td>
<td></td>
<td>220-800</td>
</tr>
</tbody>
</table>
The diversion analysis for the design of the safeguards approach must include two general extreme diversion strategies: (1) the abrupt diversion of at least one assembly, and (2) the protracted diversion of a significant quantity of nuclear material by the removal of some fuel elements from assemblies over an extended time period.

Abrupt diversion includes the following considerations (listed in the order of probable occurrence): 1) Substitution of unirradiated fuel assemblies with counterfeit or dummy assemblies not containing fissile nuclear material, 2) Substitution of unirradiated fuel assemblies with dummy assemblies containing fewer fissile fuel elements (pins), 3) Interchange of fuel assemblies with blanket assemblies, 4) Interchange of fuel assemblies with assemblies having a lower concentration of nuclear fissile material, and 5) Substitution of irradiated fuel assemblies with dummy assemblies.

Protracted diversion of fuel elements (pins) would be a safeguards concern only for reactors with on-site assembly dismantling and examination facilities to evaluate fuel performance.

The safeguards approach includes selection of material balance areas (MBAs) and key measurement points (KMPs) for physical inventory and flow to verify: (1) nuclear material flow from receipt of fresh assembly to dispositions of intact and non-intact irradiated assemblies or fuel elements; and (2) physical inventory in storage areas such as receiving vessels, buffer storage tanks, preparation or conditioning cells, and sodium and nonsodium storage tanks for unirradiated and irradiated assemblies. To maintain continuity of knowledge for verification of the flow and physical inventory of assemblies and elements, C/S measures would supplement the item-accountability at the KMPs.

The current IAEA goals for "significant quantity," SQ, and the proposed guidelines for conversion time as applied to the demonstration/commercial LMFBR reference fuel handling systems are outlined in Table II.

3.1. Item Flow in Fuel Handling System

3.1.1. Item Accountability

The item accountability of fuel and blanket assemblies in the fuel-handling storage or reactor operations is measured by adding all receipts to and subtracting all removals from the initial inventory. During the normal flow of assemblies, nuclear materials control emphasizes administrative controls supplemented
FIG. 1. Reference assembly flow plan for fuel handling systems of demonstration and commercial LMFBR plants.
by secure storage for the assemblies, sealed containers or transport casks. Although these methods of assembly inventory are effective for control of nuclear materials, there exist inherent limitations in both sensitivity and timeliness. Limited measurement capabilities and the uncertainties of the nuclear material content of the assemblies may obscure the diversion of significant quantities of nuclear materials. Timeliness may be limited by practical difficulties of accessibility, economics of operation, and consequent reduction in the frequency of item inventory verification.

Item accountability and the accountability of the nuclear material contained in the assembly are equivalent only when there is assurance that the structural integrity of the assembly and its contents has been maintained. The use of neutron interrogation techniques for nondestructive assay (NDA) would provide direct assurance that the integrity of the nuclear material content has not been compromised since the fabrication of the assembly.

3.1.2. Containment and Surveillance (C/S)

The principal containment and surveillance function applicable to item accountability is monitoring of assembly storage areas, assembly flows, and assembly handling. The C/S measures would produce a record of any unplanned changes in the item inventory by application of closed-circuit TV (CCTV), radiation monitors, and loading sensors on assembly handling machines. C/S measures include the monitoring of strategic areas not specifically planned for storage of nuclear material. The integrity of IAEA instrumentation, devices, and seals would be supported by surveillance records.

3.1.3. Operator Fuel Handling System Control Points (OCP)

The operational stages of the fuel handling system are designed for moving assemblies within the plant. The operation consists of transferring the assemblies into and out of the reactor and, by crane and transport casks, to adjoining supporting facilities for storage, inspection, decontamination, and shipment to off-site processing plants.

3.1.3.1. OCP Reference Plan for Demonstration and Commercial Reactors

Figure 1 represents a reference assembly flow plan for the fuel handling systems of demonstration and commercial reactors. Each major stage in the handling sequence is identified with a listing of the operating functions at that stage and denoted as a numbered operator control point (OCP).
3.1.3.2. OCP Accounting of Assemblies

Operation of all fuel handling systems is automated except those involving unloading of fresh fuel assemblies. The automated systems control and record the operation of each handling mechanism. These programming operations include the positioning of the rotating reactor plugs, fuel handling grippers, fuel transfer mechanisms, and storage racks. The movements, transfers, and positions for each assembly are recorded and stored in a computer-based data-acquisition-and-retrieval system. The data bank contains the detailed identification and fuel content information for each assembly and should be incorporated into the inspection plan.

3.1.4. Safeguards Approach: Material Balance Areas (MBAs) and Key Measurement Points (KMPs)

The diversion analysis and safeguards approach requires the selection of material-balance areas and key measurement points that would overlay and be consistent with the operators system of nuclear materials accountancy and control. The number of MBAs and KMPs will depend on the specific reactor-plant design, the plant operating mode, and the accessibility of the assemblies for implementing appropriate safeguards measures. The MBAs contain the closing inventory balance in the item accountancy of all identified assemblies.

The total LMFBR fuel handling facilities are divided into two MBAs: (1) the assembly handling and storage operations facilities (MBA-ABEF) and (2) the reactor operations facility (MBA-CD). Figure 2 represents the preliminary safeguards plan for the reference demonstration/commercial reactors.

3.1.4.1. Assembly Handling and Storage Operations Facilities (MBA-ABEF)

The MBA-ABEF incorporates all in-air handling and storage operations (which are accessible areas) external to the in-sodium storage cell and the reactor vessel (both of which are inaccessible areas). For safeguards purposes, in-air handling and storage for fresh fuel and blanket assemblies are considered essentially as extensions of the handling and storage operations of the high-plutonium-throughput fuel fabrication plant, the only difference being in the exchange of assembly ownership and possible shipper/receiver discrepancies in fuel content and specifications. Consequently, approach over this stage of the handling and storage operation is to extend to the MBA-ABEF the IAEA inspection criteria and efforts applied at the fabrication plant.
FIG. 2. Preliminary safeguards plan for reference demonstration and commercial LMFBR plants.
3.1.4.2. Reactor Operations Facility (MBA-CD)

The MBA-CD incorporates the handling and storage operations directly relevant to the power operation of the reactor. The fuel and blanket assemblies are preconditioned in the sodium environment for reactor loading. The under-sodium environment of MBA-CD presents the advantage of a "quasi-containment" system, and would facilitate the use of C/S devices to supplement safeguards system accountancy measures.

3.1.4.3. Key Measurement Points (KMPs)

The KMPs for physical inventory include the assembly storage cells in each of the MBAs. The KMPs for flow include the verification of inputs and outputs (including measured discards) at each of the MBAs. The KMPs for physical inventory are denoted KMP-(A, B...); those for flows are denoted as KMP-(1, 2...). The KMPs have been selected to form a safeguards structure with the sequence of operating stages coincident with the sequence of OCPs.

4. SAFEGUARDS MEASURES

The introduction of fuel assemblies with a high content of Pu/235U into the receiving and shipping areas of the fuel cycle does allow a more direct near-real-time assay profile of the disposition of Pu/235U. Information on the timely disposition of assembly flow is obtained from the selective application of C/S techniques [2-4] and from the material accountancy by NDA methods (HLNCC) [3,5].

4.1. C/S Methods

The safeguards techniques for containment and surveillance can use a combination of instrumentation, CCTV monitoring, fuel-assembly identification, and tamper-indicating methods such as close-up photography to record the unique signatures of characteristic weld lines. The devices considered for the monitoring of fresh and spent fuel and blanket assemblies in transit, storage, and handling are radiation monitors, load sensors on cranes and fuel handling machines, acoustic monitors, and assembly identification devices based on ultrasonics.

4.2. Direct Accountability of Fissile Material in Fresh, Spent, And Blanket Assemblies

The safeguards approach for the long-term handling and interim storage of fresh and spent fuel assemblies, the international transfer of ownership of the fuel with resolutions of shipper/receiver balances, may require direct identification of
fissile content and accountability of nuclear material and would involve a combination of C/S sensor information, inspector observations, and NDA measurements.

The characteristic of the LMFBR system of core assemblies is the almost constant plutonium concentration in reactors with core conversion ratios of approximately 0.7 and greater. Although the relative concentrations of the plutonium isotopes may vary over extended periods of burnup, the spontaneous fission (SF) neutron yields from the plutonium are almost constant over the burnup period extending beyond 100,000 MWd/t as shown in Fig. 3.

The significance of the neutron source strength is that the curium-isotope concentrations can be determined from a difference measurement of SF neutron source strengths between the fresh fuel and the spent fuel in a given assembly. The resulting curium concentration gives a direct measure of the irradiation history of the assembly. This difference-method of determining the curium concentration has the advantage of using the time decay of $^{242}\text{Cm}$ as a consistency check in the physical-inventory verification of the stored assemblies. The high concentration of $^{242}\text{Cm}$ at high burnups is unique to LMFBRs, and enables the inspector to make a timely in-field assessment of a possible anomaly in the flow of nuclear material in the assembly handling operations.

FIG. 3. Plutonium isotopic and $^{241}\text{Am}$ neutron yield as a function of burnup.
The plutonium dominates the SF neutron yield rate over the burnup periods anticipated for the blanket assemblies in all classes of reactors.

5. SAFEGUARDS INSPECTION EFFORT

The inspection activities, the frequency of inspections, and the inspection efforts are outlined in the sequence of the assembly flow within each of the MBAs.

5.1. Inspection Activity MBA-ABEF: Assembly Handling and Storage Operation Facilities

5.1.1. KMP-1: Receiving Area

This strategic point is the initial flow KMP and will require inspection activities for each receiving operation. A cask design with a 9-assembly capacity would require 11 receiving operations, and 11 inspections per year for an average 100-assembly-per-year refueling schedule. The inspection effort should be planned to cover about 11 man-days for this stage of the fresh-fuel handling sequence.

5.1.2. KMP-A: In-air Storage

The primary function of the IAEA inspector at this inventory KMP is verification activities in the in-air storage of the fresh fuel and blanket assemblies. The inspection effort may range from 17 to 34 man-days per year, if the sensor readings for assembly identification and for verification of the assembly integrity are combined with the NCC measurements. An additional 15 man-days of inspection effort are required to satisfy the average timeliness guideline of 2 weeks for the fresh fuel assemblies.

5.1.3. KMP-2: Transfer from In-air Storage to In-sodium Storage

The inspection effort at this flow KMP required to verify the ICR's from MBA-ABEF to MBA-CD is anticipated to be approximately 6-8 man-days per inspection for the transfer of 100 assemblies.

5.1.4. KMP-3: Inspection and Preparation Cell

The transfer of irradiated assemblies from the in-sodium decay cooling storage cell KMP-B (MBA-CD) to KMP-F (MBA-ABEF) for shipment off-site, or to KMP-D or KMP-E (MBA-ABEF) for extended storage, would flow through KMP-3. The assemblies are cleansed of sodium and inspected by the operator before being transferred.
5.1.5. KMP-D: In-sodium Extended Storage

For those facilities designed for in-sodium extended storage, the inspection approach would depend to a large extent on containment and surveillance, such as verification of storage seal integrity. The inventory verification would consist of assembly counting, assembly identification, and storage-position index as reported by the operator. The principal instruments for performing this inventory verification could be the under-sodium viewing (USV), or the under-water optical viewing (UWOV) modified for a sodium environment.

5.1.6. KMP-E: In-water or In-gas Storage

The inspection effort anticipated for this method of extended storage should not differ greatly from that for the in-sodium extended storage (KMP-D).

5.1.7. KMP-F: Buffer Storage for Shipment Preparation

The inspection effort in KMP-F is required to independently verify irradiated fuel and blanket assemblies placed in temporary storage in preparation for off-site shipment. A complete assay of the nuclear material content in the assemblies at this stage would provide a timely measure of fissile-fuel for credit to the reactor operator and an effective IAEA independent verification of the closing balance by assembly unit accountancy and by the nuclear material content in each assembly.

5.1.8. KMP-4: Shipping Area

Flow KMP-4 is coupled with KMP-F and should verify inventory changes with the same frequency of inspection as that assigned to KMP-F. Inventory-change verification may require 5-8 man-days per year.

5.2. Inspection Activity MBA-CD: Reactor Operations Facility

In addition to the reactor loading operations, the inspector must verify the transfer of fresh fuel and blanket assemblies from in-air storage (KMP-A) to in-sodium preparation storage (KMP-B) and verify the transfer of irradiated fuel and blanket assemblies from in-sodium storage (KMP-B) to the inspection cell (KMP-3).

5.2.1. KMP-B: In-sodium Storage of Fresh Fuel and Blanket Assemblies

For the storage of fresh fuel and blanket assemblies, KMP-B would require an inspection effort consistent with the timeliness
### TABLE III. Safeguards-inspection Effort for Material Balance Area MBA-ABEF in Reference LMFBR Plants: 100-assembly/yr Flow Rate

Material Balance Area (MBA) ABEF: Assembly Handling and Storage Operations Facilities

<table>
<thead>
<tr>
<th>Key Measurement Points</th>
<th>Inspection Activity</th>
<th>Number of Inspections per year</th>
<th>Man-days per Inspection</th>
<th>Total Man-days per year</th>
</tr>
</thead>
<tbody>
<tr>
<td>KMP-1 Receiving Area</td>
<td>Examination of shipment/receiver records and reports. Check integrity of transport</td>
<td>11</td>
<td>1</td>
<td>11</td>
</tr>
<tr>
<td></td>
<td>cask seal or assembly canister seal. Examine containment and surveillance system</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>records and servicing.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>KMP-A In-Air Storage</td>
<td>Examination of records. Cross-correlation of records and reports for consistency.</td>
<td>11</td>
<td>1.5-3</td>
<td>17-34</td>
</tr>
<tr>
<td></td>
<td>Independent verification of assembly inventory, accounting, identification, and</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>physical integrity. NDA measurements and calibration. Examine containment and</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>surveillance system records and servicing.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>KMP-2 Transfer from In-Air Storage to In-Sodium Storage</td>
<td>Examination of records. Cross-correlation of records and reports for consistency in transfer of fresh fuel and blanket assemblies from MBA-ABEF to MBA-CD.</td>
<td>2</td>
<td>6-8</td>
<td>12-16</td>
</tr>
<tr>
<td>KMP-3 Inspection and Preparation Cell</td>
<td>Examination of records. Cross-correlation of records and reports for consistency in transfer of irradiated fuel and blanket assemblies from MBA-CD to MBA-ABEF.</td>
<td>2</td>
<td>6-8</td>
<td>12-16</td>
</tr>
<tr>
<td>KMP-D In-Sodium Extended Storage</td>
<td>Examination of records. Cross-correlation of records and reports for consistency. Independent verification of assembly inventory, accounting and identification. Examine containment and surveillance system records and servicing.</td>
<td>6</td>
<td>10-20</td>
<td>60-120</td>
</tr>
<tr>
<td>KMP-E In-Water or In-Gas Extended Storage</td>
<td>Examination of records. Cross-correlation of records and reports consistency. Independent verification of assembly accounting, identification, and physical integrity. Examine containment and surveillance system records and servicing.</td>
<td>6</td>
<td>10-20</td>
<td>60-120 60-120</td>
</tr>
<tr>
<td>KMP-F Buffer Storage For Shipment Preparation</td>
<td>Examination of records, cross-correlation of records and reports for consistency. Independent verification of assembly accounting, identification, and physical integrity, NDA measurements and calibration. In the Experimental Reactors, the independent verification is to be maintained on the fuel element level of accounting, identification and physical integrity, and NDA measurements.</td>
<td>5-8</td>
<td>1.5-3</td>
<td>8-24</td>
</tr>
<tr>
<td>KMP-4 Shipping Area</td>
<td>Examination of shipment records and reports. Independent verification of assembly accounting and identification. Apply and check transport cask or assembly canister seals. Examine containment and surveillance system records and servicing.</td>
<td>5-8</td>
<td>1</td>
<td>5-8</td>
</tr>
</tbody>
</table>

Total Man-days per year
MBA-ABEF 128-228
<table>
<thead>
<tr>
<th>Key Measurement Points</th>
<th>Inspection Activity</th>
<th>Number of Inspections per year</th>
<th>Man-days per Inspection</th>
<th>Total Man-days per year</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fresh Fuel and Blanket Assemblies</td>
<td>Examination of records. Cross-correlation of records and reports. Independent verification of assembly accounting and identification. Examine records of transfer machine movements and positioning for cross-correlation with related KMPs. Examine containment/surveillance system records and servicing.</td>
<td>2</td>
<td>2-3</td>
<td>4-6</td>
</tr>
<tr>
<td>Irradiated Fuel and Blanket Assemblies</td>
<td>Examination of records. Cross-correlation of records and reports. Independent verification of assembly accounting and identification. Examine records of transfer machine movements and positioning for cross-correlation with subsequent KMPs inventory and reactor history data and loading configuration.</td>
<td>2-3</td>
<td>2-3</td>
<td>4-9</td>
</tr>
<tr>
<td>KMP-C Reactor Vessel</td>
<td>Examination of records: core loading configurations, internal charging machine movement, reactor operating data, and operator's burn-up data.</td>
<td>1&lt;sup&gt;a&lt;/sup&gt;</td>
<td>15&lt;sup&gt;a&lt;/sup&gt;</td>
<td>15&lt;sup&gt;a&lt;/sup&gt;</td>
</tr>
<tr>
<td></td>
<td></td>
<td>3-6&lt;sup&gt;b&lt;/sup&gt;</td>
<td>10-6&lt;sup&gt;b&lt;/sup&gt;</td>
<td>30-36&lt;sup&gt;b&lt;/sup&gt;</td>
</tr>
<tr>
<td>Total Man-days per year</td>
<td>Demonstration and Commercial Reactors</td>
<td>23-30</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Experimental Reactors</td>
<td>38-31</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<sup>a</sup> - Demonstration/Commercial Reactors
<sup>b</sup> - Experimental Reactors
guideline of an average 2 weeks' conversion time. However, the inspection effort may be limited to the reactor loading operations. Consequently one inspection during the in-sodium preparation (1-2 months) period may suffice. A second inspection activity could establish consistency of the records and update the information data bank prior to reactor loading.

5.2.2. KMP-B: In-sodium Storage of Irradiated Fuel and Blanket Assemblies

The storage of irradiated fuel and blanket assemblies at KMP-B requires an average inspection frequency every 2 months. Allowing 90-180 days for decay cooling suggests that 1 or 2 inspections be planned in addition to the change of inventory inspections.

5.2.3. KMP-C: Reactor Vessel

The frequency of inspection should be planned to coincide with the operator's reloading schedules. The inspection schedule may vary from 1 to 2 months for some experimental reactors to 1 yr for demonstration/commercial power reactors.

5.3. Summary Remarks on the Safeguards Effort

The inspection effort for the LMFBR systems included: optimizing surveillance-inspection activities, reviewing data acquired by sensing instruments, reviewing reactor power plant operation records for cross-correlation and consistency, and to independently verify the material inventories by item accountability and by direct measurement of material content in each item through use of NDA techniques. The approach emphasizes review and evaluation of data relating to reactor power operation, fuelling schedules, verification of unit assembly flows and storage, identification of operating anomalies, records of nuclear material shipment and receipts, and data relating to movement of operating equipment in the fuel handling system (including the indexing pattern of the nonconcentric rotating-plug system).

The estimates of the safeguards inspection effort for the MBA-ABEF and the MBA-CD are summarized in Tables III and IV, respectively. The total annual IAEA inspection effort may range from 151 to 258 man-days for the demonstration and commercial reactors to 166 to 279 man-days for the experimental reactors. The safeguards approach and inspection activities will depend on trade-offs between many factors, two of which are IAEA resources and timeliness of detection.
6. IMPLEMENTATION OF THE SAFEGUARDS SYSTEM

The safeguards systems for the LMFBR will be developed by the gradual implementation of available techniques at operating reactor systems. These techniques will be supplemented with more advanced methods as they are developed and field-tested by the Operations Division. The adoption of these improved measures would necessarily have to include the following considerations:

- the safeguards methods that are available for use in LMFBRs and the limitations of these techniques;
- the inspection time required to get effective readings for item accounting, assembly identification, verification of the integrity of nuclear material content, and verification of the assembly structural integrity;
- the relative levels of safeguards assurance that these techniques can be expected to give the IAEA;
- the impact on assembly costs of applying seals for verification of the structural integrity;
- the impact of inspection on reactor designs and operations;
- the impact on the head-end (chopping stage) of the reprocessing plant resulting from a safeguards-influenced assembly design;
- and the impact of inspection on the operations of the current fuel handling systems.


REFERENCES


SAFEGUARDS APPROACHES FOR MULTI-UNIT CANDU POWER STATIONS*

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International Atomic Energy Agency, Vienna

Abstract

SAFEGUARDS APPROACHES FOR MULTI-UNIT CANDU POWER STATIONS.

The safeguards approach for the multi-unit Candu stations Bruce and Pickering is based on a safeguards scheme that, in addition to material accountancy, provides for independent physical verification by inspectors of fuel flow and inventories within the facility through the use of containment and surveillance systems, NDA measurements and bundle counting. The instrumented scheme comprises input and output bundle counters, yes/no monitors, convenience seals, film cameras, fuel verifiers and ultrasonic under-water seals. Until the instrumented schemes are fully implemented inter-bay fuel transfer verifications are carried out by inspectors frequently present on site.

1. INTRODUCTION

There are three multi-unit Candu reactor stations under IAEA safeguards, namely Bruce A with four reactor units in operation; Bruce B with four units under construction; and Pickering with four units in operation and four units under construction. The reactors of the two Bruce stations have each an electrical power of 750 MW(e), and the Pickering reactors have 520 MW(e) each. All three stations are situated in Ontario in Canada.

The Candu reactors are on-load fuelling natural uranium reactors. Each of the two Bruce stations have a common containment structure and common fuelling machines for all reactor units of the station; Pickering has an individual containment and individual fuelling machines for each reactor unit.

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* This paper was written in collaboration with D. Sinden, of the Atomic Energy Control Board, Ottawa, Canada.

1 CANada Deuterium Uranium.

2 Reactor unit No.5 at Pickering was loaded with fuel in July 1982.
2. FACILITY DESCRIPTION

2.1 Fuel

The fuel unit in Candu reactors is a bundle consisting of 37 or 28 Zircaloy-clad UO$_2$ elements. The length and diameter of a bundle is approx. 0.5 x 0.1 m and the natural uranium content is about 19 kg.

The fresh fuel is received and stored at the reactor stations in cardboard boxes, each containing 36 bundles. Each bundle is identified by a serial number stamped into the end plates. This cannot, however, be used for identification purposes in safeguards schemes because of the difficulty in reading the number on irradiated fuel that is stored under water.

2.2 Buildings and Structures

In the Bruce power station nuclear material is stored and handled in four interconnected reactor buildings, one service building and one ancillary building. Two fuelling machine systems pick up a supply of fresh fuel bundles from loading machines in the Service Building and travel in a duct that passes underneath all four reactors as shown in Fig. 1. The bundles are loaded into any of the four reactors and an equal amount of spent (irradiated) bundles is received by the fuelling machine and returned to the Service Building where the bundles are discharged via elevators into the Primary Storage Bay. After several months of storage the bundles are transferred via a duct into a Secondary Storage Bay as shown in Fig. 2. In both bays the bundles are placed on trays stacked on top of each other.

The Pickering power station consists of eight separate reactors, each with its own containment and pair of fuelling machines. The fresh fuel storage is common for all eight
reactors, while there is one primary and one secondary storage bay for units 1 to 4 and one primary and one secondary for units 5 to 8, as shown in Fig. 3. Fresh fuel boxes are transported from the common storage area into each reactor containment, where the bundles are received by the fuelling machines and loaded into the reactor. The 12 spent fuel bundles are discharged from the reactor core by the fuelling machines and transferred via elevators to the Primary Storage Bay where they are stored in baskets. After several years the baskets are moved to the Secondary Storage Bay in a flask that is carried on a surface vehicle.
FIG. 3. Pickering A fuel handling system.
3. Material Flow and Diversion Assumptions

The flow of nuclear material through the Bruce and Pickering stations is shown schematically in Fig. 4. The four areas within a multi-unit Candu station in which nuclear material is contained and the typical nominal inventories, expressed in significant quantities SQ, are the following:

<table>
<thead>
<tr>
<th>Area</th>
<th>SQ</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fresh Fuel Storage Area</td>
<td>up to 30</td>
</tr>
<tr>
<td>Reactor Cores</td>
<td>30 per unit</td>
</tr>
<tr>
<td>Primary Storage Bay</td>
<td>up to 300 (Bruce) and up to 900 (Pickering)</td>
</tr>
<tr>
<td>Secondary Storage Bay</td>
<td>up to 3000</td>
</tr>
</tbody>
</table>

For a definition of SQ see 'IAEA Safeguards: Glossary', IAEA/SG/INF/1, para.89.
1 SQ of fresh fuel is approx. 560 bundles; 1 SQ of spent fuel is approx. 110 bundles.
Table I. RECOGNIZED DIVERSION PATHS

<table>
<thead>
<tr>
<th>Material</th>
<th>One SQ (bdls)</th>
<th>Detection time (months)</th>
<th>Diversion path</th>
<th>Complexity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fresh fuel in storage area</td>
<td>560</td>
<td>12</td>
<td>From storage by crane/loader/vehicle After having been charged into fuelling machines</td>
<td>Low</td>
</tr>
<tr>
<td>Irr. fuel in reactor cores</td>
<td>110</td>
<td>3</td>
<td>Bdls. removed from fuelling machines via existing ports or into flask in fuelling machine main tenance area</td>
<td>Medium</td>
</tr>
<tr>
<td>Irr. fuel in storage bays</td>
<td>110</td>
<td>3</td>
<td>From storage bay by crane/flask/vehicle Reverse flow of spent bdls. into reactor containment</td>
<td>Low</td>
</tr>
</tbody>
</table>

The inspection goal is to detect the diversion of 1 SQ within a time frame consistent with the detection time assuming abrupt diversion, and within one year assuming protracted diversion. The detection time used for fresh fuel in the Candu safeguards schemes is one year and for irradiated fuel in the reactor cores and storage bays 3 months.

In the safeguards schemes the possibility is considered of diversion of fresh fuel from the Fresh Fuel Storage Area, and of irradiated bundles from the storage bays. Diversion from the reactor cores is also considered, assuming diversion

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4 For a definition of detection time and for abrupt and protracted diversion see 'IAEA Safeguards: Glossary', IAEA/SG/INF/1, paras. 90 and 8.
strategies other than those requiring major structural modifications of the reactor buildings. When priorities for implementation programmes are being assigned it is assumed that irradiated fuel is more likely to be the target of a diversion action than fresh fuel. Diversion of parts of a fuel bundle is not considered.

Table I shows the recognized diversion paths.

4. SAFEGUARDS SCHEMES

4.1 Design Principles

From a safeguards point of view the Candu reactor is characterized by

- on-load fuelling
- large inventory of fuel bundles
- access difficulties for routine fuel inventory verification.

The basis for the safeguards schemes contemplated for Candu reactors is - as in all IAEA safeguards approaches - the material accountancy and the examination by inspectors of facility records and State reports. In addition the schemes provide for independent physical verification by inspectors of fuel flow and inventories within the facility through the use of containment and surveillance, NDA measurements and bundle counting.

The bundle flow is directly verified by measurements at the input into the reactor cores, at the output from the cores, and at transfers from the Primary to the Secondary Bay. The inventories in reactor cores and storage bays are indirectly verified by the input/output measurements in combination with containment and surveillance measures. The basic safeguards scheme is shown in Fig. 5.

The bundle flow is measured by core input counters, by core discharge counters and by inspector's visual counting together with NDA measurements. The surveillance is carried out by film camera systems, and containment measures include yes/no monitors, convenience seals and, in storage bays, ultra-sonic under-water seals.

In addition, the fresh fuel in the storage area is verified at annual physical verifications by counting and NDA measurements. Whenever spent fuel is shipped from the facility, inspectors have access for verification and sealing of the transport flask.
FERRARIS and WREDBERG

Core Inv. Verif.: New Inventory = Previous Inventory - F_{IN} - F_{OUT}

Bay Inv. Verif.: New Inventory = Previous Inventory + F_{OUT} - F_{S}

F_{S} = Fuel Shipment

**Fig. 5.** Candu basic safeguards system.

**Table II. Primary and Secondary (Back-up) Safeguards Systems**

<table>
<thead>
<tr>
<th>Nuclear material area</th>
<th>Primary safeguards system</th>
<th>Secondary (back-up) safeguards system</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor core</td>
<td>1. Input counters</td>
<td>a) Examination of reactor data</td>
</tr>
<tr>
<td></td>
<td>2. Discharge counters</td>
<td>b) Random fuel channel verification</td>
</tr>
<tr>
<td></td>
<td>3. Yes/No monitors</td>
<td></td>
</tr>
<tr>
<td></td>
<td>4. Convenience seals</td>
<td></td>
</tr>
<tr>
<td>Storage bay</td>
<td>1. Film camera surveillance</td>
<td>1. Ultrasonic seals</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2. Fuel verifier</td>
</tr>
</tbody>
</table>

Annual Physical Inventory Verification by Inspectors

Containment including Yes/No Monitors and Convenience Seals

Sealed Fuel

Surveillance by Redundant Film Camera Systems
For independent verification of fuel flow or inventory in the reactor cores and the storage bays a secondary system for back-up verification is assigned to each primary safeguards system in case of system failures or significant anomalies.

Table II shows the relation between primary and secondary (back-up) safeguards systems.

For the film camera surveillance system redundant equipment is used, i.e. at each camera location a twin-camera unit is installed. Furthermore, the different camera locations in bay areas overview the same surveillance area. Also the ultrasonic under-water sealing system in the storage bays is redundant, i.e. there are two seals for each sealed storage module.

4.2 Bruce scheme (Figure 6)

The safeguards measures for the core inventories include:

Input counters, common for all four reactor units
Discharge counters, also common
Yes/no monitors in fuelling machine ports.

As back-up core inventory verification measures to resolve anomalies, the Agency is studying the operational impact and effectiveness of two methods. One is to read reactor instrument data, such as channel outlet temperature, channel flow and integrated thermal power. The other is - as an ultimate back-up measure - random channel defuelling.

Film camera surveillance is applied to both the Primary and the Secondary Storage Bays. The fuel trays in the Secondary Bay are stacked in storage modules that are sealed with ultrasonic seals. The seals are checked frequently using statistical sampling. An optical tool or attribute verifier will be used as a back-up inventory verification measure for fuel in the bays.

Until the discharge bundle counters are installed and until all spent fuel that has been stored up to that time in the Primary Bay is transferred to the Secondary Bay, inspectors will continue to be present during all transfers of fuel between the two bays. Inspectors' verification is done by counting of bundles and attribute verification using a gamma spectrometer at random.

Fresh fuel is verified annually. A new procedure involving opening of fuel boxes is being studied. The fresh
FIG. 6. Bruce safeguards system.
fuel in the storage areas of the multiunit Candu reactor facilities will be verified as part of the natural uranium fuel cycle in Canada according to a coordinated safeguards approach involving simultaneous inventory verifications at all facilities in the cycle.

4.3 Pickering Scheme (Fig. 7)

The Pickering safeguards scheme is conceptually the same as the Bruce scheme. The differences are related to the number of input and discharge counters and the storage and handling of the spent fuel. The fuel is stored in baskets stacked into cages (modules) in the Primary Bay for up to 4 years. The cages will, therefore, be sealed with ultrasonic seals both in the Primary and Secondary Bays. The transfer corridor between the bays will be under film camera surveillance. Until the instrumented scheme is fully implemented all transfers to the Secondary Bay are done with inspectors present for counting and attribute gamma verification.

5. INSPECTION ACTIVITIES

With fully implemented safeguards schemes in the Bruce and Pickering stations the inspection activities by the IAEA inspectors will comprise the following:

Servicing of film cameras once or twice a month depending on the type of films used, i.e. black and white or colour films;

Bi-monthly examinations of facility records and reading/servicing of input and discharge counters, as well as replacement of some of the yes/no monitors and checking of convenience seals;

Annual physical inventory verification of the fresh fuel inventory; checking of ultrasonic seals in the storage bays.

If no significant and unexplainable anomalies between operator refuelling records and counter readings are found at the bi-monthly inspections, and if the surveillance film reviews do not show any movements of undeclared fuel out of the storage bays, then the fuel stacked in the bays from the previous inspection will be sealed. The sealing is done by the operator in the presence of an inspector who will read and record the seal characteristics.
FIG. 7. Pickering safeguards scheme.
If, on the other hand, the inspectors will find significant anomalies that cannot be satisfactorily explained by errors or equipment failures, then back-up verification activities will have to be carried out as indicated in Table II. If the anomalies concern the fuel in the storage bays, the non-sealed fuel can be verified by the fuel verifier and the sealed fuel can be verified by checking the seals. The detailed process of back-up verification will be dependent on the surveillance system performance. When anomalies concern the core fuel the inspectors have the possibility of performing back-up verification by a detailed investigation of reactor operating data, and in extreme circumstances defuelling of randomly selected channels of the reactor core would be considered.

6. IMPLEMENTATION

6.1 At present

The safeguards instrumentation at the two multi-unit stations that are at present in operation comprises the following:

Bruce A:
- Film cameras in the Primary and Secondary Storage Bays;
- A gamma spectrometer in the Secondary Bay.

Pickering A, i.e. units 1 to 4:
- A gross gamma basket verifier in the Secondary Bay (a gamma spectrometer is ready to be installed);
- About 20 ultrasonic seals in the Secondary Bay;
- One discharge bundle counter for units 1 and 2.

In Pickering the first transfers of spent fuel bundles from the Primary to the Secondary Bay began in early 1978. Inspectors are present at each transfer of a flask containing 256 bundles stacked in baskets with 32 bundles each. All bundles are counted by the inspectors and some baskets are randomly verified using the gamma verifier and a statistical sample plan. The transfer campaigns occur normally during 10 days each month.

The film cameras in Pickering are serviced every month and the facility records are examined every second month when, in addition, the discharge bundle counter readings are checked.
with the operators' refuelling records. The bundle counter was installed in 1974 and has performed satisfactorily; the total discrepancy in the number of discharged bundles during the detection time period, namely 3 months, has never exceeded 10% of 1 SQ.

In Bruce A the first fuel transfers to the Secondary Bay started early 1979. As in Pickering all transfers are verified by inspectors who count the bundles that are laid on trays with 24 bundles each. Bundle trays are randomly verified with the gamma spectrometer that was installed in March 1981. There is normally one transfer campaign every second month lasting 4 weeks each.

The annual inspection effort is approximately 120 inspection-days for each station; 110 of these are required for the inter-bay fuel transfer verifications.

6.2 In progress

a) The "A" stations

The safeguards instruments to be installed in Bruce A and at the first four units of the Pickering station will be supplied to the Agency through the Canadian Support Programme. The supply of input and discharge counters and the ultrasonic sealing system has been established as tasks within the current support programme. The discharge counters are expected to be installed at Bruce in December 1983 and at Pickering during 1984. The first input counter is planned to be installed in Bruce A during the second half of 1982. For Pickering an input counter feasibility study is expected but no time schedule for such a study is yet established.

The ultrasonic sealing systems for the two stations are supposed to be similar to that of the Candu 600 instrumented scheme for which the first seals are expected to be attached during the second half of 1983.

Fuel verifiers of the Cerenkov type developed for the Candu 600 scheme will be available in the Primary Storage Bays in Bruce and Pickering. For verifying the fuel in the Secondary Bays the already available gamma spectrometers will be used.

Additional containment and surveillance in bay and transfer areas will be provided by the Agency's own twin camera Minolta system, as well as yes/no monitors and convenience seals where applicable.
b) The "B" Stations

The instruments for Bruce B and the 5th to 8th units of Pickering will be procured within the Agency's regular budget for safeguards equipment. Designs identical to the "A" stations will be used for discharge counter electronics and as far as possible also for counter collimator and shielding equipment. Furthermore, the ultrasonic seals will be of the same design in all the three stations.

There is no fixed time schedule established for the entire implementation of safeguards equipment in the B stations, other than for the bay surveillance systems which will be installed at the end of 1982 at Pickering B and in the spring of 1983 at Bruce B.

The verification by inspectors present at the inter-bay fuel transfers will continue until all fuel in the primary bays that has not been counted by discharge counters has been moved to the secondary bays. The secondary bay at Pickering B will not be constructed until several years after unit 5 has been put into operation. The fuel in the primary bay will, therefore, be sealed. At both Bruce and Pickering all spent fuel in both primary and secondary bays will be transferred to seismic-proof storage modules. These transfers are expected to start during 1983. The transfer operation will continue for several years.

7. DISCUSSION

The distinctive feature of the Candu reactors, namely the continuous material flow through the reactor system in an on-load refuelling scheme, presents a challenge to the safeguards inspector with respect to his task of independent verification of the material inventory. In addition, any safeguards approach for multi-unit stations must – in comparison with single unit facilities – take into account the numerical factor, i.e. the material flow scheme is common for four reactor units, and for eight units with respect to the fresh-fuel handling in Pickering. Furthermore, the inspector must consider the dimensional factor, namely the extensive size of these stations and the multiple accesses.

When considering the total inspection effort at Bruce and Pickering, the following should be observed. The inter-bay fuel transfer verifications at each station take up approximately 110 out of a total of 120 inspection-days per
year. This inspection effort is not directly comparable with the inspection of other reactor types but should be considered as part of the safeguarding of the fuel cycle. The inter-bay transfers at Bruce and Pickering correspond to a material flow that in other fuel cycles involves two facilities i.e. from one reactor facility to an intermediate storage facility. In the case of the Candu reactors this material flow takes place within one and the same facility and the inspection effort is, thus, limited to the reactor facility.

Having a fully implemented safeguards scheme at his disposal the inspector of the Bruce and Pickering stations will be able to carry out his verification task with a minimum of time spent on site and with a level of confidence that fully corresponds to the Agency's established standard of safeguards objectives. (A considerable part of the total inspection effort will, however, be assigned to service and review of surveillance equipment outside the facilities.) In the meantime, until a complete instrumented scheme is fully implemented, the inspection goals are met with inspectors frequently present on site for verification of material flow. A similar strategy in combination with camera surveillance may be considered as a supplementary measure in lieu of bundle input counters, the feasibility of which has not yet been demonstrated at Pickering. To enable resident inspectors to be used, a field office was set up in Toronto in September 1980.
SAFEGUARDS ASPECTS OF MULTINATIONAL FACILITIES

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Abstract

SAFEGUARDS ASPECTS OF MULTINATIONAL FACILITIES.

To take account of the many and varied economic, ecological and sociological requirements, particular mechanisms for cooperation must be applied so that a transfer of nuclear energy from the industrialized countries to the developing countries is facilitated. In this context aspects of non-proliferation and supply assurance attain a special significance.

A possible interaction with safeguards in a multinational facility should be analysed, particularly the way in which by implementing INFCIRC/153, §81 by a type of internal control on the basis of "interrelation with other states" inspection effort and expense could be reduced in such a facility and thus a multinational facility could be given a safeguards credit. Previous studies, particularly by the IAEA within the framework of the study on "Regional Nuclear Fuel Cycle Centres" as well as at the INFCE, have not taken this aspect of a safeguards credit into consideration but have rather regarded the multinationalization or internationalization of the fuel cycle as an additive measure. With respect to the uses of nuclear energy in industrially developing countries, a phased plan is suggested for the development of international cooperation. A typical feature is that each form of cooperation is subject to the national legislation of the host country in question. The results presented here cannot be regarded as final conclusions but rather as an initial approach by means of which possible interdependences between various countries and their interaction with the effectiveness of safeguards can be qualitatively represented.

1. Introduction

As the most recent development in the international energy situation has shown, the necessity of utilizing all available sources of energy all over the world has become unavoidable. An expanding use of nuclear energy in the industrialized countries and to an increasing extent also in the developing countries will thus also result. In order to take account of the many and varied economic, ecological and sociological requirements particular cooperation mechanisms must be applied in order to facilitate the transfer of nuclear energy to the developing countries. In this context aspects of non-proliferation and assurance of supply take on particular significance.
A large number of cooperation possibilities which would be sufficient for these two aspects have been discussed and worked out in the past. One of these possibilities which is already being put into practice at the moment and which in future will certainly be employed more frequently is multinational cooperation in one nuclear facility. In this context a possible interaction with safeguards in a multinational facility of this type should be analyzed, particularly the way in which inspection effort and expense could be reduced by implementing the INFCIRC/153, § 81, on the basis of "interrelation with other states" by a type of internal control in a facility of this sort and thus a safeguards credit could be given to a multinational facility. Furthermore, it is intended to investigate the degree to which the concentration of inspection effort in different countries could be balanced out. Previous studies, particularly by the IAEA as part of the study on "Regional Nuclear Fuel Cycle Centres" [1] as well as at the INFCE [2], have not taken this aspect of a safeguards credit into consideration, but have rather considered the multinationalization or internationalization of the fuel cycle as an additive measure.

In addition to the further technological development of safeguards which are exclusively devoted to increasing confidence and are not to be regarded as preventing a misuse of nuclear material, it also seems appropriate on the part of the IAEA to create conditions for reducing the danger of proliferation within the framework of foreign policy strategies in the various regions of the world. Institutional measures in general and the multinational cooperation discussed here can be seen in this context.

The results presented here cannot be regarded as final conclusions but rather as an initial attempt by means of which possible interdependences between various states and their interaction with the effectiveness of safeguards can be qualitatively illustrated.

2. Nuclear Energy Scenarios

Looking back now on about 30 years' worldwide development of the peaceful uses of nuclear energy the significance of this energy source in energy economics can be characterized as follows: in 1981 277 nuclear power stations with a net capacity of 159.2 gigawatt were in operation in 23 countries, 266 units with a net capacity of 211.7 gigawatt were under construction and orders were placed for a further 110 facilities with a capacity of 109.4 gigawatt [3]. Judging from these figures we can presume that in the mid-nineties an installed nuclear power station capacity of 480 gigawatt will be in operation in 33 countries. Results from a study at the Nuclear Research Centre Jülich will be mentioned.
here, representative of many strategy considerations in the field of the long-term development of nuclear energy. According to this an installed nuclear power station capacity of about 3970 gigawatt results for the year 2030, distributed among 74 countries. Fig. 1 illustrates the considerable expansion of nuclear energy up to the year 2030. The states with possible breeder utilization at this point in time are represented by speckled shading.

The Figure also shows very clearly the extent to which the IAEA safeguards would have to be expanded as the central non-proliferation instrument. However, rather than expanding technical measures of IAEA safeguards, institutional measures can be adopted in order to facilitate the application of safeguards to contribute to an increase in efficiency. This approach is particularly valid for large complex nuclear process facilities such as reprocessing and enrichment plants. The most recent international discussions on the development of safeguards in such facilities have indicated that particularly here exclusively technical solutions can only be implemented with difficulty.

If one returns to Fig. 1 then it can be seen that countries with breeder utilization could carry out meaningful back end services for the bordering states without breeder programmes as part of the supply policy. In which case this strategy must assume that countries with large nuclear programmes (about 30 gigawatt) will join in large-scale breeder utilization and reprocessing capacity must be made available for this in advance [4]. Similar models are conceivable for the front end of the nuclear fuel cycle. This type of front end and back end of the fuel cycle services should be carried out within the framework of suitable multinational cooperation, which should satisfy both the non-proliferation aspect as well as the supply assurance of the customers involved. The extent to which coupling supply assurance and non-proliferation can be achieved by e.g. multinational facilities will be discussed in detail in the next section.

3. Multinational Cooperation

For political and economic reasons there are already a series of examples of international cooperation in Europe. There are two examples in the enrichment sector: firstly the cooperation between three nations within the framework of URENCO-Centec and secondly between four nations in EURODIF. Moreover, there are reactors which are operated within the framework of EURATOM. Eurochemic, on the other hand, represents an international attempt at the back end of the fuel cycle in the same way as United Reprocessors, in which the United Kingdom, France and Germany are involved. There is also international cooperation within EURATOM in the research and development sector.
FIG. 1. Possible worldwide distribution of nuclear power plants in 2030.
There are also a number of examples of multinational cooperation outside the nuclear field, such as Intelsat or even Scandinavian Airlines System.

In addition to the aspect of supply assurance with enrichment or reprocessing services as well as the non-proliferation of nuclear weapons, there are still a series of further criteria which are important for the description and possible selection of this type of multinational cooperation. A list of such elements, which is certainly not exhaustive, would include:
- technology transfer,
- efficiency,
- political independence,
- vulnerability to sanctions,
- environmental aspects.

Moreover, possible effects of safeguards must also be taken into detailed consideration.

The importance and weighting of these individual criteria are difficult to describe. Thus for example environmental and acceptance questions with respect to the provision of front end and back end services involving foreign nuclear materials are of great significance for the host country of such a facility. Assurance of supply plays a particularly significant role for a foreign participating country. There are also different problems with respect to a possible technology transfer in the reprocessing and enrichment technologies. For this reason different points of view result for states which have the appropriate technology available and those who still want to gain possession of this technology. States with large nuclear programmes must fulfill different requirements from those with small nuclear programmes. It can thus be seen from these few aspects that the problems are complex and that detailed analyses must also be carried out from various standpoints before we can achieve a suggestion for a package relating to practicable multinational cooperation.

A spectrum of possible multinational cooperation [5] is now conceivable which would extend from purely national facilities up to facilities under international organizations with extraterritorial rights, as for example regional fuel cycle centres [1]. Between these two extremes one can find a series of models with increasing internationalization or multinationalization, as shown in Fig. 2. It seems first of all appropriate to distinguish between national and international models. In this context we mean by national not that the state is the proprietor of this facility but rather that the legal system according to which the facility is operated can be regarded as national. Models with financial participation, with international personnel, management
### FIG. 2. Assessment of institutional models.

<table>
<thead>
<tr>
<th>CRITERION</th>
<th>NATIONAL FORMS OF COOPERATION</th>
<th>INTERNATIONAL ORGANIZATION</th>
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<tbody>
<tr>
<td></td>
<td>NATIONAL FACILITIES WITH NPT-SAFEGUARDS AND</td>
<td></td>
</tr>
<tr>
<td></td>
<td>ONLY * FINANC. PARTIC. MANAGEMENT OPERATG. PERSONNEL</td>
<td>MULTINATIONAL UNDERTAKING + NPT-SAFEGUARDS WITHOUT WITH RENUNCIATION OF SOVEREIGNTY</td>
</tr>
<tr>
<td>Proliferation Hindrance</td>
<td></td>
<td></td>
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<tr>
<td>Political Independence</td>
<td></td>
<td></td>
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<tr>
<td>Assurance of Supply</td>
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<tr>
<td>Efficiency</td>
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<tr>
<td>Technology Transfer</td>
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<tr>
<td>Environmental Aspects</td>
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<tr>
<td>Acceptance</td>
<td></td>
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<tr>
<td>Vulnerability to Sanctions</td>
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</tbody>
</table>

- Comparable with reference model
- Advantages
- Great advantages
- Disadvantages
- Great disadvantages

NPT: Treaty on the Non-Proliferation of Nuclear Weapons
and multinational undertakings with various degrees of loss of sovereignty for the host state are possibilities in the national sector. In the field of international organization, models are listed with and without loss of sovereignty for the host state and extraterritorial rights.

Fig. 2 summarizes a qualitative analysis and shows that an increasing prevention of proliferation can also be identified with growing multinationalization or internationalization. These barriers result from additional obligations and contractual relations which the participating countries enter into with increasing internationalization. Parallel to this there is also a growing ability to impose sanctions which would result in the case of a breach of these contractual obligations. Nevertheless, it also appears from these analyses that models with international organizations or with extraterritorial rights are unacceptable. Problems of political independence and public acceptance of models which make such inroads into the sovereignty of a state are the reasons for this observation. According to the simple aspect of supply assurance it can be established that a purely national facility under IAEA safeguards and e.g. additionally under EURATOM safeguards is certainly the most favourable solution conceivable.

However, if one considers the problems discussed in the introduction that national facilities should also be used for the front end or back end of the fuel cycle when foreign nuclear material is involved then the following phased plan takes shape from the spectrum of possible models described

- national facility,
- national facility with provision of front end and back end of the fuel cycle services,
- national facility with financial participation,
- national facility with multinational management,
- national facility with multinational operating personnel,
- subsidiary.

In the stages "financial participation", "multinational management" and "multinational operating personnel" this phased plan also takes account of the growing technological needs of the developing countries and culminates in their own facility as a subsidiary once the appropriate size of the nuclear programme has been achieved.
4. Possible Safeguards Implications of Multinational Models

After a phased plan has been developed for a possible application of multinational models, particularly in order to promote nuclear energy in countries with fairly small nuclear energy programmes, we should then investigate in more detail the possible consequences which could thus result for international safeguards. The central question in this context is whether Art. 81 d, INFCIRC/153 can be implemented in a more qualified way by the application of such models and whether a safeguards credit for multinational facilities of this type thus results with respect to a reduction in inspection time and expense.

Multinational models are accompanied by three basic advantages for the non-proliferation of nuclear weapons:

- reduction in the number of sensitive facilities,
- contractual barriers against misuse (e.g. seizure),
- internal control by multinational cooperation.

The last sector is undoubtedly of particular significance for the possible implications of safeguards. This will be more closely investigated on the basis of the following considerations with respect to diversion problems.

A diversion of fissionable material refers to nuclear facilities in which the nuclear material is subject to international safeguards. In the same way as measures for its discovery, a diversion presupposes that all necessary information about the nuclear material present is available.

Among this is

- the description of the material according to amount, physical type and chemical composition,
- the description of the paths which the material takes through the facility,
- the description of the measuring methods by means of which the operator traces the material in his facility and documents it, as well as
- the description of the process steps to which the material is subjected.

In the final analysis diversion attempts result in withholding information about the material to which they are contractually entitled from the safeguards authority in some way or another. Several examples of opportunities for tampering can be listed:
- data are not made available,
- IAEA measurements are interfered with,
- tampering with the shipper/receiver inventory change reports for the same transaction,
- forging reports and records,
- preparation of incomplete, inaccurate and inconsistent records,
- subsequently tampering with the balance,
- use of dummy fuel elements,
- tampering with identification features.

The forms of cooperation listed in the phased plan are investigated in the Table with respect to the discovery possibilities which they provide in the case of a diversion.

If one considers the Table then in general a greater possibility of discovering anomalies can be seen with the increasing phases of the models. This results from the detailed knowledge and information connected with this cooperation with respect to
- facility operation and planning,
- nuclear materials management,
- financing and budgeting, as well as about
- possible anomalies in the facility.

More detailed quantifiable statements can only be gathered here after the contractual elements of the various multinational models have been established. The type and number of the participating states would also have to be taken into consideration as further criteria. The model with multinational operating personnel as such as well as in the subsidiary could provide the greatest possibility of discovering anomalies.

5. **Summary**

Current plans for expansion lead us to expect the application of nuclear energy in about 33 countries in the mid-nineties. In the longer term, the uses of nuclear energy will be distributed among more than 70 countries by the year 2030. While on the one hand we are dealing with the basic assurance of the nations' energy supply there is at the same time deep concern in all responsible governments with respect to the growing nuclear weapons potential of the nuclear weapons states as well as at the possibility of
### TABLE: Possibilities of Discovering an Anomaly with Various Cooperation Models

<table>
<thead>
<tr>
<th>Facility</th>
<th>Possibilities of Discovering an Anomaly</th>
</tr>
</thead>
<tbody>
<tr>
<td>own requirements only</td>
<td>- IAEA safeguards</td>
</tr>
<tr>
<td>with front end and/or back end services</td>
<td>- IAEA safeguards</td>
</tr>
<tr>
<td>with financial participation</td>
<td>- IAEA safeguards</td>
</tr>
<tr>
<td></td>
<td>- further information on facility operation</td>
</tr>
<tr>
<td>with multinational management</td>
<td>- IAEA safeguards</td>
</tr>
<tr>
<td></td>
<td>- determination of the facility can only be modified by the whole management</td>
</tr>
<tr>
<td>with multinational operating personnel</td>
<td>- IAEA safeguards</td>
</tr>
<tr>
<td></td>
<td>- actual operation of the facility is managed and monitored in detail by multinational personnel</td>
</tr>
<tr>
<td>subsidiary</td>
<td>- IAEA safeguards</td>
</tr>
<tr>
<td></td>
<td>- further discovery possibilities depending on the employment of multinational management or operating personnel</td>
</tr>
</tbody>
</table>
the emergence of further such states. Supply assurance and non-proliferation are therefore the basic motives for international cooperation in the nuclear sector. Cooperation increases confidence.

With respect to the uses of nuclear energy in industrially developing countries, a phased plan is suggested for the development of international cooperation. Presuming that there is a nuclear facility in a supplier state then the further stages are

- a facility which also keeps capacity available for other states (services)
- a facility in which other states are financially involved,
- a facility with multinational management,
- a facility with multinational operating personnel,
- a subsidiary abroad.

A typical feature is that each form of cooperation is subject to the legislation of the host country in question and is thus politically and socially acceptable. The basis for confidence towards foreign countries is considerably broadened by employing foreign personnel, whereby on the one hand the supply for the partner state is guarded more effectively against unforeseen encroachments on the part of the host state, a technology transfer is aided and possible misuse of the facility by the host state can be made known to the world at large in good time. In this way the international safeguards of nuclear material are effectively supplemented and corresponding to the aim of INFCIRC/153 the operation of nuclear facilities is not hindered or even made impossible by the compulsion for stricter and stricter methods of control on the part of the IAEA. Instead of this the possibility of reducing safeguards with the simultaneous promotion of the peaceful uses of nuclear energy for the well-being of mankind actually emerges through forms of multinational cooperation of the type described.

The ideas discussed here cannot yet represent a comprehensive approach to a solution but should rather provide a basis for further more detailed analyses particularly by the IAEA within the framework of evaluating the effectiveness of safeguards. Although the models presented here will only come into effect if nuclear energy is increasingly employed and can only comprise a small selection of the possible forms of multinational cooperation, we should not forget that there are already a large number of effectively operating models, such as EURATOM, URENCO and EUROdif, for which it would also be worthwhile to carry out analyses with respect to the extent to which a safeguards credit could already be applied now.
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STATE AND FACILITY MATERIAL CONTROL SYSTEMS

(Session 3)
Chairman

A. NILSSON
ASEA-ATOM's URANUS SYSTEM FOR PRODUCTION CONTROL, ECONOMIC CONTROL AND SAFEGUARDS

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Abstract

ASEA-ATOM's URANUS SYSTEM FOR PRODUCTION CONTROL, ECONOMIC CONTROL AND SAFEGUARDS.

ASEA-ATOM needs a system for production and economic control because: (1) the uranium is the valuable property of the customer; (2) short delivery times are a prime means of competition; (3) the manufacture of fuel necessitates frequent enrichment changes and much enrichment blending; (4) minimizing uranium stock reduces interest costs. A system which meets the above needs will with minor modifications meet safeguard requirements. URANUS is an integrated man/computer system with manual input and automatic data treatment and reporting of information. The man/computer interface is monitored by the automatic checking for plausibility of all input. An item is a quantity of material which is treated as a unit in production. Each item receives a unique identification number which may be used only once. If the qualitative properties of an item are changed a new item number is issued. Items have the following properties: (1) Type of material; (2) Lot No.; (3) Project No.; (4) Type of container; (5) Operation No.; (6) Enrichment(s), nominal and analysed; (7) MBA (external and internal); (8) Origin(s); (9) Quantity per origin; (10) Uranium content; (11) Impurities; (12) All changes of quantity; (13) References to previous and/or contributory items; (14) Notes. Items are reported on forms and fed into URANUS by authorized personnel using personal passwords or are directly reported by terminal from certain operations. Pertinent information is entered for each project. The URANUS computer system consists of terminal activated on-line and batch modules which are briefly described. Output is information for planning, economic control and safeguards. Safeguards information consists of the following. For flow and blending control: delivery information; change of nominal enrichment; enrichment blending; internal transactions that change an enrichment/origin account; material status report (MSR); item inventory per MBA; general ledger. For physical inventory: material balance report (MBR); book item inventory per MBA; physical item inventory per MBA; uranium balance per origin and enrichment.
1. ASEA-ATOM's FUEL FACTORY

ASEA-ATOM's fuel factory in Västerås, Sweden, manufactures BWR and PWR fuel from customer owned UF\textsubscript{6}. Customers are Swedish and foreign utilities that have procured yellow cake, conversion to UF\textsubscript{6} and enrichment in various countries. The fuel factory has a capacity of ca. 340 tonnes UO\textsubscript{2} per year.

2. WHY URANUS?

ASEA-ATOM needs a powerful and effective system for production control and economic control for the following reasons.

1. The uranium is the valuable property of the customer.
2. Short delivery times, that is short throughput times, are a prime means of competition.
3. In particular the manufacture of BWR fuel necessitates frequent enrichment changes and consequently much enrichment blending.
4. The minimizing of ASEA-ATOM's present and future stock of uranium, which is used to shorten scrap recycle times and thus throughput times, reduces interest costs.
5. Minimizing permanent losses to the environment and in permanently deposited waste and temporary losses to deposits in production equipment reduces costs.

It is ASEA-ATOM's position that a system which meets the above needs will with minor modifications meet quantitative safeguard requirements. However, the requirements of safeguards as regards accountability to countries of origin necessitate certain additional facilities.

3. WHAT IS URANUS?

URANUS is an integrated man/computer system with manual input and automatic data treatment and reporting of information. The man/computer interface is monitored by the automatic checking for plausibility of all input.

The Swedish Nuclear Power Inspectorate approved URANUS for safeguards on February 16, 1981.
4. INPUT

4.1. The item concept

The basis of the URANUS system is the item, which is defined as follows:

An item is a quantity of material which is treated as a unit in production, for example the material in a container or a fuel rod. An item is also a \( \text{UO}_2 \) powder/pellet lot in a pellet line.

Each item receives a unique identification number which may be used only once. This number together with references to previous and/or contributory items makes it possible to trace material backwards through production.

If the qualitative properties of an item are changed a new item number is issued. This does not apply to changes in quantity or location.

All containers which contain uranium are marked with item numbers. QC samples are marked with the number of the item from which the sample was taken.

4.2. Item properties

Item properties are the following:

1. Type of material. The various types of material have each their acceptable range of uranium content. Certain types of material are limited to certain internal MBA's.

2. Lot No. A lot is a quantity of powder with the same pelletizing characteristics and also the pellets made from a powder lot. A lot may consist of one or more items. A lot is also a quantity of fuel rods with the same manufacturing history. A lot belongs to a certain enrichment in a certain project.

3. Project No.

4. Type of container. Certain types of material are limited to certain types of container. The various types of container have different
maximum capacities. Certain types of container are limited to certain internal MBA's.

5. Operation No. Certain production operations are sub-divisions of certain internal MBA's.

6. Enrichment(s), nominal and analyzed. Certain enrichments belong to certain projects and rod types.

7. MBA (external and internal). External MBA's are for received but not yet checked material and for checked material. Internal MBA's are the uranium store, the conversion, pellet and assembly shops and the fuel bundle stores.

8. Origin(s). Certain origins belong to certain projects and enrichments.

9. Quantity per origin. At certain operations the quantity must be consistent with the changes of quantity and the referenced previous and contributory items.

10. Uranium content. This must be compatible with Type of material.

11. Impurities
12. All changes of quantity
13. References to previous and/or contributory items.

4.3. Checking item input

The terminal connection to the computer of all sources of item input such as scales, sampling, analytical laboratory instruments and material identification is not economically feasible. Manual input is always necessary and this introduces human errors which may be quite as large as sampling, weighing and analytical errors. In the URANUS system human errors are kept under control by the automatic cross checking for plausibility of all item input before admission to the data base. In this context plausibility cross-checking means that only permitted combinations of item properties (see 4.2) are admitted to the data base.
As a complement to these automatic checks the system produces as a feed-back material balances for the internal MBA's which are used for checking and control.

The entire system is periodically checked by the taking of physical inventories.

4.4. Item reports

Most items are reported manually on forms and fed into URANUS by authorized personnel using personal passwords. Items are reported on the following occasions:

1. All material passing external MBA boundaries.
2. All material passing internal MBA boundaries except QC samples and contaminated waste water and exhausted ventilation and pneumatic conveyor air.
3. Certain production operations.
4. QC rejects.
5. Changes of nominal enrichment.

Items are directly reported by terminal from:
1. Fuel rod loading.
2. Fuel bundle assembly.

4.5. Project input

Pertinent information is entered for each project, such as:
1. Project number and name.
2. Nominally required uranium quantity per enrichment and origin.
3. Quantity delivered by the customer per enrichment and origin.
4. Required number of fuel rods per rod type. A rod type is defined by:
   - Mechanical configuration and dimensions
   - Enrichment pattern
   - Burnable absorber pattern.
5. Required pattern of rod types in the bundle.
6. Required number of fuel bundles.

5. SOFT AND HARDWARE

The URANUS computer system consists of terminal activated online and batch modules of which a short description follows.

The online modules are designed to register data and to alter previously registered data. Online modules also display object information on a terminal screen and/or print it out on a terminal printer. An object can be a project, an item, an enrichment in an internal MBA, a production operation etc.

The modules for registration and alteration can only be activated by authorized persons using personal passwords whereas the informative modules can be activated by anybody.

Batch modules are those which require more CPU time and produce many pages of information. These are executed during night-time.

Each online module consists of two sub-modules, of which one describes pictures containing text and input fields and the other checks and treats the input. The picture description part uses ASEA's Basic Application System (BAS) as a sub-system. The two sub-modules which make up an online module are integrated to ASEA's ROSAM On-Line System (AROS). AROS handles terminals and transaction codes and carries out all tasks which are necessary for data base management.

The online modules can be activated from terminals by giving their transaction codes. When entering or altering information the transaction code may be followed by the appropriate value. AROS identifies the terminal and the transaction code and hands over control to BAS and to the appropriate online sub-module for the checking of the input. If every
thing is in order AROS calls the appropriate sub-module to update the database. If the new input is incorrect or implausible the operator is requested to correct the unacceptable values. The intensity of light in the input fields of the picture displayed on the screen indicates the incorrect value.

URANUS is designed for execution in an IBM 3033 computer. All of the modules are written in PL/1. At present the system includes four viewing screen terminals and two high speed printers.

6. OUTPUT

6.1. Planning information

Online:
1. Project information
2. Item information
3. Quantity of scrap per project and lot.
4. Project status in the assembly shop: Produced rods and bundles.
5. QC accepted rods per project, rod lot and operation.
6. Project status in the conversion and pellet shops: Produced quantity of $\text{UO}_2$ per powder/pellet lot per certain operations.
7. Project inventory status: Project requirement, customer delivery, production status, delivered to customer.

Batch (over night):
1. Produced quantity of $\text{UO}_2$ and number of rods and bundles per certain production operations: Weekly.
2. Inventory per external or internal MBA.
3. Inventory per project.
4. Item lists.

Special module (online):
Enrichment and impurity blending calculations.
6.2. Delivery information (online)

1. Verification of correct fuel bundle assembly.
2. Printout of protocol for each fuel bundle including
   - Fuel rod (item) numbers, UO₂ weights and positions.
   - Fuel rod enrichments, types and origins, including the case of several enrichments and origins in the same fuel rod.
   - Quantities of UO₂, uranium and uranium 235 per enrichment.
   - Total quantities of UO₂, uranium and uranium 235.

6.3. Economic information (batch: over night)

1. Finished project: Received and delivered quantities per enrichment are listed and totalled.
2. Finished project item list: Items still registered on a finished project are listed as an aid in deciding to which project they are to be transferred.
3. Insurance information: As a base for calculating insurance premiums a monthly list is issued giving quantity per enrichment and internal MBA.

6.4. Safeguard information

6.4.1. For flow and blending control

Online:
1. Delivery information.² See 6.2/2
2. Change of nominal enrichment²
3. Enrichment blending²
4. Internal transactions that change an enrichment/origin account²

Batch (over night):
2. Item inventory per MBA: As required.
3. General ledger: Print out before the monthly IAEA inspection.

² Printout sent to the Swedish Nuclear Power Inspectorate.
6.4.2. For physical inventory (batch: over night)
   2. Book item inventory per MBA.
   3. Physical item inventory per MBA.
   4. Uranium balance per origin and enrichment.

7. ORIGINS
   Material delivered by a customer for a certain project may have several origins. The following rules have been developed for the transfer of origins from incoming to outgoing material and MUF.
   1. Shipments of project material, that is fuel and analytical samples, shall have the same origin(s) as the material received for the project in question.
   2. All other outgoing material, that is emissions, deposited material and by-products, and MUF shall have the same distribution of origins as the material in SWA during the inventory period. Material in permanent storage is not used to calculate this distribution.

   These rules are implemented with the help of URANUS (see fig.1).

8. URANIUM INVENTORY SIMULATOR UFSIM
   During the manufacture of UO$_2$ fuel certain quantities of uranium leave the main uranium process stream. This uranium is in surplus rods and pellets, in QC rejected rods and pellets and in waste such as waste water, vacuum cleaner waste, ventilation filter waste etc. This material's various forms and impurity levels result in different recycle times until the uranium can be returned to the main uranium stream.

   To cover this temporary diversion of uranium more uranium must be fed into the main process stream than the customer has delivered for the project. The fuel factory must thus own a certain quantity of uranium from which this temporary shortage can be met. The use of UFSIM is to study how the necessary size of the factory's own stock of
uranium of different enrichments varies as a function of time for a given order stock and for given production schedules, in order to be able to decide periodically on the economically appropriate size and composition as regards enrichments of this stock.

Input to UFSIM is generated by URANUS.
EXPERIENCE WITH A COMPUTERIZED ACCOUNTANCY SYSTEM
AT A FABRICATION PLANT
FOR HIGHLY ENRICHED URANIUM

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Federal Republic of Germany

Abstract

EXPERIENCE WITH A COMPUTERIZED ACCOUNTANCY SYSTEM AT A FABRICATION PLANT FOR HIGHLY ENRICHED URANIUM.

In 1975 a computerized accountancy system for safeguards was installed at the NUKEM fabrication plant at Hanau, Federal Republic of Germany, and since then improvements have been made to it. The nuclear material is controlled on the basis of batch data during its flow through the different storage and fabrication areas. In a near-real-time accountancy manner all material movements are reported to the central nuclear material accountancy group of the plant who feed the data into the computer. Advantages are a quick overview and an immediate listing of total and itemized inventories of plant sections. The physical inventory taking can be performed four times as fast as without computer. For users no special computer knowledge is necessary; the system is easy to manage and delivers suitable data to the international inspectors, who have not requested a system change. But it is planned to improve the system by more terminals and additional software features.

1. INTRODUCTION

At the NUKEM fabrication plant at Hanau, Federal Republic of Germany, different kinds of nuclear materials are processed.

The highly enriched uranium is used to produce fuel elements for materials testing reactors (MTRs) and high-temperature reactors (HTRs). The main processing steps and areas are:

1. Chemical processing including scrap recovery
2. MTR processing lines (alloy and cermet)
3. HTR and THTR processing lines (kernel, particle and pebble)
4. Uranium oxide (pellet and rod)
5. New lines, e.g. uranium silicide.
The depleted uranium metal is melted down and machined to become shielding parts, etc. Depleted uranium is also used as simulating material for developing new processing steps and new products.

Natural and low enriched uranium and its compounds are received for re-treatment, processing and storage of up to large amounts.

Good material management is necessary for fulfilling plant requirements. To meet nuclear regulations and safeguards requirements a safeguards system has been implemented and improved in cooperation between the operating and the safeguarding authority [1]. To speed up the information transfer a computer system [2] is used to process safeguards data. It has been used on a routine basis since 1975 and provides an overview and a balance of all nuclear material at any time.

The tag inventory taking can be performed within 1 day with computer listing instead of 4 days without computer.

2. STRUCTURE OF ACCOUNTANCY SYSTEM

A data base structure has been developed based on the particular requirements for material management and safeguards.

A first structuring element is that plant areas are specified by odd and storage areas by even account numbers (only first digit). The second structuring (Fig. 1) element is the separation into five material balance areas (MBAs), using separate account numbers.

The next structuring element is the creation and assignment of account numbers for the kinds of material in the storage areas and for responsibilities and processing steps in the plant areas (second and third digits).

Another structuring element is the customer's name or the order name.

The physical inventory listing (PIL) of the storage areas enables the batches to be identified, and the PIL of the plant areas shows summarized inventories according to the different orders. The order identification is connected to the specific customer.

The exact locations of items in the storage areas are shown by means of the storage number, the box number, the account and the batch number.
FIG. 1. Material balance areas.

KBMP1 : receipt or shipment
KBMP2 : product shipping
KBMP3 : loss and waste
KBMP* : shipper / receiver difference

FIG. 2. Example of physical inventory listing.
<table>
<thead>
<tr>
<th>Lager-Position</th>
<th>Konto-Nr.</th>
<th>Konto-Nr.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Von:</td>
<td>An:</td>
<td></td>
</tr>
<tr>
<td>Von:</td>
<td>An:</td>
<td></td>
</tr>
<tr>
<td>Auftrags.-Bez.:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Material:</td>
<td></td>
<td></td>
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<tr>
<td>Stück:</td>
<td></td>
<td></td>
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<tr>
<td>Menge, g:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>U-235, %:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Uran, g:</td>
<td></td>
<td></td>
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<td>U-235, g:</td>
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<td></td>
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<td>Thorium, %:</td>
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<td></td>
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<td>Thorium, g:</td>
<td></td>
<td></td>
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<td>Brutto, g:</td>
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</tr>
<tr>
<td>Tara, g:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Netto, g:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Datum</td>
<td>Unterschrift</td>
<td></td>
</tr>
</tbody>
</table>

Kommentar (max. 38 Stellen) | Bei Angabe von
U %, U-Anal.-Nr. und Isotopen-Anal.-Nr. bis 20 Stellen weniger.

Isotopen-Anal.-Nr.

Zus. Bemerkungen =
(wird nicht in EDV gespeichert):

*= Messungs-Code

FIG. 3. *Batch data card.*
FIG. 4. Safeguards software structure.
For the plant areas the exact physical locations of the material are not recorded, but only the amounts of material in each processing step/area.

In summary, each item is identified by its key measurement point (KMP), the batch number, the account number, etc. (Fig. 2).

3. INPUT DATA FLOW

A material card is attached to each material item in the storage and plant, showing the item data, e.g. kind of material, composition, weight, enrichment (Fig. 3).

When the item is transferred to another location the data of the material card and any necessary additional data (e.g. source, destination) are fed into the computer.

The input phase is performed in a dialogue manner. The form of the data, the plausibility of account number, permission for transport, etc. are checked. The batch data files are updated. The transfer data of material are recorded and stored on magnetic tape.

4. SOFTWARE STRUCTURE

In simplified form the program system and data files are shown in Fig. 4. There are data files for accounts, directories and system listings. Each account number is connected to a single account file. Each account file contains its account identification, the batch identification and the batch data (Fig. 5).

The directory files are used to facilitate and accelerate the access to the data sets.

The system listing files contain all account numbers and orders and enable plausibility to be checked.

There are two programs mainly used to handle the data, namely BUCH and PROT. The program BUCH (control program for different sub-routines) executes the dialogue for booking according to the kind of transport (e.g. from storage to plant or from plant to plant area). The sub-routine PRUEF checks the plausibility, e.g. it checks to see whether the orders and account numbers are defined and whether the material transfer is permitted; as an example, a direct material transfer from a plant area to the shipping area is not allowed.
### FIG. 5. Data file structure.

<table>
<thead>
<tr>
<th>WORD-NR.</th>
<th>CONTENT</th>
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<tr>
<td>3</td>
<td>Date</td>
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<tr>
<td>4</td>
<td>Measurement Code</td>
</tr>
<tr>
<td>6</td>
<td>Storage Location</td>
</tr>
<tr>
<td>7</td>
<td>Order-Number</td>
</tr>
<tr>
<td>10</td>
<td>Batch ID Number</td>
</tr>
<tr>
<td>11</td>
<td>Number of Pieces</td>
</tr>
<tr>
<td></td>
<td>Height and Fractional Weight of Isotopes</td>
</tr>
<tr>
<td>25</td>
<td>Order Name</td>
</tr>
<tr>
<td>33</td>
<td>Material Description</td>
</tr>
<tr>
<td>41</td>
<td>Comment</td>
</tr>
<tr>
<td>58</td>
<td>EURATOM Codes</td>
</tr>
<tr>
<td>59</td>
<td>EURATOM Codes</td>
</tr>
<tr>
<td>64</td>
<td>Extension</td>
</tr>
</tbody>
</table>
With the sub-routine WOSPE all booked data are kept in a disc file for later documentation on magnetic tape. The program PROT is used to print the PIL and the listings arranged according to account number or order name or storage location.

There are also support programs, e.g. to update and extinguish orders or to correct and search for errors/failures.

The permanent inspectors can obtain different paper listings and the PIL on magtape, which is used in the Euratom computer for further evaluation.

The programs are written in Fortran IV. Apart from those mentioned above, there are also computer system programs, e.g. real-time/timesharing executive system; file manager; Fortran-compiler; Editor.
5. HARDWARE CONFIGURATION (Fig. 6)

The material accounting and safeguarding system is installed on a Hewlett Packard 1000 F computer, which is also used for other technical-scientific calculations and for calculating the isotopic composition, reading the paper tape punched by a mass spectrometer.

The periphery of the computer includes: 50 MB discs; 5 MB cartridges; 2 magtapes; 2 display terminals; 1 teletype; 1 card reader; 1 paper tape reader.

The hardware, which is regularly maintained, is very reliable and has a high availability.

It is ensured that only members of the accountancy group have access to the accountancy programs at any time.

6. FUTURE EXTENSION

Although the safeguards authority have not requested any improvement in the computer systems it is necessary to speed up the booking of material transfer and to eliminate manual work in order to improve the near-real-time character of the system. The input of material transport data will not be performed by the members of the accountancy group any more but instead will be carried out directly by the plant personnel at the entrance/exit of the plant by means of data capture terminals. Time will be saved and errors will be minimized by this direct access and additional plausibility check programs. The members of the accountancy group will have more time for controlling and for their specialist safeguards tasks.

It is also planned that the inspectors will have direct access to the stored data via a terminal. The inventory change reports and the material balance reports which have been compiled manually up to now will be printed by the computer.

REFERENCES

EXPERIENCE IN SSAC TRAINING AND TECHNOLOGY TRANSFER*

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Abstract

EXPERIENCE IN SSAC TRAINING AND TECHNOLOGY TRANSFER.

Each year since 1980 an international training course on the implementation of States' Systems of Accounting for and Control of Nuclear Materials (SSACs) has been offered in the USA under the auspices of the Nuclear Non-Proliferation Act of 1978. The courses are sponsored by the US Department of Energy in cooperation with the International Atomic Energy Agency (IAEA) and conducted by the Los Alamos National Laboratory, with the assistance of other organizations. The purpose of the courses, which are described in detail in the paper, is "to provide practical training in the design, implementation and operation of a national system of accounting for and control of nuclear materials that satisfies both national and IAEA international safeguards objectives." On odd numbered years, the course emphasis is on bulk-processing facilities; on even years the emphasis is on item-dominant facilities. Internationally known authorities are selected as course lecturers from the IAEA and Los Alamos, as well as government, industry, and national laboratories in both the USA and abroad. Lectures are supplemented with workshops, panel discussions, tours of Los Alamos safeguards laboratories, and visits to operating nuclear fuel-cycle facilities. Attendance at the 1981, 1982 and 1983 SSAC courses has averaged 26 student participants, with almost as many nations represented. The overall response to the courses has been highly favourable, indicating that they are fulfilling a timely and important need. Key to the success has been a course format that encourages maximum exchange of safeguards technology and experience among all participants, both students and lecturers.

* This paper borrows from a lecture by G.R. Keepin presented at the International Safeguards Training Course on Basic State Systems of Accounting for and Control of Nuclear Materials, held in Yalta, USSR, October 1981.
† At present on leave of absence from Los Alamos National Laboratory.
I. INTRODUCTION

The effectiveness of the overall international safeguards regime is clearly dependent, in large measure, on the effectiveness of the States' Systems of Accounting for and Control of Nuclear Materials (SSACs), whose performance the international system must independently verify. Thus, successful implementation of effective international safeguards requires, inter alia, an effective program of training, professional development, and technology transfer in two important areas. The first is the training of inspectors for the International Atomic Energy Agency (IAEA), and the second is the training and development of expertise among those personnel in Member States who are responsible for the State's safeguards system (and in turn its component facility safeguards systems) and the interface between the State and the IAEA. These complementary areas of safeguards training and technology transfer represent a continuing challenge both for the IAEA and its Member States - whether they are developing nations with rudimentary safeguards systems or nations with expanding nuclear programmes who may wish to expand and upgrade the technical capability and overall effectiveness of their SSAC and its interaction with both facility operators and the IAEA.

Toward these important goals, the IAEA has for many years conducted training activities, both formal and informal, for its safeguards inspectors. Considerable progress has been made in this area in recent years, with the establishment of an IAEA Safeguards Training Unit that reports directly to the Deputy Director General for Safeguards, and culminating in the present structured series of training courses that start with the Introductory Course on Agency Safeguards and are supplemented by a group of follow-on advanced and specialized training courses, some offered by the IAEA and others offered by Member States.

In recent years, increasing attention has been given to the important complementary area of training and assisting Member States in establishing, operating, and upgrading their own State systems. As all States under IAEA safeguards are obligated to provide the IAEA with accounting records and reports for all their nuclear material subject to safeguards, it has become increasingly important, and indeed a practical necessity, to achieve some degree of standardization and uniformity in accounting procedures, information treatment, and the methods and format of reporting to the IAEA. This need, as well as the special importance of general SSAC training to the overall effectiveness of IAEA safeguards, has long been recognized and repeatedly pointed out by the IAEA, both
informally and in formal IAEA documentation. SSAC training has also been the subject of special panels and advisory group meetings that ultimately led to the preparation of the very useful document IAEA/SG/INF/2, "IAEA Safeguards: Guidelines for States' Systems of Accounting for and Control of Nuclear Materials" issued by the IAEA in December 1980.

Similarly, the long-recognized need for uniform upgrading of SSACs led to the series of IAEA Basic SSAC Training Courses that were held in Vienna in 1976 and 1977 and in the USSR in 1978 and 1981. Follow-on courses dealing with SSAC implementation were conducted in the US in 1979, 1980, 1981 and 1982. The overall objective of these courses is to assist Member States - both developing and developed - in the establishment and operation of an effective SSAC that serves the safeguards needs of the individual nation and at the same time is responsive to its formal obligations to other States and external organizations.

The purpose of this paper is to review and evaluate recent experience with SSAC courses, in particular those courses that were held in 1980, 1981 and 1982 in the USA under the auspices of the US Nuclear Non-Proliferation Act (NNPA 1978). These courses have been sponsored by the US Department of Energy (US DOE) in cooperation with the IAEA and presented by the Los Alamos National Laboratory, with the assistance of other organizations, as discussed below. Before turning our attention to these SSAC courses, it should be noted that Sandia National Laboratories have conducted a series of companion NNPA courses on the related subject of physical protection, again under a cooperative arrangement between the IAEA and the US DOE.

II. INTERNATIONAL TRAINING COURSE ON MATERIALS ACCOUNTABILITY AND CONTROL FOR SAFEGUARDS PURPOSES, 27 MAY - 6 JUNE 1980; SANTA FE/LOS ALAMOS, NEW MEXICO, USA

The purpose of this 2-week training course, the Proceedings of which were published as Los Alamos National Laboratory report LA-8260-C, was "to provide practical training in the design, implementation and operation of a national system of nuclear material accountability and control that satisfies both national and IAEA international safeguards objectives." There were 26 attendees from 23 nations, namely Brazil, Canada, Chile, Denmark, Egypt, the Federal Republic of Germany, the German Democratic Republic, Greece, Hungary, India, Indonesia, Ireland, Israel, Italy, Japan, Kenya, Korea, Pakistan, the Philippines, Portugal, Taiwan, Turkey and the United States of America. Participants also came from the EURATOM organization of the Commission of the European Communities in Luxembourg. The course emphasized safeguards
requirements, necessary resources, the latest technological developments, and implementation as applied to power and research reactors and associated spent-fuel storage facilities.

The first week of the course covered the general principles and practice of safeguards - its evolution, basic elements, and current application of material accountability and control, inspection and verification on the national and international levels, as well as current practice in specific types of nuclear facilities.

The second week of the course covered the instrumentation and technology required to implement modern safeguards systems. The lecture material was correlated with, and supported by, tours and demonstrations (at the Los Alamos Safeguards R&D Laboratories) of up-to-date instrumentation and equipment. Detailed descriptions were given of current safeguards practice and actual operating experience in existing power reactor and research reactor facilities.

The second week of the course culminated in the "product" of the course, that is, the workshop on facility safeguards systems design. For the workshop, participants were divided into two groups, each of which independently considered the design of an SSAC for a postulated reference nuclear power programme. The reference power plant was a 1000-MW(e) pressurized water reactor (PWR), located in a non-nuclear weapons State. Both groups were assigned the problem of developing a safeguards approach and an SSAC for the reference facility using the facility design information provided.

The course concluded with individual design subgroup reports and an evaluation of the workshop results, as well as a detailed overall evaluation of the entire course. Feedback from the 1980 course was received in the form of direct comments from both attendees and lecturers, as well as detailed course evaluation forms that were completed by all course attendees. Suggestions for future course curricula included more emphasis on the IAEA/State system interface, more on NDA technology, more time for the Systems Design Workshop, and expanded coverage of fuel-cycle facilities safeguards and accounting/auditing methods. In general, there was a clear preference for the practical "how to" sessions rather than more theoretical topics. The valuable information contained in the completed course evaluation forms (as well as other input and comments on the 1980 NNPA course) was factored into planning and curriculum development for subsequent SSAC courses.
Major emphasis in this advanced SSAC course, the Proceedings of which were published as Los Alamos National Laboratory report LA-8901-C, was on the principles and practical methods used in establishing and operating nuclear material accounting and control systems at bulk handling facilities, particularly low-enriched uranium (LEU) conversion and fuel-fabrication plants. The course was conducted by the Los Alamos National Laboratory, the Battelle Pacific Northwest Laboratory, and the Exxon Nuclear Company, Inc. A total of 27 attendees took part in the 16-day course, including representatives from Belgium, Brazil, Canada, Egypt, France, India, Iraq, Israel, Italy, Japan, Korea, Luxembourg, Malaysia, Mexico, Pakistan, Poland, Sweden, Switzerland, Taiwan, Turkey and Yugoslavia. The course lecture staff included safeguards experts from the IAEA, US DOE, US Nuclear Regulatory Commission (US NRC), US Department of State (US DOS), Los Alamos National Laboratory, Battelle Pacific Northwest Laboratory, Exxon Nuclear Company, Inc., and Allied General Nuclear Services (AGNS).

The schedule of sessions and lecturers/instructors is shown in Tables I-A and I-B. After an overview of historical development and current trends in nuclear safeguards, the broad subject of "State Safeguards Systems and the International Interface" was addressed in two lectures on IAEA safeguards and two lectures on national safeguards systems. This was followed by a spirited panel discussion on the "IAEA/State System Interface," which addressed a wide range of safeguards topics and helped to put in perspective the principles and actual practice of national safeguards systems on the one hand and the overlay of the IAEA international system on the other. The panel, moderated by John Boright of the US DOS, provided a valuable exchange of practical experience, concerns and problems, particularly those faced by countries just starting, soon to start, or already in the early stages of implementing State systems of accounting and control. Among several recommendations coming out of the panel was a strong endorsement of SSAC training, and the suggestion that the IAEA seek to find ways to assist in (1) getting State system people together for further exchange of experience and problems in the practical implementation of SSACs, and (2) convincing State governments of the importance of domestic, as well as international, safeguards.

The second major topic, "Safeguards Measurement Technology and Applications," included a review of "traditional"
measurement methods (chemistry, mass spectrometry, bulk measurements) and a survey of the newer techniques of non-destructive assay (NDA) and their applications in fuel-cycle facilities. This was followed by a visit to the nuclear safeguards R&D facilities at Los Alamos for a tour and demonstration of NDA instruments and methods. A full day was then devoted to the principles and practice of safeguards systems design and implementation. Modern near-real-time accountancy/process control systems were described and specific examples given of operating in-plant systems at the AGNS Barnwell reprocessing facility and at the Los Alamos Plutonium Facility.
The second week of the course opened with a brief description and general tour of the Exxon Nuclear Fuel-Fabrication Plant. The plant accounting system, key measurement points, safeguards criteria, and operating system characteristics were then described and discussed. Lectures were alternated with numerous tours and demonstrations of plant measurement equipment - both conventional chemical analysis and NDA methods. Measurement control, sampling plans, and statistical data analysis methods used at the plant were described, as were typical NRC inspection procedures for LEU fuel-fabrication plants.
The course culminated in the workshop on safeguards systems design for a fuel-fabrication plant. To permit maximum participation, students were divided into four groups. Using data provided for a reference LEU fuel-conversion/fabrication plant, each group independently developed an outline of the key features they would recommend for the material accounting and control system for the reference facility.

In the final session, attendees completed a detailed Course Evaluation Form designed to provide feedback on the effectiveness and value of the overall course and to solicit suggestions for improving follow-on courses. Some of the main suggestions for future courses were to include: (1) more coverage of other (more typical) State systems and their experience in implementation, including interaction with the IAEA; (2) more input from the IAEA on their requirements and experience with various Member State systems; (3) more information on other types of facilities such as power and research reactors and critical facilities, as well as relevant up-to-date technology for assay/verification of fresh and spent fuels, statistics, and near-real-time accounting approaches. Participants also suggested allocating more time for individual study and follow-up discussion with lecturers and course staff, more time devoted to demonstration and hands-on experience in the laboratory with new measurement instrumentation, and more time for facility tours and demonstrations.

IV. INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS, 15-30 MARCH 1982, SANTA FE/LOS ALAMOS, NEW MEXICO, AND PALO VERDE NUCLEAR GENERATING STATION NEAR PHOENIX, ARIZONA

The current plan for SSAC courses offered in the USA is to emphasize bulk-handling facilities on odd-numbered years and item-dominant facilities on even years. Thus, the 1982 SSAC course was similar to the 1980 course, with a number of improvements based on direct comments from attendees, lecturers, and advisors, as well as the course evaluation forms from the 1980 and 1981 courses. Table II presents the course outline showing session titles and names of lecturers, panel chairman, tour coordinators, and workshop leaders.

Nations represented at the 1982 course included Brazil, Canada, Czechoslovakia, Finland, France, the German Democratic Republic, Iraq, Japan, Malaysia, Pakistan, the Philippines, Republic of South Africa, Sweden, Taiwan, Turkey and Yugoslavia. One interesting addition to the 1982 course was Session 7 in which one participant from each country gave a short informal talk on nuclear activities and safeguards within
<table>
<thead>
<tr>
<th>Session</th>
<th>Topic</th>
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<tr>
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<td>Welcome and Orientation</td>
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<td>2</td>
<td>Introduction to SSAC Training Course</td>
</tr>
<tr>
<td>3</td>
<td>Overview of IAEA Guidelines for State Systems of Accounting and Control</td>
</tr>
<tr>
<td>4</td>
<td>State System Experience with Safeguarding Power Reactors</td>
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<td>5</td>
<td>IAEA Safeguards at Reactor Facilities</td>
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<td>6</td>
<td>State System Experience with Safeguarding Research Reactors</td>
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<td>7</td>
<td>Workshop Seminar on IAEA-State Systems Interface</td>
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<td>Elements of Nondestructive Assay Technology</td>
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<tr>
<td>11</td>
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his or her country. Session 7 Chairman, Hans Grümm, IAEA Deputy Director General for Safeguards, summarized the main issues raised by participants as follows: (1) legal arrangements for SSACs, (2) voluntary offers by nuclear weapons States, (3) liaison committees between the IAEA and large SSACs, (4) quality and loyalty of IAEA inspectors, (5) computerization of accounting and reporting, (6) detection of dummy fuel elements, (7) reporting and batch matching of international transfers, (8) quality assurance of safeguards measurement systems and containment and surveillance systems, (9) flexibility of key measurement points at research reactor facilities, and (10) updating of design information questionnaires.

Reflecting the course emphasis on power and research reactors, lecturers from six different countries addressed material accountancy and control procedures for these types of facilities at both the State and facility levels. By spending two days visiting Los Alamos safeguards laboratories, rather than the one day offered in previous courses, participants were able to get more hands-on experience with a wide variety of NDA equipment and to perform measurements on both fresh and spent fuel.

After the 1-1/2-day workshop, in which each of four groups was given the task of designing a safeguards system for a "reference" 1000-MW(e) PWR facility, participants visited the Palo Verde Nuclear Generating Station (PVNGS) near Phoenix, Arizona. Operated by Arizona Public Service Company, PVNGS consists of three 1300-MW(e) reactors, with one unit essentially complete and scheduled to go on-line in May 1983.

At each of the SSAC courses, there has been an awards ceremony in which participants are presented with a certificate on having successfully completed the course. In 1982, students had the unique honour of hearing an address by and receiving their diplomas from Sigvard Eklund, Director General of the IAEA from 1961 to 1981.

V. SUMMARY AND CONCLUSIONS

The highly favourable response to the SSAC courses offered in the USA indicates that the courses are fulfilling a timely and important need. The IAEA and nuclear supplier nations can contribute in a significant way toward improving international safeguards by assisting with the training of safeguards personnel at the State and facility levels throughout the international community.

As these courses have evolved, it has been found that the most critical element in their success is the wide exchange of information, technology, and experience between all
participants, both students and lecturers. Many of the people attending hold positions of major responsibility in research, operations, and technical management in nuclear material accounting and control organizations in their respective countries. The courses can be structured to take advantage of this wealth of experience, while overcoming some of the problems in communicating across political, organizational and cultural barriers. For example:

- It is advantageous to conduct the courses in an informal setting where people can interact freely, both during and after the scheduled sessions.
- Open dialogue can be encouraged through workshops, panels, student presentations, and question and answer sessions.
- It is highly desirable that both lecturers and students represent a balance of safeguards backgrounds and interests - such as advanced and developing countries, nuclear supplier and recipient States, facility operators and government organizations.
- Technical content should stress practical approaches that people can apply to their own nation's safeguards problems, including experience implementing SSACs and interfacing with the IAEA.
- Lectures should be interspersed with tours of nuclear facilities, demonstrations of operating equipment, and discussions with local development and operations personnel.

In conclusion, the major benefit of SSAC courses to date has been the transfer of safeguards technology and experience among Member States and the IAEA. States are generally in agreement that cooperation with the IAEA leads to more effective national and international safeguards and that today's safeguards problems can be resolved through the application of technology, training, and improved communication between the parties involved.
RAPID INVENTORY TAKING
BY ELECTRONIC DATA GATHERING

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Abstract

RAPID INVENTORY TAKING BY ELECTRONIC DATA GATHERING.

A Plessey bar-code reading system has been installed in a nuclear material store at AEE Winfrith and is used to identify discrete packages of various nuclear materials. A minicomputer is utilized as a data store which, coupled with the Plessey bar-code system, produces physical inventory listings or inventory discrepancy reports very quickly. The time taken for a complete inventory of the store has been reduced from 60 man-days to 5-man days.

1. INTRODUCTION

The production of a physical inventory listing (PIL) for a store with a large number of items is both time-consuming and labour-intensive when performed using manual transcription of data. There is, therefore, a strong incentive for automated data processing, coupled with data acquisition equipment, to improve the inventory taking exercise.

This paper describes a rapid inventory-taking system, incorporating bar-code labels and a portable light pen reading unit, being used in a nuclear material store at AEE Winfrith.

2. PREVIOUS INVENTORY TAKING AND RECORD SYSTEM

A detailed accounting and record system is essential for efficient operation of a nuclear material store (NMS) in order to avoid potential criticality situations and to produce obligatory returns for the safeguards inspection authorities. The original system relied exclusively on manual accounting using ledgers, but this was slow and required considerable manual effort. PILs were also carried out manually by checking container labels against location and entering details on check sheets by hand. These check sheets
were then processed manually, checked for errors, and the PIL was finally produced in a hand-written form. Each six-monthly PIL for the NMS involved twelve man-weeks effort, and resulted in a radiation dose to the operators which, although within acceptable limits, could be improved upon by a reduction in the time required for PIL taking.

3. THE NEW SYSTEM

3.1 Computer System

A DEC PDP 11/23 computer with twin RL01 5M-byte disk units, a VT100 visual display unit and an LA120 Decwriter were selected for the data processing system which would hold all the data relevant to the nuclear materials currently stored in the NMS. A software package was written in Fortran under a real time, multi-task, multi-user operating system for the day-to-day running of the store, including routines to produce all the obligatory returns to the safeguards inspection authorities.

3.2 Light Pen Units

Operational trials were set up at Winfrith in which two label reading systems were tested; an optical character recognition system and the Plessey bar-code/light pen system. The trials consisted of reading labels attached to various sizes and shapes of packages and noting the number of attempts required to successfully read the label. After the trials a recommendation was made that the bar-code system would be more suitable for data collection in a nuclear material store where the packages were of different shapes and sizes.

The Plessey bar-code label comprises a series of vertical printed bars with intermediate spaces. Each bar with its following space occupies a similar width; a thick bar followed by a narrow space represents binary "1" and a thin bar followed by a wide space represents binary "0". Successive groups of four bars and spaces are used as binary-coded decimal equivalents of numbers and certain letters or commands. The data contained on the bar code used in the NMS application is limited to a starting key measurement point (KMP) identifier, a package serial number and certain command codes. The data characters are preceded by a start code and are followed
by a check digit and a terminate code; these are used to set up the associated data capture unit so that spurious signals are not interpreted as data characters.

The data capture unit consists of a light pen incorporating a light source, a light-sensitive detector and the necessary pre-amplifier and cable terminations. A fibre-optic cluster directs light from the source through the pen tip on to the surface of the bar-code label. As the pen tip is swept across the bars of the label a proportion of light is reflected back to the detector circuits in the pen and converted to electrical signals which are then encoded into ASCII character codes. These codes are then stored in one of three storage methods, viz:

i The Plessey 1450 Portable Data Capture Unit (PDCU) which stores the data on magnetic tape.

ii The Plessey 1488 PDCU which stores the data in solid-state memory.

iii The Plessey 1115 Fixed Unit which transmits the data directly into the computer.

4. DATA

Data relating to nuclear material, the containers in which the material is stored and the movements of these containers (and materials) are held against unique reference numbers in the computer data file. These data are entered into the computer file against the package reference number as a separate operation when the package is received. (Appendix 1 summarizes the data held for each package). A reference number is allocated to each package and the bar-code label with that reference number is attached to the container. The labels are self-adhesive and are protected by a self-adhesive transparent plastic tape. Investigations are being made into the availability and value of tamper-proof labels.

Many storage safes have been checked and a seal put on them by the Euratom Inspectorate. Hence, to allow the new inventory-taking procedure to function without breaking the seals, a bar-code label is attached to the outside of the safe. The reference number encoded in this label refers to a list of the bar code labels attached to containers stored inside the safe.
5. THE NEW PROCEDURES

5.1 Inventory Taking

At present the Plessey 1450 PDCU is used to read and record data during an inventory-taking operation. The unit is equipped with a keyboard for manual input of data and a light pen for bar-code reading. Data storage is on tape cassettes.

Operation of the system is very simple. The operator starts by loading a blank tape cassette, switching the unit on in the READ Mode and typing in a start code to identify the beginning of a data sequence. The unit is now ready to "read" package bar-code labels by passing the light pen across the bars on the label. Only the label on the outside of a sealed safe containing labelled packages is "read"; the computer will later resolve that label into a list of the labels on the packages within the safe. If any label becomes unreadable by the light pen the reference number can be typed in directly via the keyboard although this facility has not yet been found necessary. At the end of the inventory taking operation the operator enters a terminate code to mark the end of the data sequence.

5.2 Data Transfer

A transmitter unit is used to transfer data from the tape cassette along a cable to the computer for storage on disk. The complete PDCU is placed in the transmitter unit and electrical and mechanical contacts are made through the base of the unit. Once data transfer has started, the operation of the PDCU is controlled by the transmitter unit and intervention by the operator is kept to a minimum. During transmission, the data from the cassette are copied to a temporary file on the computer disk to enable the PDCU to be released for further use.

5.3 Data Processing - Production of a Physical Inventory Listing and Inventory Discrepancies

The computer data file holds a complete package record indexed to package number as contained in the bar-code.

The temporary file created during the data transfer is processed sequentially, each label number opening the record for that particular package. Data relevant for a PIL is extracted and produced at a hard-copy printer in a standard
PIL format. Before printing, the data are checked automatically for anomalies and if anomalies are present, they are stored and processing continues to the next label number. All anomalies are printed when processing of the temporary file is complete.

The various anomalies which are identified are:

i Package not found

ii Package found but not on computer file

iii Package found in wrong KMP

iv Package found which should have been despatched

v Repeat monitoring of same label code by the operator.

5.4 Rapid Discrepancy Indication

A special inventory-taking computer program has been written which will process the PDCU bar-code data rapidly and will thereby identify missing packages and other selected anomalies. This program does not print out a PIL, only the anomalies.

6. EXPERIENCE

New operators are speedily trained in the use of the PDCU, and all the operators have expressed a preference for this method of data collection over the previous manual transcription method.

The electronic units have proved very reliable over two years of operation, with only one overcharged battery pack needing replacement.

There has not been a noted incident when the PDCU has incorrectly read a label and stored the data. Either the label is read correctly and the data stored or the label is not read correctly, the alarm sounds, and the data are discarded.

The PILs carried out in the NMS using the Plessey light pen equipment now take five man-days to complete in comparison with sixty man-days by the earlier manual system. The data, previously hand written in the NMS...
and reprocessed in a Special Materials Record section, is now produced in the format specified by the safeguarding authorities and can be supplied as print-out or on magnetic tape for onward transmission as required.

7. **FURTHER DEVELOPMENTS**

The possibility of switching labels is of concern and tamper-proof labels are being investigated. Other future developments include the use of the model 1488 PDCU with solid-state memory for data collection and storage, and the connection of the 1115 data capture unit and an electronic balance directly to the computer, via a cable with outlets throughout the store, for package label and gross weight verification.

**APPENDIX 1**

**INFORMATION RELEVANT TO AN INCOMING PACKAGE FOR STORAGE IN THE DATA BANK**

<table>
<thead>
<tr>
<th>Field</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Package Label</td>
<td>Numerical equivalent of Bar Code</td>
</tr>
<tr>
<td>Date In</td>
<td>Date of entry into the Store</td>
</tr>
<tr>
<td>SMTF No. In</td>
<td>Number of the Special Material Transfer Form covering the movement</td>
</tr>
<tr>
<td>Owner</td>
<td>Owner of the material, eg UKAEA</td>
</tr>
<tr>
<td>Origin</td>
<td>Code for the sender's material balance area</td>
</tr>
<tr>
<td>Uranium Enrichment</td>
<td>Weight per cent</td>
</tr>
<tr>
<td>MO No.</td>
<td>Material Order Number</td>
</tr>
<tr>
<td>No. of items</td>
<td>Number of items making up a single package</td>
</tr>
<tr>
<td>Description</td>
<td>The code for the contents and container</td>
</tr>
<tr>
<td>Customer Ref.</td>
<td>Customer Reference</td>
</tr>
<tr>
<td>Measurement code</td>
<td>Euratom 2 character code</td>
</tr>
</tbody>
</table>
Chemical Form

These data are optional but may be helpful at a future time

Physical Form

Container type

Weights - all weights in grams

Gross weight

Nett weight

Weight of Plutonium

Weight of Depleted Uranium

Weight of Total Enriched Uranium

Weight of Uranium 235

Weight of Thorium

Weight of Uranium 233

Plutonium Type

Moderator Type - if present in large quantity

Weight of Moderator

Chemical Analysis Number

KMP Key Measurement Point

Location

Comments 1)

Comments 2) For general comments

Comments 3)

Comments 4)

SLOT 1) For use by Special Materials Section

SLOT 3)

Transaction Two character code for the type of inventory change occurring
АВТОМАТИЗАЦИЯ СИСТЕМЫ УЧЕТА ЯДЕРНЫХ МАТЕРИАЛОВ НА АЭС С РЕАКТОРАМИ ВВЭР-440 ДЛЯ ЦЕЛЕЙ ГАРАНТИЙ

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Abstract – Аннотация

COMPUTERIZATION OF THE NUCLEAR MATERIAL ACCOUNTING SYSTEM FOR SAFEGUARDS PURPOSES AT NUCLEAR POWER PLANTS WITH WWER-440 REACTORS.

The paper sets forth the basic principles underlying nuclear material accounting at nuclear power plants with WWER-440 reactors. It briefly describes the general structure and individual units in a program for computerized accounting. The use of this program is illustrated by the actual accounting data from the fifth unit of the Novovoronezh nuclear power station. The NUMIS program seems to be of interest both for the purposes of IAEA safeguards and for nuclear power plant operators in countries where power plants with WWER-440 reactors subject to IAEA safeguards are either in operation or under construction. The research in question was conducted initially under an IAEA research contract; the system is now being developed further and tested under the IAEA-USSR technical and scientific co-operation programme on safeguards.

ВВЕДЕНИЕ

Опыт осуществления гарантий МАГАТЭ на крупных АЭС с реакторами ВВЭР-440 свидетельствует о том, что в целях обеспечения более оперативного и эффективного
анализа учетной информации во время проведения независимых проверок инспекторами Агентства, а также повышения надежности и оперативности представления отчетов МАГАТЭ, существует необходимость создания автоматизированной системы учета.

Широкая программа строительства крупных АЭС с реакторами ВВЭР-440 за рубежом приведет не только к значительному росту мощностей ядерной энергетики, но и к резкому возрастанию потоков ядерных материалов. В связи с этим значительно повышается роль автоматизированных систем учета ядерных материалов на АЭС.

В СССР проводятся разработка, испытание и демонстрация такой системы, которая отвечала бы как требованиям МАГАТЭ, так и нуждам операторов АЭС. Первоначальный вариант программы, который был разработан в рамках исследовательского контракта с МАГАТЭ [1], в результате испытаний на реальных учетных данных был подвергнут ряду изменений и усовершенствований.

Цель данного доклада — информировать Агентство и заинтересованные в этой системе государства о ходе дальнейшего развития проекта, в частности, об испытаниях системы на реальных учетных данных IV блока НВАЭС.

1. ПРИНЦИПЫ УЧЕТА И ТРЕБОВАНИЯ К УЧЕТНОЙ ДОКУМЕНТАЦИИ

Основные принципы учета ядерных материалов (ЯМ) на АЭС с реакторами ВВЭР-440, которые лежат в основе программы автоматизированного учета, вытекают из следующих технологических и физических особенностей станции [2, 3]:
- на АЭС ЯМ присутствует в виде неразборных тепловыделяющих сборок, имеющих идентификационные номера. Это позволяет считать каждую сборку партией и вести поштучный учет;
- партии ЯМ перемещаются по станции редко, их движение носит дискретный характер и каждая сборка локализована либо на складе, либо в реакторе, либо в хранилище. Это дает основание ввести на АЭС три основных типа ключевых точек измерения: склад свежих топливных сборок, реактор, хранилище облученного топлива. В зону материального баланса входят все три ключевые точки измерения;
- в качестве исходной информации о сборках, поступающих на АЭС, используются паспортные данные завода-изготовителя. К ним относятся: идентификационный номер, начальное обогащение, общий вес U и его изотопа \(^{235}\text{U}\) в сборке и т.п.;
- для определения изотопного состава в выгруженном топливе используются сведения службы эксплуатации АЭС, которая проводит расчеты трехмерных распределений выгорания по активной зоне на основе типовых программ нейтронно-физического расчета ВВЭР-440 [4]. Зная распределение выгорания, можно рассчитать по стандартным таблицам изотопный состав топлива в выгруженных сборках.

Таким образом, принципиальная схема учета, представленная в программе "NUMIS", выглядит следующим образом [1]:
- осуществляется регистрация всех поступающих на АЭС топливных сборок. По соответствующей схеме расчета на терминалах заносятся идентификационные
номера, общие количества U и 235U в сборках, значения начального обогащения и условный код зоны баланса материалов (ЗБМ), из которой поступила сборка, с регистрацией даты поступления сборок на АЭС;
— предусмотрена регистрация перемещения сборок в ЗБМ АЭС: со склада необлученного топлива в реактор, из реактора в хранилище облученного топлива (или обратно на "дожигание"), перестановки в реакторе и хранилище с регистрацией даты перестановки и изменения координат местоположения сборок;
— для сборок, выгруженных из реактора в хранилище отражается изменение изотопного состава топлива — ядерные потери (NL), выраженные в уменьшении количеств U и 235U, и выработка изотопов плутония (NP);
— регистрируется отправка сборок с АЭС — идентификационные номера и код ЗБМ, в которую отправляются сборки;
— на основе учетных документов составляются отчеты типа MBR, ICR, PIL, требуемые Агентством.
Учетная документация на ЯМ является важной составной частью системы учета и контроля ЯМ на установках топливного цикла и непосредственно на АЭС с реакторами ВВЭР-440. Общие требования МАГАТЭ к учетной документации АЭС изложены в разделах "Система учетных документов" и "Система отчетов" документа [5] и сводятся к следующему:
а) предусматривается ведение учетных документов:
— материально-балансовых, в которых для каждой ЗБМ отражаются все изменения инвентарных количеств ЯМ;
— эксплуатационных, в которых содержатся эксплуатационные данные установки, используемые для определения изменений в количестве и составе ЯМ;
б) по каждой зоне материального баланса в МАГАТЭ должны представляться следующие отчеты:
— отчеты об изменениях инвентарных количеств ЯМ (ICR);
— отчеты о результатах физической инвентаризации (PIL);
— материально-балансовые отчеты (MBR), показывающие материальный баланс в ЗБМ за некоторый промежуток времени.

2. КРАТКОЕ ОПИСАНИЕ ПРОГРАММЫ "NUMIS"

Программа "NUMIS" (Nuclear Material Information System) является информационно-поисковой и обрабатывающей программой, которая позволяет хранить в памяти ЭВМ полный объем учетных данных по ЯМ на АЭС и анализировать эти данные. Программа одновременно может хранить в банке данных все необходимые учетные сведения о 16 000 топливных сборках для 4-х реакторов за 30 лет (приблизительный срок службы АЭС). Физическим носителем банка данных служит магнитный диск.
В целях унификации программа "NUMIS" ориентирована на использование ЭВМ "единой серии", ЕС ЭВМ, получивших широкое распространение. Язык программии-
Управляющая программа

Загрузочный модуль

- организация банка данных
- ввод информации о конструкции
- ввод данных о партиях ЯМ
- ввод эксплуатационных данных
- контроль

Модуль обработки банка данных

- запись
- редактирование
- поиск данных

Модуль печати

- печать входной информации
- печать изотопного состава топлива
- печать отчетов ICR
- печать отчетов PIL
- печать отчетов MBR

Рис. 1. Общая блок-схема программы "NUMIS".

рования — FORTRAN-IV. Программу можно эксплуатировать на широком классе компьютеров, использующих операционную систему типа IBM или ей подобную.

Структура программы (рис. 1) носит модульный характер и состоит из 3-х крупных модулей (загрузочный модуль, модуль обработки банка данных, модуль печати) и управляющей программы.

Каждый из функциональных модулей включает в себя набор простых специализированных подпрограмм. Обмен информацией между подпрограммами осуществляется через банк данных и COMMON-блоки.

Модульная структура программы отвечает двум требованиям. Во-первых, она обеспечивает легкую взаимозаменяемость и модернизацию отдельных блоков. Во-вторых, программа является незамкнутой, т. е. к ней могут быть просто подключены новые блоки, соответствующие дополнительным требованиям к системе учета.
В функции загрузочного модуля входит ввод разнообразных учетных данных в оперативную память ЭВМ и помещение информации о конструкции и эксплуатационных данных в банк. Носителями информации о конструкции и сведений о движении топливных сборок являются "бумажные" документы, формы или рабочие записи оператора. В этом случае данные приходится вводить в компьютер с перфокарт (или с терминала при его наличии). При испытаниях программы "NUMIS" использовался ввод данных с помощью перфокарт. Входная информация очень разнообразна и объемна, поэтому для ввода данных с помощью перфокарт применяны специальные подпрограммы бесформатного ввода, разработанные в ИАЭ им. И. В. Курчатова [6]. Эти подпрограммы позволяют вводить наборы данных, чередуя в произвольной последовательности числовую и текстовую информацию. При этом каждый элемент данных не обязательно должен вводиться в строго фиксированном формате. Такая система очень удобна для пользователя, она упрощает подготовительную запись наборов данных на типовые бланки для последующей набивки на перфокарты или для ввода данных с терминала.

Прежде чем записать входную информацию в банк данных, необходимо пропустить данные через программу контроля. Данная программа осуществляет поиск грубых ошибок (таких, например, как превышение допусков в отклонении фактического обогащения от номинала, или отсутствия на АЭС склада, реактора, хранилища с данным именем и т.д.). Подобная программа контроля резко снижает количество случайных ошибок при подготовке информации. Единственный параметр, который необходимо тщательно проверять путем построчной проверки, это — идентификатор кассеты, поскольку он не подвергается логическому контролю.

Модуль обработки банка данных предназначен для анализа и обработки загруженных в оперативную память сведений о перемещениях сборок и внесения соответствующих редакций в банк данных. Модуль обработки банка данных последовательно для каждой сборки осуществляет следующую группу операций:
- выборка из COMMON-блока сведений о данной сборке и упорядочивание их;
- анализ ассоциативного файла и определение номера записи о данной сборке в банке данных, если такая запись уже была сделана ранее. Дескрипторами при поиске служит идентификатор сборки;
- считывание из банка данных нужной записи;
- определение изотопного состава топлива для сборок, выгружаемых из реактора;
- корректирование выбранной записи;
- помещение откорректированной записи в банк на место старой;
- коррекция ассоциативного файла.

Параллельно, путем суммирования по всем сборкам, определяются интегральные количества получений, отправлений ЯМ, ядерного производства и потерь. Эти величины, вместе с соответствующими датами, записываются в специальный, так называемый, балансовый файл банка данных для формирования отчетов типа MBR.

Кроме вышеописанной основной функции модуль обработки банка данных снабжен специальными подпрограммами, которые позволяют осуществлять выборки сведений:
— о сборках с заданными идентификаторами;
— о сборках, находящихся в заданной ключевой точке измерения (склад, реактор или хранилище).
Такой поиск может оказаться полезным при проведении физической инвентаризации или инспекции.
Модуль печати включает в себя ряд специальных подпрограмм, осуществляющих печать различных входных данных и результатов расчета. По каждой ЗБМ государство должно периодически представлять в Агентство отчеты типа ICR, PIL, MBR. Соответственно в модуле печати предусмотрены подпрограммы, позволяющие автоматически печатать такие отчеты в виде таблиц, аналогичных формам Агентства R.01.1; R.02/C; R.03. Подобные отчеты могут быть направлены в Центральную службу ГСУК. Центральная служба обычно формирует отчеты для представления в Агентство в иной форме, например на магнитной ленте. Программа "NUMIS" способна создавать отчеты типа ICR, MBR, PIL на магнитной ленте с дальнейшей пересылкой их в национальный центр учета.

3. ИСПЫТАНИЕ АВТОМАТИЗИРОВАННОЙ СИСТЕМЫ УЧЕТА ЯДЕРНЫХ МАТЕРИАЛОВ "NUMIS" НА РЕАЛЬНЫХ УЧЕТНЫХ ДАННЫХ НВАЭС

В качестве объекта для испытания программы использовалась Нововоронежская АЭС (IV блок), полностью отражающая все особенности типовой станции с реакторами ВВЭР-440. Рассмотрены все возможные технологические операции с топливом. Зона материального баланса включала в себя склад свежего топлива, реактор, хранилище облученного топлива. Также учитывались данные при получении свежего топлива, отправлении отработавшего топлива и перемещении сборок внутри зоны материального баланса. Испытание работоспособности автоматизированной системы учета проводилось на большом количестве данных (~ 100 кассет), включающих в себя данные за 4 кампании реактора.

В результате испытания программы "NUMIS" на реальных учетных данных АЭС выявились необходимость введения четырех дополнительных схем расчета:
— проверка на идентичность кассет в зоне материального баланса;
— коррекция идентификатора кассеты в файле данных о перемещениях топливной сборки;
— коррекция начальных весов U и 235U в файле данных о перемещениях топливной сборки;
— коррекция значения глубины выгорания топливной кассеты, выгружаемой в хранилище облученного топлива.
Необходимость введения первых двух схем расчета обуславливается тем, что на примере реальных учетных данных, представленных станцией, были выявлены более 30 кассет парной идентификации в зоне баланса ЯМ. Наличие парных кассет в зоне материального баланса отрицательно сказывается на анализе ассоциативного файла.
и, в конечном итоге, на определение номера записи о парной сборке в банке данных, ибо лескриптором при поиске кассеты служит идентификатор сборки. Иначе говоря, при анализе банка данных на случай парных кассет в ЗБМ произойдет со временем искажение информации в файле данных о перемещениях парных топливных сборок. Надо отметить, что в настоящее время завод-изготовитель перешел на пятизначную идентификацию топливных кассет, что позволит значительно снизить вероятность поступления идентичных кассет в зону материального баланса. Тем не менее проверка на идентичность кассет в зоне материального баланса будет весьма полезна.

Программа коррекции идентификатора кассеты в файле данных о перемещениях топливной сборки была вызвана предыдущей схемой расчета (проверкой на идентичность кассет) и в случае обнаружения таких кассет производится коррекция идентификатора кассеты в файле данных, путем добавления какого-либо индекса к существующей идентификации. Другая причина, побудившая создавать программу по коррекции идентификатора кассеты, заключается в исправлении механических ошибок, вызванных записью данных оператора на бланке набивки перфокарт или использование данных оператора с помощью дисплейного ввода.

Программа коррекции начальных весов U и 235U в файле данных о перемещениях топливной сборки была вызвана возможностью исправления случайных ошибок при набивке начальных весов U и 235U на перфокарты в пределах допусков в отклонении фактического обогащения от номинала. При этом корректируется также балансный файл.

В случае ошибочной записи в банк данных значения глубины выгорания топливной кассеты в файле данных о перемещениях топливной сборки, необходимо провести коррекцию этого значения с последующим исправлением изотопного состава топлива. Одновременно корректируется балансный файл.

ЗАКЛЮЧЕНИЕ

В результате испытания программы "NUMIS" на реальных учетных данных НВАЭС можно сделать следующие выводы:

1. Испытание программы продемонстрировало ее работоспособность и удобство в эксплуатации. Программа позволяет автоматизировать учет всех стандартных технологических операций, связанных с получением, хранением, перемещением, отправкой ядерных материалов.

2. Программа удовлетворяет как требованиям Агентства в отношении учета ЯМ, так и собственным нуждам станции. Предусматривается автоматизированное составление отчетов типа ICR, PIL, MBR — возможно представление подобных отчетов на магнитной ленте. К тому же программа выполняет ряд информационно-поисковых операций, упрощающих ревизию системы учета и облегчающих проведение физических инвентаризаций.

3. Были выявлены в зоне материального баланса несколько десятков кассет парной идентификации, что безусловно противоречит основным принципам учета и
затрудняет проведение автоматизированного учета. В связи с этим появилась необходимость введения в программу двух дополнительных схем расчета: а) проверка на идентичность кассет в зоне материального баланса; б) коррекция идентификатора кассеты в файле данных о перемещениях топливной сборки.

Периодическая проверка на идентичность кассет в зоне материального баланса будет весьма полезна, особенно на случай новых поступлений в зону.

4. В целях исправления случайных ошибок в значениях начальных весов U, \(^{235}\)U в пределах допусков отклонения фактического значения обогащения от номинала, а также значение глубины выгорания топливной кассеты в файле данных о перемещениях топливной сборки выявила необходимость введения в программу еще двух дополнительных схем расчета: а) коррекция начальных весов U и \(^{235}\)U в файле данных о перемещениях топливной сборки; б) коррекция значения глубины выгорания кассеты в файле данных о перемещениях топливной сборки.

5. Для стран, не имеющих развитой национальной системы учета на основе автоматизации, данная программа имеет первостепенное значение для целей гарантий и учета. Наряду с этим для стран с хорошо развитой национальной системой учета программа "NUMIS" могла бы служить исходным звеном в системе автоматизированного учета ядерного материала: установка — центр государственной системы учета и контроля — Агентство.

6. Дальнейший этап развития данной работы заключается в проведении испытания программы в условиях АЭС, на которых ядерного материала: установка — центр государственной системы учета и контроля — Агентство.

Авторы доклада выражают свою признательность тт. Бабаеву Н.С. и Позднякову Н.Л. за значительный вклад в развитие исследовательского контракта [1].

**ЛИТЕРАТУРА**


NEKOTORYE KONSTRUKCIIONNYE I TECNOLOGICHESKIE OSOBNENNOSTI AES S REAKTOROM BBEP-1000 PRIMEJNIETELNO K GARANTIYAM MAGATE

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Abstract—Аннотация

SOME STRUCTURAL AND ENGINEERING FEATURES OF NUCLEAR POWER PLANTS WITH WWER-1000 REACTORS WHICH RELATE TO THE APPLICATION OF IAEA SAFEGUARDS.

The IAEA has been successfully using procedures and technical devices for the application of international safeguards to nuclear power plants with the WWER-440 water cooled and moderated reactor. The USSR has started constructing a new series of nuclear power plants with the WWER-1000 water moderated and cooled reactor, and the first unit in this series went into operation in 1980 at the Novovoronezh nuclear power station (the fifth unit of the station). Using it as an example, the authors consider the structural features and engineering characteristics which may be of use in developing safeguards procedures and devices for power plants of this type in the future. They also discuss some structural modifications which may be incorporated in the subsequent units in the series.
## Таблица 1. Сравнительные характеристики реакторных установок ВВЭР

<table>
<thead>
<tr>
<th>Параметры</th>
<th>ВВЭР-440</th>
<th>ВВЭР-1000 (головной блок серии — 5-ый блок НВАЭС)</th>
<th>ВВЭР-1000 I этап модификации</th>
<th>ВВЭР-1000 II этап модификации</th>
</tr>
</thead>
<tbody>
<tr>
<td>Электрическая мощность, МВт</td>
<td>440</td>
<td>1000</td>
<td>1000</td>
<td>1000</td>
</tr>
<tr>
<td>Тепловая мощность, МВт</td>
<td>1375</td>
<td>3000</td>
<td>3000-32000</td>
<td>3000-3200</td>
</tr>
<tr>
<td>Эквивалентный диаметр активной зоны, см</td>
<td>288</td>
<td>312</td>
<td>316</td>
<td>316</td>
</tr>
<tr>
<td>Высота активной зоны, см</td>
<td>250</td>
<td>350</td>
<td>350</td>
<td>350</td>
</tr>
<tr>
<td>Удельная мощность, кВт/л</td>
<td>83</td>
<td>111</td>
<td>107-115</td>
<td>107-115</td>
</tr>
<tr>
<td>Число тепловыделяющих сборок (ТВС)</td>
<td>349</td>
<td>151</td>
<td>163</td>
<td>163</td>
</tr>
<tr>
<td>Число твэлов в ТВС</td>
<td>126</td>
<td>331</td>
<td>331</td>
<td>331</td>
</tr>
<tr>
<td>Форма и тип ТВС</td>
<td>шестигранник с чехлом</td>
<td>шестигранник без чехла</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Размер кассеты под ключ, мм</td>
<td>144</td>
<td>238</td>
<td>234</td>
<td>234</td>
</tr>
<tr>
<td>Загрузка топлива в активную зону (UO₂) t</td>
<td>47,6</td>
<td>75,5</td>
<td>80</td>
<td>80</td>
</tr>
<tr>
<td>Обогащение свежего топлива в стационарном режиме, %</td>
<td>3,5</td>
<td>3,3/4,4</td>
<td>3,3/4,4</td>
<td>4,0-4,4</td>
</tr>
<tr>
<td>Средняя глубина выгорания топлива, МВт-сут/кг</td>
<td>28,6</td>
<td>27/40</td>
<td>27/40</td>
<td>40</td>
</tr>
<tr>
<td>Наружный диаметр и шаг расположения твэлов, мм</td>
<td>9,1/12,2</td>
<td>9,1/12,75</td>
<td>9,1/12,75</td>
<td>9,1/12,75</td>
</tr>
<tr>
<td>Длительность кампании топлива, год</td>
<td>3</td>
<td>2/3</td>
<td>2/3</td>
<td>3</td>
</tr>
<tr>
<td>Число перегрузок за кампанию</td>
<td>3</td>
<td>2/3</td>
<td>2/3</td>
<td>3/6</td>
</tr>
</tbody>
</table>
(ВВЭР). МАГАТЭ в течение многих лет успешно применяет гарантии в ряде стран к АЭС советского производства с реакторами ВВЭР-440 и накопило значительный опыт по осуществлению гарантий на них [2, 3]. В последние годы СССР приступил к производству нового поколения АЭС с реакторами более высокой мощности ВВЭР-1000, начав промышленную эксплуатацию головного блока серии таких станций в 1980 г. вблизи г. Воронеж.

Этот блок, обладающий мощностью 1000 МВт (ел.), является головным для большой серии АЭС, которые планируется построить как в СССР, так и в других странах [1, 4], в которых МАГАТЭ осуществляет гарантии. В этой связи ниже на примере головного блока рассматриваются некоторые особенности АЭС этого типа, которые, по мнению авторов, являются важными для последующей разработки процедур и выбора технических мер осуществления гарантий на таких установках. Опыт эксплуатации головного блока [5,6] подтвердил правильность основных проектных и конструкционных решений, хотя это и не исключает некоторых модернизаций на последующих блоках с целью оптимизации технических параметров и упрощения компоновки АЭС в целом.

В табл.1 приведены сравнительные характеристики АЭС с реакторами ВВЭР-440, ВВЭР-1000 (головного блока), а также двух этапов последующих планируемых модификаций для серийных АЭС с реакторами ВВЭР-1000. Первый этап модификации, осуществляемый на Южно-Украинской и Калининской АЭС, предусматривает, главным образом, внесение изменений в конструкцию самого реактора. Здесь следует отметить, что в активной зоне реакторов этих АЭС предусматривается разместить 163 тепловыделяющих сборки (ТВС), вместо 151, при сохранении внутреннего диаметра корпуса и шага расположения тепловыделяющих элементов (твэлов), что достигается за счет использования бесчехловых конструкций ТВС. Предполагается, что на втором этапе модернизации основные изменения, принятые для конструкции реактора на первом этапе, будут сохранены и основное внимание будет сосредоточено на применении более совершенного оборудования, упрощении компоновки АЭС, дальнейших технологических усовершенствованиях. Ожидается, что в результате ряда технических мероприятий, например организации транспортировки внутриреакторных устройств во время перегрузки под водой, объем защитной оболочки сократится на 20-30%.

С точки зрения гарантий важно отметить, что для ВВВР-1000, вводимых в строй до конца десятилетия, не следует ожидать использования иного ядерного топлива, чем двуокись урана [6].

На 5-ом блоке Нововоронежской АЭС (НВАЭС) установлен один реактор ВВЭР-1000. Реактор находится внутри защитной оболочки, доступ под которую во время работы ограничен. На рис. 1 представлена схема движения ядерного материала (ЯМ) на этой станции. Свежие ТВС поступают на АЭС по железной дороге. В каждом вагоне прибывают контейнеры, в каждом из которых содержатся 2 ТВС. Вагоны въезжают в здание узла свежего топлива, контейнеры выгружаются из них краном и помещаются на стеллажи в горизонтальном положении, где они хранятся.
Рис. 1. Схема движения ядерного материала на 5-ом блоке НВАЭС с реактором ВВЭР-1000:

- — движение свежего топлива;
- — — движение облученного топлива.

dо начала перегрузки реактора. Во время перегрузки контейнеры устанавливаются в вертикальное положение кантователем, из них извлекаются ТВС, которые затем загружаются в чехлы цилиндрической формы. Используются два типа чехлов, вмещающих 7 и 19 ТВС, в отличие от чехлов для ВВЭР-440, вмещающих 30 ТВС.

Конструкция узла свежего топлива позволяет идентифицировать извлекаемые из контейнеров ТВС, а также проводить выборочные контрольные измерения неразрушающими методами.
Рис. 2. Карточка первой топливной загрузки активной зоны реактора ВВЭР-1000 5-го блока НВАЭС.
ТАБЛИЦА II. ЧИСЛО ТВС И ИХ ОБОГАЩЕНИЕ ДЛЯ ПЕРВОЙ ЗАГРУЗКИ ВВЭР-1000 (5-ый блок НВАЭС)

<table>
<thead>
<tr>
<th>Число ТВС</th>
<th>Обогащение по урану</th>
<th>Общее количество ТВС в активной зоне</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>%</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3,3</td>
<td>3,0</td>
</tr>
<tr>
<td></td>
<td>2,0</td>
<td></td>
</tr>
<tr>
<td>Без поглотителей СУЗ</td>
<td>42</td>
<td>–</td>
</tr>
<tr>
<td>С поглотителем СУЗ</td>
<td>1</td>
<td>30</td>
</tr>
<tr>
<td>Итого:</td>
<td>43</td>
<td>30</td>
</tr>
</tbody>
</table>

Чехлы с ТВС во время перегрузки транспортируются под защитную оболочку через люк. ТВС извлекаются из чехлов и устанавливаются в гнезда бассейна перегрузки, в котором они могут храниться в сухом виде или под водой. Здесь имеется возможность идентификации ТВС по их номерам. Во время загрузки свежие ТВС транспортируются из бассейна перегрузки в активную зону, а облученные из активной зоны — в бассейн выдержки. Все перегрузочные операции выполняются с помощью перегрузочной машины. На пути транспортировки облученных ТВС имеется возможность установки счетчиков ТВС. В помещении оболочки около бассейна выдержки для проведения измерений облученных ТВС можно установить стенд для проведения выборочных контрольных измерений неразрушающими методами.

Гнезда для установки ТВС в бассейне перегрузки, активной зоне и бассейне выдержки имеют координатные сети, что облегчит проверку и учет ядерных материалов для целей инспектирования.

Один из рассматриваемых вариантов хранения облученного топлива предусматривает, что после 2-3-летней выдержки облученных ТВС в бассейне выдержки под защитной оболочкой они будут перенесены в долгосрочное хранилище.

В основу выбора картограммы первой топливной загрузки реактора ВВЭР-1000 положены принципы симметрии загрузки в плане и уплощения распределения энерговыделения по активной зоне за счет загрузки ТВС с топливом различного обогащения. На рис. 2 представлена картограмма активной зоны для первой топливной загрузки реактора ВВЭР-1000 5-го блока НВАЭС, а в табл. II приведены количества ТВС с топливом различного обогащения для этой загрузки.

Перегрузки топлива осуществляются в режиме движения следующим образом. Из центральной части активной зоны выгружаются ТВС, имеющие максимальное выгорание, на их место переставляются ТВС из средней части активной зоны, которые восполняются перестановкой ТВС с периферии.

В первые перегрузки выгружается примерно половина облученных ТВС. В дальнейшем в режиме стационарных перегрузок предполагается извлекать в среднем одну треть часть ТВС.
В рабочем состоянии (при давлении внутри реактора 160 атм) корпус реактора закрыт крышкой, и, следовательно, доступ к активной зоне исключен. Это облегчает применение гарантий, поскольку движение ЯМ нет, а крышка реактора может быть опечатана пломбами МАГАТЭ, обеспечивая инспекторам уверенность в отсутствии возможности для недозволенного облучения в реакторе неучтенных материалов и несвоевременного извлечения заявленных ТВС.

Доступ к активной зоне при снятой крышке во время перегрузки обеспечивает возможность организации проверки наличия заявленных ТВС как во время рутинных инспектций, так и во время проведения физических инвентаризаций.
Весь ЯМ на АЭС находится в герметически упакованных крупных тепловыделяющих сборках — ТВС. Схематическое изображение ТВС приведено на рис. 3. Отдельная ТВС собрана из твэлов, трубок для стержней регулирования, несущей центральной трубы, трубки для канала измерения энерговыделения, наружных стенок-чехлов, нижнего концевика для установки в гнезда и верхней головки с устройством для захвата ТВС штангой перегружочной машины. ТВС собрана таким образом, что представляет собой единое целое, и разборку, извлечение или замену одного или нескольких твэлов на АЭС без специального оборудования произвести невозможно; для этих целей ТВС должна быть отправлена на завод-изготовитель. Именно благодаря интегральной компоновке, идентификации и пересчет ТВС вместе с мерами наблюдения, сохранения и выборочных неразрушающих измерений остаются наиболее важными способами осуществления гарантий. Поскольку при поступлении ТВС на АЭС измерений ЯМ не производится, то учет количеств ЯМ в ТВС осуществляется на основе паспортных данных завода-поставщика.

На каждой ТВС имеется идентификационный номер, расположенный на скосе головки и видимый при горизонтальном или вертикальном положении ТВС, в том числе под слоем воды в бассейне перегрузки, активной зоне со снятой крышкой и бассейне выдержки. Для уточнения номера в условиях плохой видимости может быть использована одна из штатных телевизионных камер АЭС. Каждая ТВС с точки зрения гарантий является самостоятельной партией ЯМ, а идентификационный номер — ее именем.

В отличие от АЭС с реактором ВВЭР-440 обсуждаемая атомная станция имеет как реактор, так и бассейны для перегрузки свежего топлива, бассейн выдержки (хранилище облученного топлива до 2-3 лет), размещенные под единой защитной оболочкой, которая ограничивает доступ персонала в эту зону. Защитная оболочка имеет один проход для персонала и один запасной проход для этих целей, а также люк для загрузки/выгрузки свежего и отработанного топлива. Свежее топливо поступает в бассейн перегрузки, а отработанное загружается в транспортный вагон при его отправке на ядерную установку. Полагая, что при нормальной эксплуатации станции использование этих проходов и люка в период между остановками реактора, вероятно, будет крайне редким, то представляется целесообразным их опечатывание, либо применение иных технических средств наблюдения МАГАТЭ, в дополнение к средствам обычно используемым Агентством внутри реакторных залов АЭС с реакторами ВВЭР-440. Применение таких мер на АЭС ВВЭР-1000 может позволить повысить эффективность международных гарантий, т.к. даст дополнительную уверенность Агентству в сохранности значительного количества облученного ядерного материала, представляющего наибольшую стратегическую ценность на АЭС.

Размещения хранилищ ТВС и перегружочного оборудования под защитной оболочкой реактора позволяет, в основном, успешно использовать технические средства, которыми пользуется МАГАТЭ при проведении инспекций на АЭС ВВЭР-440, например, опечатывание реактора, идентификация ТВС по заводским номерам при поштучном учете в хранилище, выполнение НДА, установка оптических или электронных средств наблюдения и т.д. Однако, имеется и ряд отличий между этими
реакторными залами. Во-первых, на ВВЭР-1000 в отличие от ВВЭР-440 не использу­
зуется один и тот же бассейн как для свежих, так и облученных ТВС во время перег­
рузок топлива, что значительно упрощает анализ информации, накопленной средст­
вами наблюдения на магнитных или фильмовых лентах. Во-вторых, появляется дополнительная ключевая точка измерений — бассейн перегрузки свежего топлива. 
Следует также отметить, что под защитной оболочкой имеется специальная штатная телевизионная камера, которая может быть использована при идентификации ТВС в бассейнах. Кроме того, в этой же зоне находится ряд телевизионных камер, управ­
ляющих дистанционно с основного пульта АЭС. В этой связи представляется тех­
нически возможным параллельное использование некоторых из них для видеозапи­
сывающих устройств МАГАТЭ [7], что может быть предметом обсуждения между 
Агентством и оператором будущих АЭС с реактором ВВЭР-1000. Положительное решение этого вопроса могло бы дать экономический эффект и облегчить работы 
по монтажу телевизионных систем МАГАТЭ на станциях.

В соответствии с принципами организации гарантий, изложенными в документе 
МАГАТЭ [8], применение гарантий на АЭС с ВВЭР-1000 основывается на соблюдении принципа баланса ЯМ и учета их изменений по зонам баланса материалов (ЗБМ). В соответствии с принятой в МАГАТЭ практикой предлагается определить одну ЗБМ 
для всей АЭС, что соответствует принципам сведения баланса ЯМ на одну дату и 
минимизации числа ЗБМ [8, 9]. В таком случае ЗБМ этой АЭС будет содержать ключевые точки измерений (КТИ) двух типов, первый из которых предназначен для 
определения потока ЯМ, второй — для подведения баланса инвентарных количеств ЯМ (рис. 1).

КТИ для определения потока ЯМ будут:

КТИ-1 — получение ЯМ на АЭС;
КТИ-2 — изменение количеств (производство или использование);
КТИ-3 — отсылка ЯМ из АЭС.

КТИ для подведения баланса инвентарных количеств будут:

КТИ-А — узел свежего топлива;
КТИ-В — бассейн перегрузки как промежуточное хранилище свежего топлива;
КТИ-С — активная зона реактора;
КТИ-Д — бассейн выдержки облученного топлива;
КТИ-Е — долгосрочное хранилище облученного топлива;
КТИ-Г — другие помещения на территории АЭС, в которых возможно хранение ЯМ.

С точки зрения гарантий трудность может представить КТИ-А, т. е. узел свежего 
топлива, поскольку поступившие на АЭС ТВС хранятся в опечатанных поставщиком 
контейнерах весь период до начала перегрузки. Одним из решений может быть 
опечатывание контейнеров инспекторами МАГАТЭ или в их присутствии. Учет ЯМ 
в КТИ-А производится только по сопроводительным документам завода-изготови­
теля.

Входная информация о количествах ЯМ на АЭС поступает в виде паспортов 
ТВС, выданных заводом-изготовителем и содержащих идентификационный номер, 
начальное обогащение, общий вес урана, вес изотопа $^{235}$U и другую информацию.
Сведения об изотопном составе выгружаемого из активной зоны топлива берутся из результатов расчетов, выполняемых персоналом службы эксплуатации АЭС по апробированным для ВВЭР вычислительным программам. 

На АЭС предусмотрено ведение учетных документов материально-балансового типа для регистрации потоков и изменений инвентарных количеств ЯМ в ЗБМ и эксплуатационных документов, на основе которых производится определение изменения и состава ЯМ. 

В соответствии с документом МАГАТЭ [8] на АЭС могут составляться три типа отчетов для МАГАТЭ: 
— отчет об изменениях инвентарных количеств ЯМ (ICR); 
— материально-балансовый отчет (MBR); 
— отчет о результатах физической инвентаризации (PIL).

Автоматизация учета ядерных материалов на АЭС повышает оперативность и надежность представления требуемой информации Агентству по соответствующим формам в виде машинных распечаток или на магнитных лентах, а также отвечает интересам оператора, повышая эффективность работы учетной службы. С этой целью в СССР завершаются работы по созданию, испытанию и демонстрации программы "NUMIS", предлагаемой для внедрения на АЭС с реактором ВВЭР-440, стоящих под гарантиями МАГАТЭ. Результаты этих работ представлены в другом докладе на этот Симпозиум [10]. Предполагается, что эта программа может также быть испытана применительно к учетным данным АЭС с реактором ВВЭР-1000 после внесения в нее необходимых изменений.

ЗАКЛЮЧЕНИЕ

Представленные особенности конструкции АЭС с реактором ВВЭР-1000 и схемы потока ядерных материалов на ней свидетельствуют об их специфике по отношению к стоящим под гарантиями в течение ряда лет АЭС с реакторами ВВЭР-440. Эти особенности должны быть учтены при разработке процедур гарантий для АЭС этого типа, т. к. они могут привести к некоторым изменениям, например установлению дополнительных ключевых точек измерения, хотя основные принципы, процедуры и средства, успешно применяемые МАГАТЭ для ВВЭР-440, будут приемлемы и в этом случае. Некоторые конструкционные особенности АЭС, вероятно, могут способствовать повышению эффективности гарантий, например, в случае опечатывания всего ядерного материала под защитной оболочкой, что должно стать предметом дальнейших обсуждений.

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ЛИТЕРАТУРА


DESTRUCTIVE ANALYTICAL MEASUREMENTS

(Session 4)
Chairman
A.H.E. von BAECKMANN
FULLY AUTOMATED, WAVELENGTH DISPERSIVE X-RAY FLUORESCENCE ANALYSIS OF URANIUM AND PLUTONIUM IN REPROCESSING PLANT SOLUTIONS

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Abstract

FULLY AUTOMATED, WAVELENGTH DISPERSIVE X-RAY FLUORESCENCE ANALYSIS OF URANIUM AND PLUTONIUM IN REPROCESSING PLANT SOLUTIONS.

A fully automated, computer-operated X-ray fluorescence system is described. Based on experience gained with earlier systems, and taking into account a plant operator’s requirements, a robust, pneumatically operated modular system was constructed. The problems encountered in adapting the system to glove-boxes; the features of the sample preparation; control of the system; and the calculation of the results, are outlined. A comparison between a scanning and a 7-channel X-ray fluorescence spectrometer was experimentally established. Measurement results are reported.

1. INTRODUCTION

Several years ago the Institute for Radiochemistry at the Karlsruhe Nuclear Research Center (KFK) suggested, after detailed studies [1, 2], to automate the X-ray fluorescence assay of uranium and plutonium. Since then, intensive development work has been carried out on the computer-operated and controlled automated sample preparation in a glove-box.

Long-term laboratory measurements have shown that neither the fission products nor the impurities, nor the high sample radioactivity (1–3 GBq/mL) met in some reprocessing solutions will interfere with the assay. Thus, a chemical separation of uranium and/or plutonium is superfluous, a fact that significantly simplifies the sample preparation before measurement.

Based on a carefully tested prototype design [3], a new sample preparation and handling system was developed.
FIG. 1. (a) Diagram of original sample preparation scheme. (b) Diagram of present sample preparation scheme.
The following basic conditions were considered important and set the requirements to be met by the system:

- The overall system should be flexible enough to cope with safeguards as well as with process control requirements;
- Only robust, reliable and tested components should be used. The components should be easy to obtain; and easy to maintain or exchange;
- Movements should take place by pneumatic rather than by electrical or electromechanical means;
- A high sample throughput should be aimed at; thus, a multichannel instrument was preferred;
- A minimum amount of consumable parts should be used in order to minimize the amount of solid wastes.

All the above conditions are part of the all-important safety requirement of reduced radiation dose to operating personnel.

2. EXPERIMENTAL

2.1. Spectrometer systems

A comparison between wavelength dispersive systems was deemed necessary since the early work [1, 2] was performed with a scanning instrument (SRS), and the automated system was adapted to a multichannel instrument (MRS).

The sequential instrument enabling determination of a number of elements, one after the other, is flexible but relatively slow.

The background fluorescence radiation can be measured easily, and interferences corrected for. A disadvantage is the rather long measurement times at low element concentrations, and when several elements have to be determined in a sample.

The multichannel instrument is preferred when few elements have to be measured repeatedly on samples with a relatively constant composition. For our purpose a 7-channel instrument was chosen since it could be adapted to a glove-box with relative ease.

Each one of the seven channels has its own monochromator, detector and electronic system handling the seven independent, but not identical channels. As a result, the determination of the background and interfering fluorescence radiation becomes difficult when the samples vary in their composition in an unpredictable manner. This is, however, seldom the case in reprocessing.

The multichannel instrument was chosen because of its significantly shorter measurement time. For the three elements in question — uranium, plutonium
and thorium – measurement times of around 5 min are needed compared with 20–30 min for the sequential instrument. Provided that the sample composition is relatively well known, both instruments are comparable in quality.

2.1.1. Glove-box adaptation

The measurement of radioactive samples on a routine basis requires the use of a containment. In our case a glove-box equipped with a thin beryllium window, permitting the primary X-rays to enter the box and the fluorescence radiation to return, was adapted to the spectrometer. The adaptation to the box did not change the original measurement geometry of the instrument and the added beryllium window did not significantly absorb X-rays. Thus, the instrumental characteristics were fully maintained.

In this glove-box the sample changer system was erected. The box is connected to an adjacent sample preparation box through a port system.

2.1.2. The automated sample preparation

The sample preparation and measurement flow-sheet is shown in Fig.1. Experience with the original system has shown that several simplifications were possible. Some of these are given in Fig.1.a.

Every 407 seconds a sample can be measured. This is made possible by carrying out preparation steps on several samples when one sample is being measured.

2.1.2.1. The sample preparation units

The complete system is shown as a block diagram in Fig.2. Two important boundary conditions had to be taken into account, both originating at the WAK reprocessing plant. One is that the space available for the complete system was limited and did not permit heavy shielding in or around the boxes. Thus, a compact construction with a high system reliability was needed. The second plant requirement was the transport of the sample by a pneumatic post system to and from the spectrometer glove-box.

The samples arriving by pneumatic post are stored in a buffer with 15 positions. From here, samples are transferred to the preparation box according to their priority. The random selection of samples from the buffer is, from a process control aspect, a very convenient feature of the system.

In the preparation box aliquotation takes place without opening the sample bottle. Immediately after aliquotation the sample leaves the system via the
FIG. 2. Block diagram.

FIG. 3. (a) Sample containers. (b) Measuring beaker.
pneumatic post system. Figure 3 shows the sample bottle and the measurement beakers.

After measurement the sample is flushed to waste and the sample beaker collected in a 5-L polyethylene bottle prior to disposal.

In the two glove-boxes the 10 specially developed mechanical systems for automatic sample handling are incorporated.

2.1.2.2. Control of the system

The mechanical system performs about 50 functions or movements and is controlled by 90 non-contacting sensors. In addition, the electronic balance and the spectrometer have to be operated and controlled, and their measurement data for each sample accumulated, treated and reported. Some of these data are needed for alternative sample handling during operation. Reliability control and elimination of faulty operation are examples of these analytically necessary interventions.

The control system is flexible, and sample handling is optimized for each expected sample composition.

Frequent calibration runs with standards to verify and adjust the system are carried out. Calibration and quality-control samples constitute about 10% of the load. The balance zero-point control is part of the check routines.

The computer system requires a 16 kbyte program to handle three samples simultaneously. A special microprocessor in connection with a PDP-11/34 is used.

2.1.2.3. Calculation of the results

Each sample is measured three times for 100 seconds. Uranium, plutonium and thorium peaks are measured as well as the background intensity at the Pb-L-gamma 1 wavelength. Net intensities, their mean values and standard deviations are calculated, based on the three measurements. Outlier rejection checks are part of the calculations. The electronic balance supplies weights up to 10 g accurate to 1 mg.

The use of thorium as an internal standard to a large extent eliminates matrix interferences and permits the best calibration curve to be selected.

A comparison between the found and expected U and Pu values of the sample is made; when discrepancies exceeding predetermined limits are detected, these are reported. These features are important for the routine process control in a reprocessing plant.

A cyclic computer memory organization permits the last 2000 results to be stored. These data can be ordered, sorted and treated randomly and constitute the basis for a continuous quality control.
2.1.3. System evaluation

To be able to profit from the system described a high reliability of all components is required. The aim is to operate the instrument 200 days a year, analysing up to 200 samples a day.

All movements are made with corrosion-resistant pneumatic elements. The computer control and evaluation system are optimized. Here, simplicity and reliability are basic requirements in the interest of the operator.

The operation of the system requires the following:

The operator must enter the sample bottle into the pneumatic post system and, in a dialogue with the computer, answer 10 questions. This can take place at any time since the buffer can store and handle incoming samples without interfering with the measurements.

In addition to the 2-minute sample handling and dialogue, about 30 minutes are needed daily for preventive maintenance.

Supply of material to the boxes and removal of waste are taken care of in a conventional way.

The 60 magnetic valves of the pneumatic system are placed in a small separate box. The air from the pneumatic system is removed by the box ventilation system.

Chemicals for cleaning the sample beakers are fed by gravity through magnetic valves into the box.

The use of two separate boxes which can be rapidly disconnected from each other permits, in the event of failures in the sample preparation, manual operation of the spectrometer.

The sample buffer can, owing to the reliable and simple pneumatic post transfer system, be placed at a distance from the rest of the system.

Replicate measurements from one sample bottle can be carried out, an important feature when the sample deviates from the expected composition.

Under optimal conditions the analytical results can be delivered after 13 minutes. With the buffer filled the waiting time is 90 minutes, provided that no priority analysis is requested.

2.2. Measurement results

2.2.1. Reprocessing input solutions

A comparison of performance quality between the sequential and the multichannel instrument with real input solutions was not possible because of the lack of shielding. As has been shown, however, the γ-radiation from the
### TABLE I. COMPARISON BETWEEN SRS AND MRS

<table>
<thead>
<tr>
<th>Sample</th>
<th>Standard values (mg/g)</th>
<th>SRS&lt;sup&gt;a&lt;/sup&gt; Found values (mg/L)</th>
<th>Difference in %</th>
<th>MRS&lt;sup&gt;b&lt;/sup&gt; Found values (mg/g)</th>
<th>Difference in %</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 U:</td>
<td>176.46</td>
<td>176.8</td>
<td>0.2</td>
<td>178.3</td>
<td>1.1</td>
</tr>
<tr>
<td>Pu:</td>
<td>0.777</td>
<td>0.789</td>
<td>1.5</td>
<td>0.793</td>
<td>2.1</td>
</tr>
<tr>
<td>2 U:</td>
<td>172.97</td>
<td>173.1</td>
<td>0.1</td>
<td>174.9</td>
<td>1.1</td>
</tr>
<tr>
<td>Pu:</td>
<td>0.763</td>
<td>0.765</td>
<td>0.3</td>
<td>0.789</td>
<td>3.4</td>
</tr>
<tr>
<td>3 U:</td>
<td>176.28</td>
<td>176.6</td>
<td>0.3</td>
<td>178.7</td>
<td>1.4</td>
</tr>
<tr>
<td>Pu:</td>
<td>1.558</td>
<td>1.555</td>
<td>-0.2</td>
<td>1.566</td>
<td>0.5</td>
</tr>
<tr>
<td>4 U:</td>
<td>173.29</td>
<td>173.9</td>
<td>0.3</td>
<td>176.1</td>
<td>1.6</td>
</tr>
<tr>
<td>Pu:</td>
<td>1.548</td>
<td>1.557</td>
<td>0.6</td>
<td>1.572</td>
<td>1.6</td>
</tr>
<tr>
<td>5 U:</td>
<td>165.09</td>
<td>165.4</td>
<td>0.2</td>
<td>167.9</td>
<td>1.7</td>
</tr>
<tr>
<td>Pu:</td>
<td>0.811</td>
<td>0.823</td>
<td>1.5</td>
<td>0.824</td>
<td>1.5</td>
</tr>
</tbody>
</table>

<sup>a</sup> With Th as internal standard.

<sup>b</sup> Without internal standard.

### TABLE II. RESULTS OF INTERMEDIATE PRODUCTS AND WASTES

<table>
<thead>
<tr>
<th>Concentration range</th>
<th>Relative standard deviation (%)</th>
<th>Absolute error without Th</th>
</tr>
</thead>
<tbody>
<tr>
<td>10—100 mg/L</td>
<td>&lt; 7</td>
<td>2—8 mg/L</td>
</tr>
<tr>
<td>100—1000 mg/L</td>
<td>&lt; 1</td>
<td>5—50 mg/L</td>
</tr>
<tr>
<td>1—10 g/L</td>
<td>&lt; 0.2</td>
<td>0.05—0.3 g/L</td>
</tr>
<tr>
<td>10—100 g/L</td>
<td>&lt; 0.1</td>
<td>0.3—2 g/L</td>
</tr>
<tr>
<td>100—500 g/L</td>
<td>&lt; 0.05</td>
<td>2—7 g/L</td>
</tr>
</tbody>
</table>
sample has no influence on the measurement of U and Pu. Thus, comparisons were made on synthetic input solutions. Table I shows the results obtained under optimum conditions in which standard mixtures with uranium and plutonium concentrations as near to the sample concentration as possible were prepared in successive steps.

The systematic difference observed in the MRS results arises from the lack of a direct background measurement in the neighbourhood of the peak, and measuring without an internal standard.

Based on such measurements, however, the MRS bias can be determined and corrected for. When this is done the results are comparable in quality.

2.2.2. Final products

The measurement of the final products, plutonium and uranium, presents no problem. Before measurement an appropriate dilution is made that reaches optimum concentrations.

The measurement reproducibility is better than 0.1%; the absolute error lies between 0.5 and 0.1%.

**FIG. 4. The complete system.**
2.2.3. Intermediate products and waste

In intermediate products and waste solutions plutonium and uranium can be measured with performance characteristics as shown in Table II. The addition of Th as internal standard does not improve the values in Table II. However, Th addition will correct for systematic errors of different origins (matrix, instrumental, etc.).
3. CONCLUSIONS

The wavelength dispersive X-ray fluorescence spectroscopy system (Figs 4—6) can be a useful instrument for process control, safeguards verification purposes, and static and dynamic inventory-taking at reprocessing plants.

By the use of a fully automated system a high throughput of samples with reliable measurement results can be achieved. The use of modern computers
permits optimum on-line quality control, verification, and reporting of results with a minimum of operator interference. Thus, human errors and radiation dose to personnel can be reduced to a minimum.

The sole disadvantage of the X-ray fluorescence is that reprocessing waste streams with uranium and plutonium concentrations below 5 mg/L cannot be measured.

ACKNOWLEDGEMENTS

Part of the work described here was performed within the Joint Programme on the Technical Development and Further Improvement of IAEA Safeguards, between the Government of the Federal Republic of Germany and the International Atomic Energy Agency.

REFERENCES

THE IDA-80 MEASUREMENT EVALUATION PROGRAMME ON MASS-SPECTROMETRIC ISOTOPE DILUTION ANALYSIS OF URANIUM AND PLUTONIUM

Some preliminary results

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Abstract

THE IDA-80 MEASUREMENT EVALUATION PROGRAMME ON MASS SPECTROMETRIC ISOTOPE DILUTION ANALYSIS OF URANIUM AND PLUTONIUM: SOME PRELIMINARY RESULTS.

The IDA-80 measurement evaluation programme is concerned with the determination of uranium and plutonium in reprocessing input solutions by mass spectrometric isotope dilution. The participants were requested to report, as part of the programme, the element concentrations and isotope abundances of uranium and plutonium in two sample solutions “B” and “R”. Solution “B” had been prepared by dilution with nitric acid from a sample of original feed solution taken at the WAK reprocessing plant and contained fission products. The reference solution “R” was prepared by CBNM and did not contain fission products. The paper includes graphs of the above requested results. They show the spread of the values reported by the participating laboratories as compared with certified values supplied by CBNM, Geel and NBS, Washington. Results of laboratories claiming to perform isotope dilution analyses frequently or even continuously for more than 5 years, are marked especially. The data presented should be considered as preliminary; they are still to be discussed by participants in the programme.
1. INTRODUCTION

The IDA-80 measurement evaluation programme aims at determining how well the nuclear measurement community can determine uranium and plutonium isotopic compositions and assay uranium and plutonium in reprocessing input solutions by mass spectrometric isotope dilution analysis. Its various objectives and design have been described in detail [1,2].

In one part of the programme the participants were requested to report the element concentrations and isotope abundances of uranium and plutonium in
(a) A sample (called "R") of original feed solution taken at the reprocessing plant and diluted with nitric acid in a ratio of about 1:100. It, therefore, contained fission products.
(b) A sample solution which was prepared synthetically. The composition of this reference solution (called "R") was made similar to that of solution B, but did not contain fission products.

2. SAMPLE CHARACTERIZATION

For both solutions B and R certified values have been jointly established for the uranium and plutonium element concentration as well as for the isotope abundances, by CBNM Geel and NBS Washington, using the best available methods, instruments and measurement procedures.

These data are entered in Table I.

All values have been carefully corrected for isotope fractionation and for known instrumental errors. The uncertainties stated are preliminary and are conservative estimates containing, among others
(a) The relative external standard deviation on a single determination(1s):
   For all isotope dilutions: as determined on 6 independent spikings and subsequent measurements;
   For all isotope abundances: as taken from 6 years of measurements of similar isotope ratios by the same analyst

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1 This programme is jointly carried out by the Central Bureau for Nuclear Measurements (CBNM) at Geel, Belgium and the Karlsruhe Nuclear Research Centre (KfK) of the Federal Republic of Germany under the auspices of the European Safeguards and Development Association (ESARDA) and supported by the International Atomic Energy Agency (IAEA).

### Table I. Certified Values and Preliminary Conservative Estimates of Uncertainties for B and R Solutions

<table>
<thead>
<tr>
<th>Abundances (weight %)</th>
<th>B SOLUTION</th>
<th>R SOLUTION</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Certified value</td>
<td>Uncertainty (in % of value)</td>
</tr>
<tr>
<td>$^{234}\text{U}$</td>
<td>0.0087</td>
<td>$\pm 0.0002$ (+ 2%)</td>
</tr>
<tr>
<td>$^{235}\text{U}$</td>
<td>0.5633</td>
<td>$\pm 0.0014$ (+ 0.25%)</td>
</tr>
<tr>
<td>$^{236}\text{U}$</td>
<td>0.1783</td>
<td>$\pm 0.0006$ (+ 0.33%)</td>
</tr>
<tr>
<td>$^{238}\text{U}$</td>
<td>99.2497</td>
<td>$\pm 0.0022$ (+ 0.002%)</td>
</tr>
<tr>
<td>$^{238}\text{Pu}$</td>
<td>0.2070</td>
<td>$\pm 0.0031$ (+ 1.5%)</td>
</tr>
<tr>
<td>$^{239}\text{Pu}$</td>
<td>69.0631</td>
<td>$\pm 0.070$ (+ 0.13%)</td>
</tr>
<tr>
<td>$^{240}\text{Pu}$</td>
<td>25.6681</td>
<td>$\pm 0.051$ (+ 0.2%)</td>
</tr>
<tr>
<td>$^{241}\text{Pu}$</td>
<td>3.3352</td>
<td>$\pm 0.0080$ (+ 0.25%)</td>
</tr>
<tr>
<td>$^{242}\text{Pu}$</td>
<td>1.7266</td>
<td>$\pm 0.0086$ (+ 0.5%)</td>
</tr>
<tr>
<td>$^{244}\text{Pu}$</td>
<td>&lt;0.0003</td>
<td>&lt;0.0003</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Concentration (mg/g sol.)</th>
<th>B SOLUTION</th>
<th>R SOLUTION</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{238}\text{U}$</td>
<td>2.0492</td>
<td>$\pm 0.0061$ (+ 0.3%)</td>
</tr>
<tr>
<td>$^{239}\text{Pu}$</td>
<td>7.1928</td>
<td>$\pm 0.029$ (+0.4%)</td>
</tr>
</tbody>
</table>

Using the same procedure and (regularly) calibrated instrumentation. In actual fact the IDA samples were measured 3 times, i.e. $n = 3$. Their spread was at least as good as the "6 years" single standard deviation.

(b) The relative external standard deviation ls of the correction factor for isotope fractionation, determined with NBS uranium and plutonium isotopic reference materials over a period of at least one year prior to and during the period
in which the IDA samples were measured (n equal to at least 30).

(c) The uncertainty stated in the Isotopic Reference Materials' Certificates for the major isotope abundance ratios.

In all cases of major isotope abundance ratios and element concentrations the difference between totally independent determinations by CBNM and NBS and the final certified value was smaller than 0.15% relative.

For all minor isotope abundances (i.e. <0.1%) the agreement was better than 1% relative.

3. RESULTS REPORTED BY THE PARTICIPANTS

The 31 laboratories listed in Table II participated in the IDA-80 programme3. In figures 1 to 10 the relative deviations of the reported results from the certified values are presented in sequence of increasing values4.

Three different symbols are used in order to indicate the degree of experience the laboratories had in this type of analysis, according to their own statements made before the performance of the measurements: crossed circles indicate frequent or even continuous analytical work in this field for more than five years; crosses indicate between two and five years; and simple circles indicate less experienced laboratories carrying out such analyses rarely or for less than two years only. In the case of Pu-238 abundances, additional symbols are used (Fig. 11) to indicate whether the determinations were performed by alpha- or by mass-spectrometry.

In order to meet conditions of practical safeguards as nearly as possible, the laboratories were asked to report the results of the element concentration determinations as gram uranium (or plutonium, respectively) per gram sample solution and isotope abundances in weight percent. All the plutonium data were requested to be corrected for radioactive decay to 9 February, 1980, used as a common reference day5. No specific values were recommended for the physical constants needed in the calculations.

3 This list differs from the one published in Ref. [1] because four laboratories withdrew their participation at a later stage.

4 Please note that - as a result of this means of presentation - the "index number" of a laboratory's result differs from figure to figure and that these index numbers have no relation to the code numbers assigned to the participating laboratories in this experiment.

5 This was the day of sampling at the reprocessing plant.
<table>
<thead>
<tr>
<th>IAEA/SAL, Vienna, Austria</th>
<th>CEA/SCQ, Montrouge, France</th>
</tr>
</thead>
<tbody>
<tr>
<td>OFZ, Seibersdorf, Austria</td>
<td>PNC, Tokyo, Japan</td>
</tr>
<tr>
<td>IPEN, São Paulo, Brazil</td>
<td>JAERI, Tokai-Mura, Japan</td>
</tr>
<tr>
<td>SCK/CEN, Mol, Belgium</td>
<td>ECN, Petten, The Netherlands</td>
</tr>
<tr>
<td>NRI, Újez near Prague, CSSR</td>
<td>AWRE, Aldermaston, UK</td>
</tr>
<tr>
<td>RNL, Roskilde, Denmark</td>
<td>AERE, Harwell, UK</td>
</tr>
<tr>
<td>WAK, Karlsruhe, FRG</td>
<td>BNFL, Windscale, UK</td>
</tr>
<tr>
<td>KFA, Jülich, FRG</td>
<td>GE, Pleasanton, CA, USA</td>
</tr>
<tr>
<td>TUI, Karlsruhe, CEC</td>
<td>ANL, Idaho Falls, ID, USA</td>
</tr>
<tr>
<td>KFK/IRCh, Karlsruhe, FRG</td>
<td>ANL, Argonne, IL, USA</td>
</tr>
<tr>
<td>BAM, Berlin (West)</td>
<td>NBL, Argonne, IL, USA</td>
</tr>
<tr>
<td>KWU, Karlstein, FRG</td>
<td>ANL, Argonne, IL, USA</td>
</tr>
<tr>
<td>University of Helsinki, Finland</td>
<td>Westinghouse, Richland, WA, USA</td>
</tr>
<tr>
<td>CEN, Saclay, France</td>
<td>ORNL, Oak Ridge, TN, USA</td>
</tr>
<tr>
<td>COGEMA, Marcoule, France</td>
<td>KRI, Leningrad, USSR</td>
</tr>
</tbody>
</table>

This list differs from the one published in Ref. [1] because four laboratories withdrew their participation at a later stage.

Indication of measurement uncertainties was not requested explicitly but was given by a few laboratories.

In some cases, the laboratories supplied second results corrected for very different reasons like calculation errors, recalibration of working standards or mass discrimination corrections, several weeks or months after the first result but before the final deadline for data acceptance. In such cases, the first result reported is used in the presentation and the second (corrected) one is indicated additionally within brackets.

The number of measurement points entered in the figures vary and are in all cases smaller than the total of 31 laboratories which contributed to IDA-80. This is mainly because five laboratories were unable to complete the whole measurement programme in time, for example because of delayed receipt of samples or budgetary reductions. However, more relevant to the objective of this investigation, is the observation that one

Text continued on p. 277.
FIG. 1. IDA-80, uranium concentrations requested.
FIG. 2. IDA-80, U-234 abundances requested.
FIG. 3. IDA-80, U-235 abundances requested.
FIG. 4. IDA-80, U-236 abundances requested.
FIG. 5. IDA-80, plutonium concentrations requested.
FIG. 6. IDA-80, Pu-238 abundances requested.
FIG. 7. IDA-80, Pu-239 abundances requested.
FIG. 8. IDA-80, Pu-240 abundances requested.
FIG. 9. IDA-80, Pu-241 abundances requested.
**FIG. 10.** IDA-80, Pu-242 abundances requested.
FIG. 11. IDA-80, Pu-238 abundances requested.
laboratory could not calculate concentration values because it used volumetric aliquotation and the organizers did not supply the densities of the sample solutions. Another laboratory reported no concentration value for plutonium because of unsatisfactory reproducibility of the measurements.

As far as the Pu-238 determinations are concerned (Figs. 6 and 11), it should be noted that three participants reported no data for this particular isotope. It was apparently not determined, neither by mass- nor by alpha-spectrometry.

4. CONCLUDING REMARKS

Although the data presented should be considered as preliminary because they must still be discussed and approved by the participants at the meeting, it seems reasonable to conclude that

The uncertainties of uranium measurements are smaller than those of comparable plutonium determinations;
No striking effect of fission products on the accuracy of measurements is apparent;
"Outlier" results of the more experienced laboratories are relatively less frequent but not zero;
It seems that the alpha-spectrometric and the mass-spectrometric methods are both equally suitable for determining Pu-238 abundances.

More detailed results on the whole IDA-80 measurement programme can be expected from the extended data evaluation being carried out at present at KfK. They will be published after discussion on them at the participants' meeting scheduled for autumn, 1983.

ACKNOWLEDGEMENT

The collaboration of WAK Karlsruhe (Mr. R. Berg) consisting of making available the input samples is highly appreciated. Without the hot-cells and the help of Mrs. E. Mainka, Mr. H. Wertenbach and Mr. E. Bolz at KfK-IRCh in the reception and first treatments of the undiluted input samples, this Programme would not have been possible.

The authors gratefully appreciate the excellent co-operation with the participating laboratories which contributed, on the average, about 15 man-weeks each for this Programme. The careful chemical preparation work for the isotopic measurements by

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6In some cases uranium isotope abundances below 0.01% have not been reported.
W. Lycke at CBNM is gratefully acknowledged. Two of the authors (W. Beyrich, W. Golly) are indebted to U. Bicking and W. Rust for their help in establishing the graphs.

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DOSAGE DE TRACES D'URANIUM DANS UNE USINE DE RETRAITEMENT PAR SPECTROFLUORIMETRIE SUR SOLUTION

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Abstract—Résumé

DETERMINATION OF TRACE AMOUNTS OF URANIUM IN A REPROCESSING PLANT BY SOLUTION SPECTROFLUORIMETRY.

To establish inventory tables accurately and satisfy safeguards requirements — and also to ensure satisfactory operation of a reprocessing plant — it is essential to determine the uranium content of numerous solutions where the uranium is present only in trace quantities. For this purpose a method is proposed which relies on the fluorescence of uranyl solutions exposed to ultra-violet radiation. After a brief theoretical summary, the parameters which influence the measurements most strongly are enumerated: medium, temperature, nature of the matrix, and choice of wavelength of the incident radiation. It is then apparent that the measurement must be performed by internal calibration (using the proportional addition method) and that it is useful to obtain a fluorescence spectrum which enables us to verify the presence of uranium. The applications of this method at the La Hague plant are described, where it has been used since October 1981 by shift teams, notably to check the following points: the attack acid (before receiving the fuel), the foot of the first-cycle column, the carbonated solvent washing solution and the solutions of the effluent treatment unit. It is in fact used throughout the plant, even for checking uranium in PuO₂ oxide. The method makes it possible to avoid organic reagents such as pyridine and, in many cases, cumbersome effluent-generating separations. Determinations are possible — to give one example — by simple dilution in the fission-product concentrates and in plutonium solutions where the Pu/U ratio is as great as 1000/1. In pure solutions the detection limit with the equipment used at present is a few µg per litre in the measuring tank. In general, the accuracy is a few per cent.

DOSAGE DE TRACES D'URANIUM DANS UNE USINE DE RETRAITEMENT PAR SPECTROFLUORIMETRIE SUR SOLUTION.

Pour établir précisément les tableaux d'inventaire et satisfaire aux exigences des garanties, comme pour contrôler la bonne marche d'une usine de retraitement, il est nécessaire de déterminer la teneur en uranium dans de nombreuses solutions où il se trouve à l'état de traces.
On propose, à cette fin, une méthode utilisant la propriété que présentent les solutions d'uranyle de fluorescer lorsqu'elles sont irradiées en lumière ultraviolette. Après un bref rappel théorique, on passe en revue les paramètres qui influent le plus sur la mesure: le milieu, la température, la nature de la matrice, le choix de la longueur d'onde de la radiation incidente. On en déduit la nécessité d'effectuer la mesure par étalonnage interne en utilisant la méthode des ajouts dosés et l'intérêt d'obtenir le spectre de fluorescence qui permet d'authentifier la présence d'uranium. On décrit les applications de cette méthode à l'usine de La Hague où elle est utilisée depuis octobre 1981 par des équipes postées, en particulier pour contrôler les points suivants: l'acide d'attaque avant introduction de combustible, le pied de colonne du premier cycle, la solution carbonatée de lavage du solvant, les solutions de l'unité de traitement des effluents. On en déduit la nécessité d'effectuer la mesure par étalonnage interne en utilisant la méthode des ajouts dosés et l'intérêt d'obtenir le spectre de fluorescence qui permet d'authentifier la présence d'uranium. En solution pure, avec l'appareillage utilisé, la limite de détection est de quelques µg·L⁻¹ dans la cuve de mesure. En général, la précision est de quelques pour-cent.

1. INTRODUCTION

L'objectif d'une usine de retraitement est de récupérer d'une part l'uranium pur et, d'autre part, le plutonium pur. En-dehors de son flux principal, l'uranium se trouve à des concentrations relativement faibles, comprises en général entre le mg·L⁻¹ et le g·L⁻¹, dans des matrices diverses et souvent très complexes. Il est cependant nécessaire de mesurer ces concentrations pour contrôler les pertes, compléter et préciser les mesures de routine. Certains de ces résultats d'analyse sont utilisés pour l'établissement des garanties. Or, jusqu'à maintenant, les méthodes utilisables étaient soit lourdes et onéreuses (comme la dilution isotopique ou la fluorimétrie sur pastilles), soit peu satisfaisantes et génératrices d'effluents difficiles à traiter, voire dangereuses par la nature même des réactifs utilisés (colorimétrie au dibenzoyl-méthane en milieu pyridine, par exemple). C'est pourquoi nous avons entrepris, en 1978, une étude inspirée par la publication de Davydov et de ses collaborateurs [1] sur le dosage de l'uranium basé sur la mesure directe de sa fluorescence en solution.

2. RAPPELS THEORIQUES

Eclairée par un flux de photons de longueur d'onde convenablement choisie, une solution d'uranium (VI) est susceptible d'émettre une fluorescence de couleur verte caractéristique. Par absorption des photons incidents, la molécule UO₂(II +) passe du niveau électronique fondamental à un niveau excité d'énergie
supérieure (fig.1a). Cette transition est immédiatement suivie d’un réarrange­ment interne qui ramène la molécule à son plus bas niveau excité (fig.1b). A partir de ce niveau, le retour à l’état fondamental peut s’effectuer par différentes voies dont l’une, qui nous intéresse, est l’émission de photons de fluorescence (fig.1c). Quelle que soit la longueur d’onde excitatrice choisie, pourvu qu’elle se situe dans une bande d’absorption de l’uranium (VI), l’émission des photons de fluorescence correspondra toujours aux mêmes transitions entre les états électroniques 1 et 0. Donc le spectre de fluorescence conservera toujours la même morphologie et sera caractéristique de l’espèce fluorescente.

L’intensité de fluorescence F émise dans tout l’espace sur l’ensemble des longueurs d’onde du spectre d’émission est reliée à l’intensité Ia absorbée par la relation

\[ F = \Phi I_a \]  

(1)

où \( \Phi \) est le rendement de fluorescence, rapport entre les quantités de photons émis et absorbés par l’uranium.

Si l’on appelle \( I_0 \) l’intensité de la radiation excitatrice, I celle de cette même radiation après absorption par l’uranium et \( c \) la concentration de l’uranium, on peut écrire d’après la loi de Beer:

\[ F = \Phi (I_0 - I) = \Phi I_0 (1 - 10^{-k, c}) \]

FIG.1. Schéma de transitions électroniques.
A condition que \( k_1c \), c'est-à-dire l'absorbance, reste très inférieure à 1, l'intensité de fluorescence est directement proportionnelle à la concentration:

\[
F \approx 0,43 \Phi I_0 k_1 c = k \Phi I_0 c \quad (2)
\]

La solution peut contenir des espèces qui favorisent la désexcitation non radiative de l'uranyle et donc diminuent le rendement quantique \( \Phi \). Cet effet d'inhibition de la fluorescence est, dans la plupart des cas, régis par les collisions entre les molécules: il est alors directement proportionnel à la concentration \( [Q] \) de l'espèce inhibitrice. Le rapport entre les rendements quantiques \( \Phi \) en l'absence d'inhibiteur et \( \Phi_i \) en présence de celui-ci est alors donné par la relation de Stern-Volmer

\[
\frac{\Phi}{\Phi_i} = 1 + K [Q] \quad (3)
\]

dans laquelle \( K \) est la constante d'inhibition de Stern-Volmer.

3. PARTIE EXPERIMENTALE

Le spectre de fluorescence étant caractéristique de l'uranium, nous avons choisi de travailler avec un spectrofluorimètre Jobin-Yvon JY3C équipé de deux réseaux permettant de sélectionner et de faire varier aussi bien les longueurs d'onde d'excitation que celles d'émission et de tracer les spectres correspondants, et d'un photomultiplicateur Hamamatsu 928. La source lumineuse est une lampe au xénon. Nous utilisons en routine une lampe à enveloppe de verre dite «sans ozone», pour éviter au maximum la formation de ce gaz nocif pour l'homme et responsable du vieillissement prématuré des gants de boîtes, en particulier. Le compartiment-cuve a été remplacé soit par un puits fixé de façon étanche à la boîte à gants, soit par un puits blindé muni d'une cuve à circulation reliés à une enceinte blindée.

4. CHOIX DES CONDITIONS DE MESURE

4.1. Choix du milieu

Le but de cette étude était la mise au point d'une méthode utilisable dans une usine de retraitement, donc pour des solutions en général nitriques contenant souvent des cations très hydrolysables comme Pu\(^{4+}\). Nous avons donc recherché un mode opératoire en milieu acide, nécessitant en outre l'addition du minimum de réactifs les plus simples possible.
4.1.1. Milieu nitrique

L’intérêt d’opérer dans ce milieu est évidemment d’éviter tout ajout de réactif et, en conséquence, d’une part de résoudre les problèmes d’effluents analytiques, d’autre part d’ouvrir des perspectives sur l’utilisation de la méthode en-ligne.

L’acide nitrique absorbe dans l’UV, ce qui limite considérablement le choix des longueurs d’onde d’excitation. En outre, il a une influence néfaste sur le rendement de fluorescence. Le dosage est cependant possible, même dans des solutions relativement concentrées en acide nitrique: 3 à 4 M, mais la sensibilité est assez limitée. En solution pure, le domaine d’utilisation de la méthode s’étend de 5 mg·L⁻¹ à environ 1 g·L⁻¹.

4.1.2. Milieu phosphorique

Sill et Peterson [2] avaient déjà montré que ce milieu était favorable à la fluorescence de l’ion uranyle en solution. Davydov et ses collaborateurs [1] préconisent d’effectuer le dosage dans un milieu contenant 5% d’acide phosphorique, ce qui correspond à la concentration 0,75 M. Ils ont montré et nous avons vérifié que la fluorescence d’une solution nitrique d’uranium augmente beaucoup avec l’accroissement de la concentration d’acide phosphorique jusqu’à cette valeur, mais reste inchangée au-delà de cette valeur. La sensibilité dépend du choix des longueurs d’onde et de la concentration finale en acide nitrique, mais à conditions opératoires identiques elle est multipliée par l’addition de l’acide phosphorique par un facteur qui peut atteindre 200. Avec une concentration finale en acide nitrique voisine de 0,25 M, le dosage est possible en solution pure entre 20 µg·L⁻¹ et 200 mg·L⁻¹.

4.1.3. Milieu sulfurique

Dès 1947, plusieurs équipes [2, 3] ont proposé de doser l’uranium en milieu sulfurique en utilisant la mesure de luminescence. Nous avons constaté, non seulement que ce milieu est beaucoup plus favorable que l’acide nitrique, mais que lorsque l’on ajoute de l’acide sulfurique à une solution nitrique d’uranium, la fluorescence augmente avec la concentration d’acide sulfurique sans qu’apparaissa de limitation à cette influence, au moins jusqu’à 9 M. (fig.2). Pour cette concentration et à conditions opératoires identiques, la sensibilité est environ deux fois moindre qu’en milieu phosphorique 0,75 M.

4.2. Choix des longueurs d’onde

Quelle que soit la longueur d’onde d’excitation, la spectre de fluorescence conserve la même morphologie. La position des pics varie également peu avec
FIG. 2. Influence de la concentration en H$_2$SO$_4$ sur la fluorescence de l'uranium. 
Milieu HNO$_3$ 1,25 M - H$_2$SO$_4$ (1) = 9 M, (2) = 4,5 M, (3) = 2,2 M, (4) = 1,1 M, 
(5) = 0,5 M, (6) = 0,25 M.

le milieu (quelques nm au maximum) du moins si l'on se limite aux milieux considérés ici. Il apparaît nettement 4 pics dont les plus importants sont situés à environ 494 et 513 nm. L'obtention de ce spectre est une véritable garantie de la présence d'uranium.

Le choix de la longueur d'onde d'excitation est plus complexe, car il dépend beaucoup du milieu et de l'appareillage. La relation (1) montre qu'il faut qu'il y ait absorption par l'uranyle de la radiation incidente. Or, si les coefficients
FIG. 3. Spectres d’absorption de:
(a) $= U (VI) 1 \text{g.L}^{-1}$ en milieu $H_3PO_4 0.75 \text{M}$
(b) $= HNO_3 0.25 \text{M}$
(c) $= H_3PO_4 0.75 \text{M}$
Spectre d’excitation d’une solution d’uranium à 100 mg.L$^{-1}$ en milieu $H_3PO_4 0.75 \text{M}$,
$HNO_3 2.5 \text{M}$ (d).

D’extinction molaire de cette espèce dépendent des milieux que nous avons examinés, la forme et la position de ses bandes d’absorption changent peu.
L’excitation devra être effectuée avec une radiation de longueur d’onde soit voisine de 417 nm, soit inférieure à 345 nm (fig.3a). La relation (2) montre que si l’on désire augmenter la sensibilité, on devra choisir une radiation de plus courte longueur d’onde. Dans le cas de plus grandes concentrations, on se rapprochera du visible pour accroître le domaine de linéarité ($k_{1c}$ doit rester petit). En présence d’acide nitrique, le choix est plus limité parce que son
absorbance propre entre en compétition avec celle de l’uranium et aux plus fortes concentrations d’acide nitrique (ou de nitrate) seule est disponible la bande d’absorption voisine de 417 nm. Pour des concentrations comprises entre 0,1 et 3 M, il est possible d’obtenir 3 pics dans le spectre d’excitation (fig.3d), le pic situé vers 260 nm correspondant à un minimum dans le spectre d’absorption du nitrate et la vallée située vers 300 nm à un maximum dans ce même spectre.

La sensibilité de la méthode dépend de l’intensité de la radiation excitatrice \( I_0 \): le choix de la longueur d’onde doit donc tenir compte de l’appareillage et en particulier du spectre d’émission de la lampe source. Ainsi, la lampe dite «sans ozone» a un spectre d’émission qui décroît très vite avec les longueurs d’onde au-dessous de 300 nm. En l’absence d’acide nitrique en concentration importante, le meilleur compromis pour avoir la sensibilité maximale, avec notre appareillage, se situe au voisinage de 290 nm.

Ce choix des longueurs d’onde d’excitation et de mesure de fluorescence doit rester très empirique, car il dépend de la solution à analyser (éléments tiers absorbants ou inhibiteurs) et de nombreux autres paramètres comme:
- l’interférence du deuxième ordre à travers l’optique de mesure, due à la diffusion de la radiation incidente de courte longueur d’onde;
- l’émission Raman de l’eau, d’autant plus gênante que la radiation d’excitation a une longueur d’onde plus voisine de celle de la mesure;
- l’interférence éventuelle de l’appareil lui-même dont certains éléments (verre par exemple) peuvent fluorescer. Nous avons eu ainsi la surprise de constater que la charge mise par le fabricant pour opacifier certaines parties d’une cuve en quartz, sous une irradiation de 290 nm, était capable d’émettre un rayonnement de fluorescence dans la zone 470–530 nm.

4.3. Influence de la température

La sensibilité de la méthode diminue de 3,5 à 5% par degré au voisinage de 20°C dans les milieux que nous avons étudiés. Or, la lampe source échauffe le compartiment-cuve. Il est donc nécessaire de thermoréguler la cuve de mesure si l’on veut obtenir un signal stable. Il serait intéressant de refroidir la cuve jusqu’au voisinage de 0°C pour augmenter la sensibilité; cependant, cela présente un double inconvénient: il y a risque de condensation de vapeur d’eau sur les parois de la cuve et, par ailleurs, l’équilibre thermique est plus long à obtenir après l’introduction de la solution. Nous préférons donc thermoréguler au voisinage de la température de la pièce, c’est-à-dire à 20°C.

4.4. Influence de la matrice

Non seulement le milieu mais aussi les éléments accompagnateurs peuvent avoir une grande influence sur le signal. Certains ions ou molécules peuvent
eu mêmes fluorescer: c'est le cas, par exemple, du magnésium. D'autres peuvent exalter ou, plus souvent, diminuer le signal de fluorescence. Les raisons peuvent être nombreuses: absorption de la radiation d'excitation ou de la radiation de fluorescence, complexation, réaction chimique ou physicochimique avec l'ion uranyle excité... Par exemple, l'ion $\text{Cl}^-$ diminue la fluorescence dès que sa concentration avoisine $10^{-5} \text{ M}$. Il en est de même avec l'acide nitrique. Les effets de ces éléments dépendent de façon très générale beaucoup plus de leur propre concentration (relation 3) que du rapport de celle-ci à la concentration de l'uranium. Autrement dit, à condition que la concentration de ces éléments perturbateurs ne soit pas trop importante, la mesure reste possible car le signal de fluorescence demeure proportionnel à la concentration de l'uranium. En revanche, comme la sensibilité varie beaucoup avec la matrice, comme aussi pour une même matrice elle peut varier d'un jour à l'autre suivant la manière dont s'est amorcé l'arc de la lampe par exemple, il est absolument nécessaire d'effectuer un étalonnage interne en utilisant la méthode des ajouts connus. Il est également nécessaire d'effectuer la mesure non à longueur d'onde fixe mais en enregistrant le spectre de fluorescence: on peut ainsi évaluer le «blanc» de fluorescence, c'est-à-dire la contribution de la lumière parasite et de la fluorescence des constituants de la matrice autres que l'uranium.

5. APPLICATION EN USINE DE RETRAITEMENT

5.1. Pied de colonne du premier cycle

La première application recherchée pour l'usine de retraitement visait le dosage de l'uranium dans le pied de colonne du premier cycle, c'est-à-dire la détermination de concentrations faibles d'uranium (au niveau de la dizaine de mg·L$^{-1}$) en présence de l'ensemble des produits de fission à la concentration totale de quelques g·L$^{-1}$. C'est en effet à ce niveau (ou à celui des concentrats de produits de fission, ce qui revient au même du point de vue de l'analyste) que le problème analytique était, jusque là, le moins bien résolu.

Les essais effectués sur des solutions synthétiques de produits de fission ont montré que l'effet combiné de l'acide nitrique et des produits de fission rendait impossible le dosage par irradiation directe de la solution. L'ajout d'acide phosphorique provoque la formation d'un précipité renfermant en particulier du zirconium et du molybdène. La dilution de la prise d'essai dans de l'acide sulfurique 9 M s'est révélée tout à fait favorable: dès le premier essai à l'usine de La Hague, il a été possible, non seulement de détecter mais également de déterminer, par la méthode des ajouts connus, la concentration de l'uranium. Cette façon d'opérer n'a cependant pas été retenue par la suite parce que l'acide sulfurique, à cette concentration, attaque l'acier inoxydable de la canalisation de
FIG. 4. Dosage de l'uranium dans un concentrat de produits de fission, après dilution 1/100 par H₂SO₄ 0,05 M, puis 1/10 par H₃PO₄ 0,75 M.
A: sans ajout;
B: avec ajouts d'uranium à la concentration finale de 0,2 mg·L⁻¹;
C: avec ajouts d'uranium à la concentration finale de 0,4 mg·L⁻¹.

...remplissage de la cuve à circulation avec un dégagement gazeux qui perturbe la mesure et provoque même la vidange de la cuve.

Le mode opératoire finalement retenu est le suivant: la prise d'essai est d'abord diluée dans de l'acide sulfurique dilué (0,05 M), puis des aliquotes de cette dilution additionnées ou non de quantités connues d'uranium sont de nouveau diluées en milieu phosphorique 0,75 M. La première dilution permet d'éviter la précipitation des produits de fission par l'acide phosphorique. Elle présente l'intérêt d'abaisser la concentration de l'acide nitrique tout en restant suffisamment acide pour éviter l'hydrolyse des cations présents et, en outre,
de disposer à partir d'un unique prélèvement d'échantillon, de suffisamment de solution pour effectuer des mesures avec et sans ajout d'uranium (le remplissage de la cuve à circulation nécessite à chaque fois environ 25 ml de solution).

La sensibilité est telle dans le milieu final de mesure (on peut doser 2 μg·L⁻¹ en l'absence de «quenching») que l'on peut effectuer des dilutions importantes de façon à diminuer considérablement l'effet inhibiteur des produits de fission (relation 3). On peut ainsi doser sans difficulté l'uranium au niveau de quelques mg·L⁻¹ dans les pieds de colonne du premier cycle (fig.4). La méthode est bien évidemment applicable aux pieds de colonne des autres cycles.

5.2. Solutions carbonatées de lavage du solvant

On utilise le même mode opératoire. Dans ce cas, la première dilution par l'acide sulfurique dilué a pour effet d'acidifier la prise d'essai et de chasser les ions carbonates.

5.3. Solutions chargées en plutonium

Le plutonium (IV) a, dans les milieux que nous avons étudiés, un effet inhibiteur sur la fluorescence de l'uranium. A concentration égale en plutonium, cet effet est un peu moins prononcé en présence d'acide phosphorique qu'en milieu nitrique seul. En outre, le gain en sensibilité obtenu permet d'effectuer avec l'acide phosphorique des dilutions plus importantes. Cependant, dès que la concentration du plutonium devient trop élevée (de l'ordre de 1 g·L⁻¹), il y a formation de précipité lors de l'ajout d'acide phosphorique. Dans ce cas encore, une pré-dilution avec de l'acide sulfurique dilué est favorable. Il est ainsi possible d'effectuer le dosage sans séparation jusqu'à des rapports Pu/U de l'ordre de 10⁴ (fig.5).

En général, dans l'usine, les solutions de plutonium (IV) contiennent de l'acide nitreux, puisqu'il y a eu ajustage de valence. Or, celui-ci peut être génant: au niveau de 2·10⁻⁴ M, il diminue d'un facteur supérieur à 2 le signal de fluorescence. On peut l'éliminer par ajout d'acide sulfamique, mais en évitant que l'excès de ce dernier réactif dépasse 10⁻² M dans la cellule de mesure, car il est lui-même inhibiteur.

Nous n'avons pas encore étudié concrètement le dosage en présence de plutonium (III). En général, les solutions de plutonium (III) contiennent de l'hydrazinium en concentration importante (entre 0,1 et 1 M). Cet ion précipite lors de l'ajout d'acide sulfurique. En outre, l'ion UO₂⁺ excité par les photons est réductible par l'hydrazinium, ce qui a pour effet de diminuer le temps de vie et donc le signal de fluorescence. De plus, celui-ci risque de décroître avec le temps puisque l'uranium (IV) formé n'est pas fluorescent. Il est nécessaire, par conséquent, de détruire l'hydrazinium. Pour ce faire, nous n'avons pas trouvé à ce jour
FIG. 5. Dosage d'uranium dans une solution de plutonium à 2,15 g·L⁻¹ après dilution 4/25 par H₂SO₄ 0,05 M puis 1/50 par H₃PO₄ 0,75 M
– sans ajout (3)
– avec ajouts d'uranium à la concentration finale de 10 µg·L⁻¹ (2), 20 µg·L⁻¹ (1).

de méthode vraiment satisfaisante. Nous préconisons l’oxydation par ajout d’un léger excès de nitrite qui sera ensuite détruit par l’acide sulfamique dont l’excès sera réduit au minimum possible.

5.4. Oxyde de plutonium

Dans l’oxyde de plutonium qui sort de l’usine, comme d’ailleurs dans les solutions riches en plutonium de fin de purification, par exemple juste avant la précipitation oxalique, la teneur en uranium est trop faible pour que l’on puisse éviter la séparation de la majeure partie du plutonium.
Une méthode classique consiste à effectuer cette séparation sur résine anionique en milieu chlorhydrique 8 M en présence de chlorhydrate d’hydroxylamine. L’uranium (VI) fixé est ensuite élué par l’acide chlorhydrique 0,5 M. L’ion Cl⁻ étant très gênant, il est nécessaire de le chasser par traitement par l’acide sulfurique à fumées blanches. La prise d’essai est enfin reprise, soit par de l’acide sulfurique 9 M, soit par de l’acide phosphorique 0,75 M.

L’utilisation d’acide chlorhydrique est à éviter en usine pour des raisons de corrosion et de traitement d’effluents. Aussi a-t-il été proposé d’utiliser la propriété du plutonium (IV) de précipiter avec l’acide phosphorique. L’oxyde est donc dissous par de l’acide sulfurique additionné d’un peu d’acide fluorhydrique. L’addition ensuite d’acide phosphorique 0,75 M à la prise d’essai provoque la formation d’un précipité qui est filtré sur verre fritté. L’uranium est dosé par fluorimétrie sur la solution. Des essais synthétiques, ainsi que l’utilisation en parallèle des deux méthodes ci-dessus sur des échantillons de PuO₂ ont montré que l’uranium (VI) restait bien en solution. Par prudence, la prise d’essai est, avant précipitation par l’acide phosphorique, oxydée par ajout d’acide nitrique et de quelques millilitres d’une solution 10⁻²N de bichromate de potassium. Il semblerait, en effet, que l’uranium puisse exister à l’état tétravalent dans le milieu de dissolution, soit en raison d’une oxydation incomplète lors de la dissolution, soit par suite d’une photoréduction comme il s’en produit, à la lumière du jour, dans les solutions d’uranium en milieu phosphorique.

6. CONCLUSIONS

Cette méthode de dosage s’avère aussi sensible que la fluorimétrie sur pastilles et beaucoup moins lourde à mettre en œuvre. Moins sensible aux ions étrangers, elle permet d’éviter souvent des séparations longues et génératrices d’effluents: elle est ainsi directement utilisable pour des solutions de plutonium dans lesquelles le rapport Pu/U est <10³. Elle convient particulièrement au dosage de l’uranium dans les solutions de pied de colonne du premier cycle ou dans les concentrats de produits de fission, pour lesquels les manipulations sont réduites à de simples dilutions et ajouts de solutions de référence.

Aussi est-elle utilisée depuis octobre 1981, en routine, à l’usine de la Hague par les équipes postées en 5 X 8h. Elle a d’abord été mise en œuvre pour contrôler les points suivants:

– l’acide d’attaque qui a rincé les coques et dilué le fond de cuve de dissolution avant l’introduction d’une nouvelle charge de combustible: cela constitue le point zéro de la dissolution;
– le pied de colonne du premier cycle et le concentrat de produits de fission;
– la solution carbonatée du lavage du solvant;
– les solutions de l’unité de traitement des effluents.
Elle s’est ensuite étendue au contrôle de l’oxyde PuO₂ et des solutions riches en plutonium et son emploi se généralise à tous les contrôles d’uranium en dehors de son flux principal. Des études sont en cours pour l’appliquer au contrôle des phases organiques.

La limite de détection en solution pure est de quelques μg·L⁻¹ dans la cuve de mesure. La précision obtenue est, dans les cas favorables, de quelques pour-cent, ce qui est largement suffisant pour les contrôles de traces, pour lesquels l’obtention du spectre caractéristique permet d’éviter de graves erreurs d’interprétation.

Nous poursuivons des études pour nous affranchir de certaines limitations telles que la diffusion, le deuxième ordre ou la fluorescence propre des autres espèces présentes, dans le but non de pouvoir opérer avec des solutions plus diluées (les risques de pollution deviendraient trop importants), mais afin de faciliter l’interprétation des spectres et éviter, si possible, toute séparation même pour le contrôle de l’oxyde PuO₂. A cette fin, nous utilisons, comme source d’excitation, un laser pulsé et une électronique permettant l’obtention du spectre de fluorescence retardée, les états excités de l’uranium ayant un temps de vie moyen exceptionnellement long.

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UTILISATION D’UN ETALON INTERNE CONSTITUE D’UN MELANGE D’ISOTOPES 233 ET 236 DE L’URANIUM POUR AMELIORER LA JUSTESSE DES ANALYSES ISOTOPIQUES ET CELLE DES DOSAGES PAR DILUTION ISOTOPIQUE DE L’URANIUM.

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Abstract—Résumé

USE OF AN INTERNAL STANDARD CONSISTING OF A MIXTURE OF THE ISOTOPES 233U AND 236U TO IMPROVE THE ACCURACY OF ISOTOPIC ANALYSES AND ISOTOPIC DILUTION DETERMINATIONS OF URANIUM.

Isotopic dilution associated with thermoionization mass spectrometry is one of the techniques used to obtain accurate measurements of the uranium contained in irradiated fuel solutions. The accuracy of the results is directly dependent in particular on the accuracy of calibration of the tracer (233U) solution employed. This is normally calibrated by isotopic dilution on the basis of natural metallic uranium of guaranteed purity, and the measurement which limits the accuracy of the calibration is that of the 233/238 ratio, the value of which may be affected by a systematic error due to imperfect correction of isotopic fractionation. To correct the effects of isotopic fractionation which limit the accuracy of isotopic analyses, a method has been developed which uses a 233U/236U mixture as a standard. By mixing the internal reference standard with isotopic standards having certified 235U/238U ratios, it is possible to determine the 233U/238U ratio by internal normalization to a value whose accuracy is the same as that of the standards utilized. This internal standard is added to the sample to be analysed, and the measured 235U/238U ratios are normalized to the value previously determined; the same normalization factor is applied to the measured 235U/238U ratios, which are then known with the same accuracy as the isotopic standards. The same method can be applied to isotopic dilution measurements of uranium concentrations by using a double 233U/236U tracer instead of the single tracer at present used. The isotopic fractionation effect in the measured 233U/238U ratios can then be corrected with precision because it is known from a comparison of the measured 233U/236U ratios with the previously determined theoretical 233U/236U value. A comparison of results obtained by the conventional method and by the method proposed here is presented. Broader utilization of this method for standardization of tracer solutions and of measurements performed on real samples would undoubtedly eliminate or at least substantially limit both accidental and systematic errors.
UTILISATION D'UN ÉTALON INTERNE CONSTITUÉ D'UN MÉLANGE D'ISOTOPIES 233 ET 236 DE L'URANIUM POUR AMÉLIORER LA JUSTESSE DES ANALYSES ISOTOPIQUES ET CELLE DES DOSAGES PAR DILUTION ISOTOPIQUE DE L'URANIUM.

La dilution isotopique associée à la spectrométrie de masse à thermoionisation est l'une des techniques utilisées pour la mesure précise des quantités d'uranium contenues dans les solutions de combustibles irradiés. La justesse des résultats obtenus est, en particulier, directement liée à la justesse de l'étalonnage de la solution de traceur (uranium 233) utilisée. Cette dernière est généralement établie par dilution isotopique à partir d'uranium métallique naturel de pureté garantie, et la mesure qui limite la justesse de cet étalonnage est celle du rapport 233/238, dont la valeur peut-être entachée d'une erreur systématique due à une correction imparfaite du fractionnement isotopique. Afin de corriger les effets du fractionnement isotopique qui limitent la précision des analyses isotopiques, une méthode basée sur l'utilisation d'un mélange éton 233U/236U a été mise au point. En mélangeant l'étalon interne à des standards isotopiques de rapports 235/238 certifiés, il est possible de déterminer le rapport 233/236 par normalisation interne à une valeur dont la justesse est celle des standards utilisés. Cet étalon interne est ajouté à l'échantillon à analyser et les rapports 233/236 mesurés sont normalisés à la valeur précédemment déterminée et le même facteur de normalisation est appliqué aux rapports 235/238 mesurés. Ces derniers rapports sont alors connus avec la même justesse que celle des standards isotopiques. La même méthode peut être appliquée à la mesure des concentrations d'uranium par dilution isotopique en employant un traceur double 233/236 à la place du traceur simple actuellement utilisé. Le fractionnement isotopique sur les rapports mesurés 233/236 peut alors être corrigé avec précision car il est connu à partir de la comparaison des rapports 233/236 théorique déterminée précédemment. Une comparaison des résultats obtenus par la méthode classique et par la méthode proposée ici sera présentée. L'utilisation généralisée de cette méthode pour l'étalonnage des solutions de traceur et les mesures effectuées sur les échantillons réels permettrait certainement de supprimer ou au moins de limiter de manière très appréciable les écarts de justesse qu'ils soient accidentels ou systématiques.

La dilution isotopique associée à la spectrométrie de masse à thermoionisation est la technique la plus fréquemment utilisée pour la mesure précise des concentrations en uranium des solutions de combustibles irradiés. La justesse des résultats obtenus est, en particulier, directement liée à la justesse de l'étalonnage de la solution traceur (uranium 233) utilisée. Cette dernière est généralement établie par dilution isotopique à partir d'uranium métallique naturel de pureté garantie, la justesse de cet étalonnage est limitée, pour la plus grande part, par la mesure du rapport 233/238, dont la valeur peut être entachée d'une erreur systématique due à une correction imparfaite du fractionnement isotopique.

Ce fractionnement, qui est proportionnel à l'écart de masse, se traduit par une variation continue au cours du temps des rapports isotopiques mesurés. Il dépend de nombreux paramètres (quantité d'uranium déposée, forme chimique du
TABLEAU I. REPRODUCTIBILITE DES RESULTATS D'ANALYSES EN UTILISANT UN ETALON INTERNE

<table>
<thead>
<tr>
<th></th>
<th>235U/238U ECHANTILLON N° 1</th>
<th>235U/238U ECHANTILLON N° 2</th>
<th>235U/238U ECHANTILLON N° 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>1er dépôt</td>
<td>0,040430 ± 0,000007 (2σ)</td>
<td>0,038183 ± 0,000011 (2σ)</td>
<td>0,040385 ± 0,000010 (2σ)</td>
</tr>
<tr>
<td>2ème dépôt</td>
<td>0,040434 ± 0,000008 (2σ)</td>
<td>0,038186 ± 0,000009 (2σ)</td>
<td>0,040383 ± 0,000010 (2σ)</td>
</tr>
</tbody>
</table>

dépôt, température du filament porteur du dépôt, durée de l'analyse) dont le contrôle rigoureux est difficile à obtenir en routine.

Une solution consiste à corriger les effets de ce fractionnement isotopique, en utilisant un étalon interne [1,2,3]. Mise au point au laboratoire en 1978, cette méthode a été appliquée à la mesure des teneurs en 235U d'une série d'échantillons [4]: elle a permis d'atteindre un accord voisin de 10⁻⁴ entre plusieurs analyses de dépôts différents des mêmes échantillons (Tableau I).

1. PRINCIPE D'UTILISATION D'UN ETALON INTERNE

On ajoute à l'échantillon de rapport 235U/238U inconnu, un étalon interne constitué par une solution d'uranium 233 et 236, dont le rapport 233/236 a été préalablement déterminé avec précision par référence à des standards isotopiques. Les rapports 233/236, mesurés sur le mélange sont normalisés à la valeur théorique et le même facteur de normalisation est appliqué aux rapports 235/238 mesurés.

Pour appliquer la méthode de l'étalon interne à la mesure des concentrations d'uranium par dilution isotopique, il suffit d'employer un traceur double 233/236 à la place du
traceur simple actuellement utilisé. Le rapport 233/236 du traceur est déterminé à partir de mélanges avec des standards isotopiques, alors que sa teneur en $^{233}U$ est calculée à partir d'un étalonnage avec de l'uranium métallique naturel de pureté certifiée. Le fractionnement isotopique sur les rapports 233/238 peut être corrigé avec précision, car il se déduit de la comparaison entre les valeurs mesurées des rapports 233/236 et la valeur 233/236 théorique déterminée précédemment.

2. PREPARATION DU TRACEUR DOUBLE

La réalisation d'un traceur double implique la disponibilité de quelques dizaines de mg d'uranium 233 et 236 de grande pureté isotopique.

Pour cette étude, le traceur double a été préparé à partir de solutions nitriques d'uranium enrichi en 236 contenant 9,3% de $^{235}U$ et 1,3% de $^{238}U$, (*) et d'uranium 233 très enrichi ( > 99,5%) dans des proportions telles que le rapport $^{236}U/^{233}U$ de la solution finale soit voisin de 1. Afin d'assurer l'homogénéisation isotopique, le mélange a été évaporé à sec, repris deux fois par de l'acide nitrique concentré très pur, puis évaporé de nouveau à sec. La reprise finale du résidu a été effectuée par de l'acide nitrique molaire pour obtenir une concentration de l'ordre de 4 mg/ml.

Une aliquote de 1 ml de cette solution mère a été évaporée à sec ; la reprise du résidu a été faite par 2 ml de HNO$_3$0,2M. Cette solution diluée de traceur double a été utilisée pour toutes les déterminations de compositions isotopiques.

Les compositions isotopiques mesurées des solutions initiales d'uranium 233 et 236 et du traceur double sont données dans le tableau II.

3. DETERMINATION DE LA VALEUR ABSOLUE DU RAPPORT 233/236 DU TRACEUR DOUBLE.

3.1. Préparation et analyse des mélanges

Différentes aliquotes de solution diluée de traceur, contenant environ 2 ug d'uranium, ont été mélangées à des

(*) L'uranium 236 de grande pureté isotopique ($^{236}U > 99,5\%$) n'est pas disponible actuellement en quantité suffisante sur le marché international.
TABLEAU II. COMPOSITIONS ISOTOPIQUES, EN ATOMES %, DES SOLUTIONS INITIALES ET DU TRACEUR DOUBLE

<table>
<thead>
<tr>
<th></th>
<th>$^{233}\text{U}$</th>
<th>$^{234}\text{U}$</th>
<th>$^{235}\text{U}$</th>
<th>$^{236}\text{U}$</th>
<th>$^{238}\text{U}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Solution initiale $^{233}\text{U}$</td>
<td>99,549</td>
<td>0,430</td>
<td>0,0027</td>
<td>non décelé</td>
<td>0,0183</td>
</tr>
<tr>
<td>Solution initiale $^{236}\text{U}$</td>
<td>&lt;0,001</td>
<td>0,118</td>
<td>9,305</td>
<td>89,277</td>
<td>1,300</td>
</tr>
<tr>
<td>Traceur double</td>
<td>49,731</td>
<td>0,274</td>
<td>4,658</td>
<td>44,670</td>
<td>0,667</td>
</tr>
</tbody>
</table>

TABLEAU III. COMPOSITIONS ISOTOPIQUES, EN ATOMES %, DES STANDARDS NBS UTILISES

<table>
<thead>
<tr>
<th></th>
<th>$^{234}\text{U}$</th>
<th>$^{235}\text{U}$</th>
<th>$^{236}\text{U}$</th>
<th>$^{238}\text{U}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>NBS U-200</td>
<td>0,1246 ± 0,0003</td>
<td>20,013 ± 0,020</td>
<td>0,2116 ± 0,0006</td>
<td>79,651 ± 0,020</td>
</tr>
<tr>
<td>NBS U-500</td>
<td>0,5181 ± 0,0008</td>
<td>49,696 ± 0,050</td>
<td>0,0755 ± 0,0003</td>
<td>49,711 ± 0,050</td>
</tr>
<tr>
<td>NBS U-850</td>
<td>0,6437 ± 0,0014</td>
<td>85,137 ± 0,017</td>
<td>0,3704 ± 0,0011</td>
<td>13,848 ± 0,014</td>
</tr>
</tbody>
</table>
aliquotes de solutions des standards NBS U 500, U 850 et U 200 dont les compositions isotopiques certifiées sont rappelées tableau III.

Un traitement chimique analogue à celui décrit précédemment a été appliqué à ces mélanges pour assurer l'homogénéisation isotopique.

Des dépôts de ces mélanges ont été réalisés sur les filaments latéraux en tant que d'un assemblage à trois filaments (le filament central étant en rhénium) préalablement dégazé pendant une heure.

Les analyses isotopiques ont été faites avec un spectromètre de masse CAMECA TSN 206, équipé d'un barillet et associé à un calculateur Hewlett-Packard 9825 A, qui permet la réalisation des analyses en mode automatique.

Pour chaque dépôt de ces mélanges, des séquences successives de 10 cycles de mesures des rapports suivants :

\[(233/236)_M \text{ noté (3/6)}_M; \quad (235/236)_M \text{ noté (5/6)}_M; \quad (238/236)_M \text{ noté (8/6)}_M\]

ont été enregistrées.

On disposait par ailleurs des rapports certifiés de chacun des standards NBS :

\[(235/238)_{St} \text{ noté (5/8)}_{St} \quad \text{et} \quad (236/238)_{St} \text{ noté (6/8)}_{St} \]

et des rapports mesurés suivants :

\[(233/236)_T \text{ noté (3/6)}_T; \quad (235/236)_T \text{ noté (5/6)}_T; \quad (238/236)_T \text{ noté (8/6)}_T, \]

déduits de l'analyse isotopique du traceur double (cf. Tableau II).

3.2. Principe du calcul

Pour étalonner le traceur double il est nécessaire de faire la part dans l'analyse du mélange des isotopes 233 et 236 provenant du traceur, et celle des isotopes 235 et 238 provenant du standard afin de pouvoir normaliser les rapports \((3/6)_T\) aux rapports \((5/8)_{St}\). Dans la mesure où l'on ne dispose pas d'un uranium 236 très enrichi, les corrections à faire sont importantes, mais elles peuvent être faites avec une grande précision par la méthode de calcul suivante:

Ce calcul est conduit en trois étapes successives :

1 - Les valeurs brutes mesurées \((3/6)_T\) et \((8/6)_T\) sont utilisées pour corriger les rapports mesurés sur le mélange, afin d'obtenir une première estimation de la valeur \((5/8)_{St}\).

La comparaison de cette valeur avec la valeur certifiée du rapport \((5/8)_{St}\) permet de déduire une première
approximation du fractionnement isotopique qui s'applique à l'analyse du mélange.

Ce fractionnement isotopique est utilisé pour normaliser les rapports \((3/6)_M, (5/6)_M\) et \((8/6)_M\). Ces valeurs normalisées sont substituées dans le calcul précédent aux valeurs initiales, ce qui permet de déduire une nouvelle valeur du fractionnement isotopique.

Ce calcul est reproduit plusieurs fois jusqu'à convergence du fractionnement isotopique à une valeur constante.

2 - Le fractionnement isotopique déterminé à la fin de l'étape précédente permet de corriger la composition isotopique du mélange. De cette composition corrigée on soustrait celle du standard pour obtenir une première valeur corrigée de la composition isotopique du traceur. Cette composition isotopique du traceur, corrigée, est alors substituée à la composition isotopique brute utilisée dans l'étape 1 et les calculs de cette étape sont repris jusqu'à convergence du fractionnement isotopique à une valeur constante. Puis les étapes 1 et 2 sont successivement enchaînées jusqu'à obtenir à la fois une valeur constante du fractionnement isotopique pour l'analyse du mélange et une valeur constante du rapport \((3/6)_T\).

3 - La valeur du rapport \((3/6)_T\) ainsi déterminée est utilisée pour normaliser les rapports \((5/6)_T\) et \((8/6)_T\) mesurés lors de l'analyse isotopique du traceur et les étapes 1 et 2 sont recommencées avec ces nouvelles valeurs de la composition isotopique du traceur. Les calculs par itérations successives des étapes 1, 2 et 3 sont alors poursuivis jusqu'à convergence du rapport \((3/6)_T\) à une valeur constante qui est la valeur absolue recherchée.

Ces calculs sont effectués pour chaque séquence de 10 cycles de mesures des rapports \((3/6)_M, (5/6)_M\) et \((8/6)_M\).

3.3. Résultats de l'étalonnage

Quatre mélanges de solution de traceur double et de standards NBS ont été réalisés dans les conditions suivantes :
- Mélange I : U 500/Traceur # 1 ;
- Mélange II : U 500/Traceur # 5 ;
- Mélange III : U 850/Traceur # 5 ;
- Mélange IV : U 200/Traceur # 5.
Ces conditions ont été choisies de manière à faire varier, pour ces différents mélanges, les contributions en isotopes 235 et 238 provenant du traceur ; ainsi selon les mélanges les rapports \((235)_T/(235)_M\) varient de 1 à 8,6%, et les rapports \((238)_T/(238)_M\) varient de 0,3 à 1,3%.
TABLEAU IV. DÉTERMINATION DE LA VALEUR ABSOLUE DU RAPPORT 233/236 DU TRACEUR DOUBLE

<table>
<thead>
<tr>
<th>Conditions de l'essai</th>
<th>Nombre de séquences de 10 cycles par dépôt</th>
<th>233/236 normalisé</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>$m_i$</td>
</tr>
<tr>
<td>I</td>
<td></td>
<td>1,110725</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1,111596</td>
</tr>
<tr>
<td></td>
<td>U 500 # 1 traceur</td>
<td>1,110866</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1,110315</td>
</tr>
<tr>
<td></td>
<td></td>
<td>$\bar{m}=1,110833$</td>
</tr>
<tr>
<td>II</td>
<td></td>
<td>1,110346</td>
</tr>
<tr>
<td></td>
<td>U 500 # 5 traceur</td>
<td>1,110790</td>
</tr>
<tr>
<td></td>
<td></td>
<td>$\bar{m}=1,110386$</td>
</tr>
<tr>
<td>III</td>
<td></td>
<td>1,111287</td>
</tr>
<tr>
<td></td>
<td>U 850 # 5 traceur</td>
<td>1,110791</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1,111123</td>
</tr>
<tr>
<td></td>
<td></td>
<td>$\bar{m}=1,111128$</td>
</tr>
<tr>
<td>IV</td>
<td></td>
<td>1,110085</td>
</tr>
<tr>
<td></td>
<td>U 200 # 5 traceur</td>
<td>1,108675</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1,111116</td>
</tr>
<tr>
<td></td>
<td></td>
<td>$\bar{m}=1,110379$</td>
</tr>
<tr>
<td></td>
<td></td>
<td>$\bar{m}=1,110928$</td>
</tr>
</tbody>
</table>
De plus les rapports \((235/236)_M\) et \((238/236)_M\) de ces quatre mélanges sont compris entre 1 et 9 ; ce qui permet de vérifier la linéarité de la chaîne de mesure.

Plusieurs dépôts ont été faits pour chacun de ces mélanges.

Les valeurs normalisées du rapport 233/236 du traceur, calculées à partir des analyses de ces mélanges, sont présentées dans le tableau IV.

Pour chaque mélange, la valeur moyenne \(\overline{m}\) indiquée correspond à la moyenne pondérée des valeurs obtenues pour chacun des dépôts; elle a été calculée par la relation:

\[
\overline{m} = \frac{\sum m_i / \sigma_i^2}{\sum 1 / \sigma_i^2}
\]

avec \(m_i = \) moyenne de \(n\) séquences de 10 cycles calculée pour chaque dépôt.

L'écart type associé à \(\overline{m}\) est calculé par l'expression:

\[
\sigma^2 = \frac{1}{\sum 1 / \sigma_i^2}
\]

Ce tableau appelle les remarques suivantes :

Les valeurs de \(\overline{m}\) déduites des essais I et II, où le rapport 233/236 du traceur est normalisé à partir du même standard, ne sont pas significativement différentes, bien que les corrections dues aux contributions en \(235\) et \(238\) du traceur, soient environ cinq fois plus importantes dans l'essai I que dans l'essai II.

Le recouplement entre ces valeurs de \(\overline{m}\) et celles des essais III et IV est satisfaisant; il est compatible avec la cohérence interne des standards isotopiques d'uranium du NBS (justesse certifiée à \(10^{-3}\) et estimée à quelques \(10^{-4}\) par les auteurs).

La valeur absolue attribuée au rapport 233/236 du traceur double est : \(\overline{M} = 1,110928\); la précision relative attribuée à cette valeur est égale à \(2.10^{-4}\) au niveau de confiance de 95% en ne tenant pas compte de l'erreur sur les valeurs certifiées des standards isotopiques.
TABLEAU V. RAPPORTS ISOTOPIQUES NORMALISES DU TRACEUR DOUBLE

<table>
<thead>
<tr>
<th></th>
<th>233/236</th>
<th>234/236</th>
<th>235/236</th>
<th>238/236</th>
</tr>
</thead>
<tbody>
<tr>
<td>M</td>
<td>1,110928</td>
<td>0,006094</td>
<td>0,104240</td>
<td>0,014933</td>
</tr>
<tr>
<td>σ</td>
<td>0,000112</td>
<td>0,000008</td>
<td>0,000033</td>
<td>0,00014</td>
</tr>
</tbody>
</table>

TABLEAU VI. COMPOSITION ISOTOPIQUE EXACTE DU TRACEUR DOUBLE, EN ATOMES %

<table>
<thead>
<tr>
<th></th>
<th>233</th>
<th>234</th>
<th>235</th>
<th>236</th>
<th>238</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>49,6794</td>
<td>0,2725</td>
<td>4,6615</td>
<td>44,7188</td>
<td>0,6678</td>
</tr>
<tr>
<td>±</td>
<td>± 0,0053</td>
<td>± 0,0007</td>
<td>± 0,0029</td>
<td>± 0,0047</td>
<td>± 0,0012</td>
</tr>
</tbody>
</table>

4. DETERMINATION DE LA COMPOSITION ISOTOPIQUE EXACTE DU TRACEUR DOUBLE.

Cette valeur absolue du rapport 233/236 et les moyennes pondérées des valeurs normalisées des rapports 234/236, 235/236, 238/236 résultant de cinq analyses isotopiques du traceur double pur ont groupées dans le tableau V.

La composition isotopique de ce traceur, calculée à partir de ces rapports, est présentée dans le tableau VI.

5. DETERMINATION DE LA CONCENTRATION EN URANIUM DU TRACEUR DOUBLE.

La concentration en uranium du traceur double a été déterminée par dilution isotopique avec une solution de référence d'uranium naturel, préparée par dissolution d'un lingot d'uranium de pureté garantie.
La concentration en uranium de la solution de référence étant égale à : 484,33 μg/g, une aliquote de solution mère concentrée de traceur double a été diluée pour obtenir une concentration voisine de celle de la solution de référence. Trois mélanges de traceur dilué et de solution de référence ont été faits en pesant avec précision des quantités sensiblement égales de ces deux solutions.

Ces mélanges ont été évaporés à sec, repris deux fois de suite par de l'acide nitrique 8M, puis évaporés de nouveau à sec. La reprise finale a été faite par de l'acide nitrique 0,2M de manière à obtenir une concentration en uranium de l'ordre de 2 mg/ml.

Pour chacun des dépôts réalisés à partir de ces mélanges, des séquences successives de 10 cycles de mesures des rapports 233/236 et 238/236 ont été enregistrées une demi-heure après le début du chauffage des filaments, la durée d'une séquence étant de 8 minutes. Les moyennes des rapports 238/233, normalisés à la valeur absolue du rapport 233/236, ont été effectuées par blocs de cinq séquences successives et les concentrations en uranium correspondantes ont été calculées à partir de la formule classique de dilution isotopique :

\[ C_T = C_{\text{réf}} \times \frac{m_{\text{réf}}}{m_T} \times \frac{M_T}{M_{\text{réf}}} \times \frac{(238)_{\text{Réf}}}{(233)_{\text{T}}} \times \frac{1}{(238)} - \frac{(238)}{(233)}_M \]

dans laquelle :
- \( m_{\text{réf}} \) et \( m_T \) sont les masses respectives de solution de référence et de traceur mélangées;
- \( M_{\text{réf}} \) et \( M_T \) sont les poids atomiques de l'uranium contenu dans la solution de référence et le traceur, \((238)_{\text{Réf}}\), \((238)_T\) les abondances en \(^{238}\text{U}\) dans la solution de référence et le traceur exprimées en atomes.

À titre indicatif, les valeurs moyennes des rapports 238/233 normalisés et des concentrations correspondantes obtenues, pour quatre blocs de cinq séquences successives d'un même dépôt, sont comparées dans le tableau VII, aux valeurs moyennes des rapports non normalisés et des concentrations déduites de ces rapports.

On constate une évolution continue en fonction du temps des rapports 238/233 non normalisés, qui traduit le fractionnement isotopique, alors que les moyennes des rapports normalisés, calculées pour chacun des blocs de cinq séquences, ne sont pas significativement différentes.
TABLEAU VII. COMPARAISON ENTRE LES MOYENNES DES RAPPORTS 238/233 NORMALISÉS ET NON NORMALISÉS ET ENTRE LES VALEURS CORRESPONDANTES DES CONCENTRATIONS EN URANIUM

<table>
<thead>
<tr>
<th>Nombre de séquences de 10 cycles de mesures</th>
<th>238/233 normalisé</th>
<th>Concentration en U du traceur C₁ (µg/g)</th>
<th>238/233 non normalisé</th>
<th>Concentration en U du traceur C₂ (µg/g)</th>
<th>Ecart relatif C₂ - C₁ x1000 C₁</th>
</tr>
</thead>
<tbody>
<tr>
<td>5 m₁</td>
<td>2,329904</td>
<td>306,060</td>
<td>2,316300</td>
<td>307,870</td>
<td>5,9</td>
</tr>
<tr>
<td>σ₁</td>
<td>0,001446</td>
<td>0,191</td>
<td>0,001686</td>
<td>0,226</td>
<td></td>
</tr>
<tr>
<td>5 m₁</td>
<td>2,327975</td>
<td>306,315</td>
<td>2,322310</td>
<td>307,066</td>
<td>2,5</td>
</tr>
<tr>
<td>σ₁</td>
<td>0,001506</td>
<td>0,179</td>
<td>0,004275</td>
<td>0,549</td>
<td></td>
</tr>
<tr>
<td>5 m₁</td>
<td>2,328744</td>
<td>306,213</td>
<td>2,328232</td>
<td>306,281</td>
<td>2</td>
</tr>
<tr>
<td>σ₁</td>
<td>0,002385</td>
<td>0,296</td>
<td>0,005376</td>
<td>0,692</td>
<td></td>
</tr>
<tr>
<td>5 m₁</td>
<td>2,329456</td>
<td>306,119</td>
<td>2,333430</td>
<td>305,595</td>
<td>-1,7</td>
</tr>
<tr>
<td>σ₁</td>
<td>0,001754</td>
<td>0,212</td>
<td>0,003235</td>
<td>0,406</td>
<td></td>
</tr>
</tbody>
</table>

De ce fait, les écarts relatifs entre les concentrations déduites des rapports non normalisés et normalisés atteignent 0,59% pour les cinq premières séquences, diminuent progressivement et deviennent négatifs (-0,17%) pour les cinq dernières séquences enregistrées deux heures après le début de l'analyse.

La normalisation des rapports 238/233 permet donc d'éliminer les erreurs systématiques dues au fractionnement isotopique et d'obtenir, dès le commencement de l'analyse, des valeurs correctes, qui sont seulement entachées de l'erreur aléatoire de mesure des rapports 233/236 et 238/236.

L'ensemble des valeurs des concentrations en uranium obtenues, avec et sans normalisation, à partir des trois mélanges est présenté dans le tableau VIII. Les valeurs moyennes ‾m indiquées pour ces mélanges correspondent aux moyennes pondérées des valeurs obtenues pour chaque dépôt, pour tenir compte du nombre inégal de mesures effectuées sur ces dépôts.
TABLEAU VIII. VALEURS MOYENNES DES CONCENTRATIONS EN URANIUM DU TRACEUR OBTENUES AVEC ET SANS NORMALISATION DU RAPPORT 233/236

<table>
<thead>
<tr>
<th>Mélange</th>
<th>Dépôt</th>
<th>Nombre de séquences de 10 cycles de mesures</th>
<th>Concentration en U déduite de rapports normalisés ( C_1 , \mu g/g )</th>
<th>Concentration en U déduite de rapports non normalisés ( C_2 , \mu g/g )</th>
<th>Ecart relatif ( \frac{C_2 - C_1 \times 1000}{C_1} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>I</td>
<td>1</td>
<td>20 ( m_i ) ( \sigma_i ) 306,179</td>
<td>0,104</td>
<td>307,241</td>
<td>0,180</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>15 ( m_i ) ( \sigma_i ) 305,879</td>
<td>0,124</td>
<td>307,870</td>
<td>0,344</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>20 ( m_i ) ( \sigma_i ) 305,899</td>
<td>0,115</td>
<td>307,000</td>
<td>0,153</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>( \bar{m} ) 306,004</td>
<td>0,065</td>
<td>307,180</td>
<td>0,110</td>
</tr>
<tr>
<td>II</td>
<td>1</td>
<td>10 ( m_i ) ( \sigma_i ) 305,691</td>
<td>0,034</td>
<td>306,563</td>
<td>0,192</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>5 ( m_i ) ( \sigma_i ) 305,968</td>
<td>0,060</td>
<td>307,462</td>
<td>0,705</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>( \bar{m} ) 305,760</td>
<td>0,030</td>
<td>306,618</td>
<td>0,190</td>
</tr>
<tr>
<td>III</td>
<td>1</td>
<td>40 ( m_i ) ( \sigma_i ) 305,914</td>
<td>0,014</td>
<td>306,172</td>
<td>0,026</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>25 ( m_i ) ( \sigma_i ) 305,915</td>
<td>0,075</td>
<td>305,734</td>
<td>0,098</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>( \bar{m} ) 305,914</td>
<td>0,014</td>
<td>306,142</td>
<td>0,025</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>( \bar{m} ) 305,980</td>
<td>0,013</td>
<td>306,929</td>
<td>0,024</td>
</tr>
</tbody>
</table>
TABLEAU IX. DETERMINATION DE LA CONCENTRATION EN URANIUM D'UNE SOLUTION DE COMBUSTIBLE IRRADIE

<table>
<thead>
<tr>
<th>Durée de l'analyse</th>
<th>Nombre de séquences de 10 cycles de mesures.</th>
<th>1er dépôt Concentration en U de la solution en µg/g</th>
<th>2ème dépôt Concentration en U de la solution en µg/g</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>1ère séquence AVEC normalisation</td>
<td>2ème séquence AVEC normalisation</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Sans normalisation</td>
<td>Sans normalisation</td>
</tr>
<tr>
<td>1 h 20</td>
<td>5</td>
<td>351,845</td>
<td>348,216</td>
</tr>
<tr>
<td>2 h 00</td>
<td>5</td>
<td>351,750</td>
<td>348,519</td>
</tr>
<tr>
<td>2 h 40</td>
<td>5</td>
<td>351,452</td>
<td>349,537</td>
</tr>
<tr>
<td>3 h 20</td>
<td>5</td>
<td>351,862</td>
<td>350,219</td>
</tr>
<tr>
<td>4 h 00</td>
<td>5</td>
<td>M : 351,727</td>
<td>M : 349,123</td>
</tr>
<tr>
<td>4 h 40</td>
<td>5</td>
<td>σ : 0,19</td>
<td>σ : 0,92</td>
</tr>
<tr>
<td>5 h 20</td>
<td>5</td>
<td>M : 351,723</td>
<td>M : 351,35</td>
</tr>
<tr>
<td></td>
<td></td>
<td>σ : 0,057</td>
<td>σ : 0,88</td>
</tr>
</tbody>
</table>

Dans le cas de la normalisation, toutes les valeurs issues de dépôts et de mélanges différents sont remarquablement groupées; la moyenne pondérée M de l'ensemble est connue avec une précision relative de $10^{-4}$ au niveau de confiance de 95%. On constate un écart systématique d'environ 0,3% entre cette valeur et la moyenne des concentrations déduite des rapports 238/233 non normalisés.

6. UTILISATION DU TRACEUR DOUBLE POUR DOSER L'URANIUM CONTENU DANS UNE SOLUTION DE COMBUSTIBLE IRRADIE.

Ce traceur double a été utilisé en routine pour déterminer la concentration en uranium d'une solution d'un combustible irradié dans un réacteur de la filière à eau légère dans des conditions analogues à celles de l'étalonnage.

Pour éliminer l'interférence due à la présence de $^{238}\text{Pu}$ dans le combustible la séparation du plutonium contenu dans le mélange : (traceur double, combustible) a été réalisée sur une colonne de résine échangeuse d'ions.
Comme précédemment des séquences successives de dix cycles de mesures des rapports 233/236 et 238/236 ont été enregistrées et les rapports 238/233 ont été normalisés à la valeur absolue du rapport 233/236 en effectuant la correction due à la présence de $^{236}\text{U}$ dans l'échantillon (0,4\%).

Les concentrations en uranium ont été calculées à partir des rapports 238/233, normalisés et non normalisés, mesurés sur deux dépôts du même mélange; elles sont présentées dans le tableau IX. Ces résultats permettent de faire les commentaires suivants:

Les concentrations, calculées à partir des rapports non normalisés, évoluent de façon continue au cours de l'analyse et les courbes de fractionnement isotopique ne sont pas reproductibles d'un dépôt à l'autre.

Par contre la reproductibilité des concentrations, déduites des rapports normalisés, est remarquable pour les deux dépôts pendant toute la durée de l'analyse; l'écart entre les deux moyennes est inférieur à $3.10^{-4}$.

7. CONCLUSION.

Ces résultats montrent que la méthode de l'étalon interne permet d'améliorer à la fois la justesse et la reproductibilité des dosages d'uranium par dilution isotopique.

Cette technique se révèle particulièrement intéressante pour les déterminations précises de concentration en uranium effectuées dans le cadre des contrôles de matières nucléaires et des mesures de bilan de matières fissiles.

REFERENCES

OPERATIONAL EXPERIENCE WITH THE CERIC OXIDATION, FERROUS REDUCTION AND DICROMATE TITRATION METHOD FOR PLUTONIUM IN FAST REACTOR FUEL REPROCESSING

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UKAEA Dounreay Nuclear Power Development Establishment, Thurso, Caithness, Scotland, United Kingdom

Abstract

OPERATIONAL EXPERIENCE WITH THE CERIC OXIDATION, FERROUS REDUCTION AND DICROMATE TITRATION METHOD FOR PLUTONIUM IN FAST REACTOR FUEL REPROCESSING.

A new titrimetric procedure was described at the 1978 IAEA nuclear safeguards technology symposium in which plutonium is oxidized to plutonium VI by cerium IV in nitric acid solution, the excess oxidant is destroyed in a series of redox reactions, and the plutonium VI is reduced by a measured excess of iron II which is then back-titrated with potassium dichromate. This paper describes the further development of this method, and the operational experience gained in routine use. The absence of bias in the basic procedure was demonstrated using NBS plutonium metal standard. Iron, chromium, manganese, zinc, molybdenum, uranium, americium, and mixed fission products are shown not to interfere at the levels expected in nitric acid solutions of irradiated fast breeder fuel. Both vanadium and neptunium interfere quantitatively. Examination of the reaction between plutonium VI and the arsenic III used to reduce excess cerium IV suggests that it could be a source of bias, but detailed examination of the recommended procedure shows there is no problem. Temperature variations between 15°C and 30°C are shown to be tolerable. Routine operational experience for accountancy and safeguards purposes has been very good for both input and product streams.

1. INTRODUCTION

A new titrimetric procedure described at the 1978 IAEA safeguards technology symposium (1) was specifically designed to meet the requirement of plutonium accountancy during the reprocessing of irradiated plutonium-based fast breeder fuel. The method was applicable to nitric acid feed solutions, without separation of plutonium from fission products or uranium, using simple in-cell manipulations and producing results with high precision and negligible bias.
FIG. 1A
REDUCTION OF 1M CERIUM IV
WITH ARSENIC III

FIG. 1B
OXIDATION OF ARSENIC III
WITH PERMANGANATE VII

FIG. 1C
REDUCTION OF MANGANESE VII
WITH OXALIC ACID

FIG.1. Amperometric detection of intermediate reactions.
This paper describes the further development and performance testing of the method, and the operational experience gained in routine use.

2. METHOD PRINCIPLE

Plutonium in 1M nitric acid solution is oxidised to plutonium VI with cerium IV. At this acidity 1 millimole of cerium IV will oxidise more than 60 mg of plutonium IV in less than 5 minutes. Sulphamic acid is added to prevent nitrite-induced side reactions. Excess cerium IV is reduced by dropwise addition of sodium arsenite solution, catalysed by osmium tetroxide, a slight excess of arsenite being added. This excess is oxidised by dropwise addition of 0.2N potassium permanganate solution until an excess of permanganate is present. The excess of permanganate is reduced by the dropwise addition of 0.1M oxalic acid solution, iron III being used to catalyse the reduction. A small excess of oxalic acid does not interfere in the subsequent plutonium determination.

These reduction and oxidation stages can be followed amperometrically and leave the plutonium in the hexavalent state. Sulphuric acid is added, followed by a measured amount of standardised ferrous ammonium sulphate solution in excess of that required to reduce the plutonium VI to plutonium IV. The excess iron II and any plutonium III formed are back-titrated amperometrically to iron III and plutonium IV with standard potassium dichromate solution. The ferrous ammonium sulphate is standardised regularly by titration with standard potassium dichromate solution.

The plutonium content is calculated from the plutonium VI to plutonium IV change, by reference to the standard potassium dichromate. The latter may be checked either directly or indirectly against an internationally recognised plutonium reference material using the same titration procedure, or against NBS potassium dichromate SRM 136C.

The amperometric curves expected for the intermediate reactions used to destroy excess cerium IV are illustrated in Fig 1, while the final amperometric end point is illustrated in Fig 2. The experimental procedure has been refined since the earlier symposium (1) to improve the reliability of the method. The current procedure is detailed in the Appendix. Changes include washing down the titration vessel walls between each addition, controlling the drop size and reaction time of some of the intermediate reagents and the detection of completion of each.
FIG. 2. Typical amperometric titration curve.
### Table I

<table>
<thead>
<tr>
<th>Procedure</th>
<th>Results: (% Recovery)</th>
<th>Mean</th>
<th>Coefficient of variation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Recommended</td>
<td>99.90</td>
<td>100.07</td>
<td></td>
</tr>
<tr>
<td>Titration</td>
<td>100.11</td>
<td>99.95</td>
<td>99.98</td>
</tr>
<tr>
<td>Procedure*</td>
<td>99.96</td>
<td>99.98</td>
<td>99.95</td>
</tr>
<tr>
<td>Gravimetric</td>
<td>100.00</td>
<td>99.90</td>
<td></td>
</tr>
<tr>
<td>Procedure</td>
<td>100.04</td>
<td>99.95</td>
<td>99.99</td>
</tr>
</tbody>
</table>

* Results calculated by reference to NBS potassium diochromate.

The speed of locating a precise end point has been increased by using a fixed volume, incremental addition technique for the final dilute titrant with graphical interpolation.

The procedure described is currently being considered by an ISO working group as a standard method for plutonium determination.

## 3. Method Performance

Most of the additional development work has centered on assessing the performance of the method in more detail, including the effect of method variables and extraneous cations (see sections 3.1 to 3.3).

The absence of bias in the basic procedure was demonstrated by dissolving three portions of NBS plutonium metal SRM 949e in dilute sulphuric acid and analysing by the recommended technique. The solutions were also cross checked by a standard gravimetric technique. The results, summarised in Table I, show that with pure plutonium the method meets the requirements both for precision and absence of bias.

### 3.1 Effect of Impurities

The impurities considered were those which are likely to occur during the nitric acid dissolution of fuel from stainless steel cans, or those known to occur as constituents in irradiated
## TABLE II

### Effect Of Cations

<table>
<thead>
<tr>
<th>Cations</th>
<th>Amount added</th>
<th>% recovery</th>
<th>No data point</th>
<th>Significance test on difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fe$^{3+}$</td>
<td>2.0 mg</td>
<td>100.02</td>
<td>6</td>
<td>Not sig</td>
</tr>
<tr>
<td></td>
<td>10.0 mg</td>
<td>100.12</td>
<td>4</td>
<td>Not sig</td>
</tr>
<tr>
<td></td>
<td>50.0 mg</td>
<td>99.94</td>
<td>2</td>
<td>-</td>
</tr>
<tr>
<td>Zn$^{2+}$</td>
<td>10.0 mg</td>
<td>100.03</td>
<td>6</td>
<td>Not sig</td>
</tr>
<tr>
<td>Mo$^{6+}$</td>
<td>10.0 mg</td>
<td>100.00</td>
<td>6</td>
<td>Not sig</td>
</tr>
<tr>
<td>V$^{V}$</td>
<td>0.05 mg</td>
<td>100.39</td>
<td>5</td>
<td>Significant</td>
</tr>
<tr>
<td></td>
<td>2.5 mg</td>
<td>117.4</td>
<td>4</td>
<td>Equivalent to one electron charge</td>
</tr>
<tr>
<td>Cr$^{3+}$</td>
<td>1.5 mg</td>
<td>100.01</td>
<td>6</td>
<td>Not sig</td>
</tr>
<tr>
<td></td>
<td>5.0 mg</td>
<td>100.03</td>
<td>4</td>
<td>Not sig</td>
</tr>
<tr>
<td></td>
<td>10.0 mg</td>
<td>99.93</td>
<td>2</td>
<td>-</td>
</tr>
<tr>
<td>Mn$^{2+}$</td>
<td>1.5 mg</td>
<td>100.08</td>
<td>6</td>
<td>Not sig</td>
</tr>
<tr>
<td></td>
<td>5.0 mg</td>
<td>100.24</td>
<td>4</td>
<td>Significant at 5% probability level</td>
</tr>
<tr>
<td></td>
<td>10.0 mg</td>
<td>100.08</td>
<td>2</td>
<td>-</td>
</tr>
<tr>
<td>Am</td>
<td>2.9 mg</td>
<td>100.08</td>
<td>2</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>5.8 mg</td>
<td>100.23</td>
<td>4</td>
<td>Significant at 5% probability level. See text</td>
</tr>
<tr>
<td></td>
<td>10.0 mg</td>
<td>100.08</td>
<td>2</td>
<td>-</td>
</tr>
<tr>
<td>Np</td>
<td>0.92 mg</td>
<td>102.5</td>
<td>4</td>
<td>Significant. See text</td>
</tr>
<tr>
<td></td>
<td>4.6 mg</td>
<td>112.4</td>
<td>4</td>
<td></td>
</tr>
<tr>
<td>U</td>
<td>200.0 mg</td>
<td>100.1</td>
<td>6</td>
<td>Not significant</td>
</tr>
<tr>
<td>Fission Products</td>
<td>See Text 100.05</td>
<td>5</td>
<td>Not significant</td>
<td></td>
</tr>
</tbody>
</table>
fuel. The levels of impurity tested were at high but reasonable levels for those impurities whose level can be reasonably predicted, and at arbitrary levels for the unpredictable impurities; 35 to 40 mg plutonium was used in all the impurity trials.

The results, in Table II, show that iron, zinc and molybdenum do not interfere at the 10 mg level. Vanadium causes a high result equivalent to a one electron change, probably vanadium III to vanadium IV. Some but not all batches of iron used for interference tests at the 50 mg level have given an apparent small bias. This could be due to traces of vanadium in the iron used.

Initial tests with chromium and manganese present resulted in biased results, until it was realised that the shape of the amperometric end point used to follow the destruction of excess cerium IV had altered. Adding arsenite until the current dropped to a minimum produced the biased results. By continuing the arsenite addition until the current started to rise again, bias effects were eliminated and the results in Table II were obtained, although large amounts of manganese give more variable results than normal, associated with a trace of brown precipitate. The amperometric curves are compared in Fig 3.

Both uranium and fission products are major constituents of irradiated fast breeder fuel. Fission product interference tests were performed using an actual plant fission product solution because of the varied chemical states that may be present. A first cycle raffinate from the fuel reprocessing plant was used, derived from fast reactor fuel of over 8% burn up. The quantity added was about 50% more than would be associated with the normal maximum aliquot of fuel solution. Uranium was added as uranyl nitrate, in excess of 50% more than normally present. As Table II shows, there is no interference from either under these realistic conditions.

Both americium and neptunium are present in irradiated fuels, at levels dependent on fuel history and isotopic contents. Excessive levels of both were examined. 6 mg of added americium appeared to produce a small positive bias, but this is probably due to the presence of about 1% of either neptunium daughter or of plutonium in the americium used. Neptunium interferes quantitatively. Adding 0.9 mg of neptunium resulted in an error equivalent to 0.82 mg of plutonium, while adding 4.6 mg neptunium produced an error equivalent to 4.7 mg plutonium. Fortunately the neptunium content of a plutonium based fast breeder fuel should normally be less than one thousandth of the plutonium, and if necessary a correction can be applied.
FIG. 3. Titration curves for the cerium/arsenite reaction.
3.2 Intermediate reactions

The excess cerium IV used to oxidised plutonium to plutonium VI at the start of the procedure is subsequently reduced by a series of 3 reactions, as outlined in Section 2. These were assessed for possible interfering side reactions in the original paper. One possible side reaction, the reduction of plutonium VI by arsenic III, could not be ruled out because of an absence of data in the literature, although both initial tests and subsequent experience suggested it does not cause any problems.

This reaction was specifically studied using spectrophotometric techniques. Arsenic III levels 30 times higher and plutonium levels one tenth of these used in a normal titration were required to achieve easily measured reaction rates.

The initial reaction is a reduction of plutonium VI to plutonium V. Initial rate measurements were not completely reproducible, but the rate increased as the acidity decreased from 5M to 0.1M nitric acid, and as the arsenic III concentration increased. Typical results are summarised in Table III. Extrapolation of this data to the conditions present during a titration suggests that at most only a very small proportion (considerably less than 0.1%) of any plutonium VI should be reduced. The results do emphasise the necessity to control the addition of excess arsenic III, and the advisability of using an acidity greater than 0.5M. Additionally, the formation of plutonium V does not directly cause a bias in the titrimetric procedure as it is quantitatively re-oxidised on addition of manganese VII.

However plutonium V disproportionates to give plutonium IV or plutonium III both of which could cause a bias, the rate of
disproportionation increasing with increasing acidity (2). Thus any bias caused by reduction of plutonium VI by arsenic III is minimised by using the lowest practicable acidity for this stage of the procedure.

The studies of the reduction of plutonium VI by arsenic III suggest that while slight changes in the acidity of the recommended procedure are probably unimportant, the quantity of excess arsenic III added, and the time available for its reaction, could be important. Consequently the practical effect of varying these parameters was examined. The results obtained when the waiting period after adding the arsenic III was omitted altogether or increased to 5 minutes, or when up to 8 drops excess arsenic III was added, showed no change compared to typical results obtained using the recommended procedure. Thirty nine results covering these variations gave a mean recovery of 99.99% with a coefficient of variation (for a single result) of 0.12%. The procedure is therefore sufficiently robust to be unaffected by any likely practical variations.

### TABLE IV

Temperature Effect

<table>
<thead>
<tr>
<th>Temperature</th>
<th>Mean recovery (%)</th>
<th>No. of data points</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>9°</td>
<td>99.99</td>
<td>8</td>
<td>less precise results due to sluggish reactions</td>
</tr>
<tr>
<td>15°</td>
<td>100.01</td>
<td>4</td>
<td></td>
</tr>
<tr>
<td>21°</td>
<td>100.03</td>
<td>4</td>
<td></td>
</tr>
<tr>
<td>30°</td>
<td>100.00</td>
<td>8</td>
<td></td>
</tr>
<tr>
<td>34°</td>
<td>100.18</td>
<td>6</td>
<td>results less precise, difference not significant</td>
</tr>
<tr>
<td>39°</td>
<td>100.36</td>
<td>6</td>
<td>results less precise, difference significant at 95% probability level</td>
</tr>
</tbody>
</table>
3.3 Effect of Reaction Temperatures

A series of plutonium control titrations were performed in a titration vessel fitted with a thermostated water jacket, which was either heated or cooled. Results (Table IV) obtained at temperatures between 15°C and 30°C were no different from normal control results under ambient temperatures. At lower temperatures the end point of the intermediate reaction became sluggish, and the results were slightly less precise. At temperatures much above 30°C results were again more variable, with a statistically significant bias developing. The bias and increased variation is assumed to reflect an increased reaction rate for a side reaction which is normally inconsequential at ambient temperatures. Adjustment to the detailed procedure may enable it to be used at higher temperatures should this be required.

3.4 Method Experience

The method has been in regular use for plutonium accountancy in an irradiated plutonium/uranium oxide fuel reprocessing facility. During this period 800 aliquots of plutonium nitrate product material have been analysed in a glove box, and 700 aliquots of irradiated fuel solution in a remote handling shielded facility, with no significant problems. Up to 10 plutonium aliquots can be analysed by one operator in an 8 hour day, in either remote or glove box facilities. Typical performance data is shown in Table V obtained by several
operators over a period of weeks. The main problem has been the preparation and maintenance over several weeks of adequately precise plutonium control solution, in the far from ideal conditions of a routinely used glovebox or remote handling cell. There do not appear to be any critical steps in the method for an operator who is capable of obtaining a precision of 0.1% in any other analytical procedure.

4. CONCLUSIONS

The titrimetric procedure described at the previous IAEA Safeguards Conference has been examined in more detail for both robustness and freedom from bias. The basic procedure has been shown to give unbiased results with NBS plutonium metal standard. The levels of uranium and fission products encountered in a solution of irradiated fast reactor fuel do not interfere. Common cations such as iron, chromium and zinc do not interfere when present at the 5 or 10 mg level. Neptunium and vanadium both interfere quantitatively, but the levels encountered in a plutonium based fuel are low.

Although reduction of plutonium VI by arsenic III is identified as a potential interference, the quantities of arsenic III required to achieve measurable plutonium reduction are large, and the recommended titration conditions minimise the likelihood of such interference. Deliberate variations in the titration conditions had no effect, thus demonstrating the robustness of the procedure. Temperature variations between 15°C and 30°C have no effect. The same method is applicable to both input and product solutions from fast reactor fuel reprocessing, without preliminary separation or large dilution steps. Only simple operations are required, in a single vessel at room temperature. The method promises to be a satisfactory plutonium counterpart for the familiar modified Davies and Gray method for uranium (3).

APPENDIX: RECOMMENDED PROCEDURE

1. Weigh a portion of sample containing between 10 and 60 mg of plutonium and not more than 10 milliequivalents of free acid into a 100 ml beaker.

2. Add 1 ml of 1M ceric ammonium nitrate solution (in 1M nitric acid), wash down the walls of the beaker with 30 ml of ferric nitrate/sulphamic acid reagent (1) and insert a pair of gold electrodes (0.5 cm² per side). Stir gently

(1) 0.005M ferric nitrate and 0.1M sulphuric acid in 1M nitric acid.
for 5 minutes and apply a potential (usually 200-300 mV) to the electrodes to give a current of 15-20 \( \mu \text{A} \).

3. Add 0.1 ml of 0.25% osmium tetroxide solution (in 2M sulphuric acid). Add 0.5M sodium arsenite solution dropwise until the observed current falls to a steady reading of about 10 \( \mu \text{A} \). Continue to add single drops of sodium arsenite solution, waiting 10 sec between each addition, until 1 drop results in the current reading starting to rise again after passing through a minimum (see fig 1A). At this stage wash down the walls of the beaker with 10 ml of ferric nitrate/sulphamic acid reagent.

4. Stir gently for at least 1 minute but not more than 2 minutes after adding the final drop of arsenite solution, and then immediately add 0.2N potassium permanganate solution dropwise until the current falls to about 2 \( \mu \text{A} \) (see Fig 1B). Then add 1 further drop, resulting in a sharp increase in current to about 10 \( \mu \text{A} \) and a stable pink colour. Wash down the beaker walls with 10 ml of ferric nitrate/sulphamic acid reagent and stir for 3 minutes.

5. Add 0.1M oxalic acid solution dropwise until the addition of 1 further drop causes no further reduction in current (see Figure 1C). The pink permanganate colour should disappear. Add 5 drops (about 0.05 ml) in excess. Wash down the walls of the beaker with 10 ml of 1M sulphuric acid solution.

6. Add sufficient 0.1M ferrous ammonium sulphate solution from a weight burette to reduce the plutonium VI and leave a suitable excess for the titration. Wash down the beaker walls with 5-10 ml of 1M sulphuric acid solution.

7. Stir gently for 3-5 minutes. Ensure that the microammeter is showing a steady current of 10-20 microamps.

8. Add 0.05N potassium dichromate standard solution from a weight burette until just before the end point, ie until the current reading is about 5 \( \mu \text{A} \). (Figure 2 shows a typical titration curve). Wash down the walls of the beaker with 1M sulphuric acid solution.

9. Continue the titration by adding 100 microlitre increments of dilute (0.002N) potassium dichromate solution from a delivery pipette. After each addition record the current reading; continue the titration until at least two current readings after the end point (indicated by a steady, low current) have been recorded.
10. Locate the exact end point from a graph of current versus number of 100 microlitre increments of dilute potassium dichromate solution. The end point is the number of increments \( n_1 \) at the point of intersection shown in Figure 2.

11. The ferrous ammonium sulphate solution should be standardised at least once per day, proceeding as above, but without the plutonium aliquot in step 1, and adding about 0.5 ml of 1M ceric nitrate in step 2 and a weighed amount (about 5 g) of 0.1M ferrous solution in step 6.

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AUTOMATED URANIUM TITRATION SYSTEM

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Presented by Y. Tsutaki

Abstract

AUTOMATED URANIUM TITRATION SYSTEM.

An automated titration system based on the Davies-Gray method has been developed for accurate determination of uranium. The system consists of a potentiometric titrator with precise burettes, a sample changer, an electronic balance and a desk-top computer with a printer. Fifty-five titration vessels are loaded in the sample changer. The first three contain the standard solution for standardizing potassium dichromate titrant, and the next two and the last two contain the control samples for data quality assurance. The other forty-eight measurements are carried out for sixteen unknown samples. Sample solution containing about 100 mg uranium is taken in a titration vessel. At the pretreatment position, uranium(VI) is reduced to uranium(IV) by iron(II). After the valency adjustment, the vessel is transferred to the titration position. The rate of titrant addition is automatically controlled to be slower near the end-point. The last figure (0.01 mL) of the equivalent titrant volume for uranium is calculated from the potential change. The results obtained with this system on 100 mg uranium gave a precision of 0.2% (RSD, n = 3) and an accuracy of better than 0.1%. Fifty-five titrations are accomplished in 10 hours.

1. INTRODUCTION

The potentiometric titration method, which was introduced by Davies and Gray [1], and improved by Eberle and Lerner [2], is the most common analytical method for measuring uranium content in samples associated with safeguards inspection activities. Because of the increasing number of samples to be analysed at many nuclear facilities, some papers describing automation derived from a modified Davies and Gray method have been published [3—5].

In addition to saving manpower for analytical work, the elimination of operator bias and calculation errors without decreased precision can be achieved with automation. The standard modified Davies and Gray method requires large amounts of uranium and correspondingly large amounts of reagents, which results in much liquid waste.

For these reasons our laboratory has developed an automated uranium titration system which can titrate up to 55 samples in 100-mL cells containing about 100 mg uranium.
2. SYSTEM DESCRIPTION

2.1. Hardware

A block diagram of the system developed is shown in Fig.1. Details of each part are given below.

**Computer system**

The computer system includes a Hewlett-Packard 9835A desk-top computer with 128 kbytes of core memory, a 9876A graphics printer, a 98337A plotter ROM, and associated interfaces such as 98032A(GP-IO), 98034A(HP-IB)
and 98036A(RS-232C). This system controls mechanical and electrical operations through a control unit, processes titration data, prints out analytical results in report form, and plots the titration curve of each measurement and the X-R control chart if necessary.

**Control unit**

Various digital signals from the system components are incorporated into this unit, which is linked to the computer through a GP-IO interface. On the control panel, many switches — for selecting manual or automatic operation, setting reagent volume, controlling the sample changer, rotation controlling of magnetic stirrers, setting the waiting period between titrant deliveries — are provided. These switches allow an operator to manipulate the system components.

**Titration cell and sample changer**

A commercially available 100-mL beaker is used as a titration cell, which contains 100 mg uranium in volumes up to about 5 mL. The beakers are transported on a sample changer. Five beakers are placed in an aluminium magazine coated with Teflon film. The sample changer connected to the control unit has space for 11 magazines in one loading, which means the ability to titrate up to 55 samples in one loading. Each magazine moves along an X-Y axis to the pretreatment and titration position by being pushed by bars driven by four motors.

**Piston burette and reagent reservoir**

Five sets of model APB-117 20-mL auto-piston burette (Kyoto Electronics Manufacturing Co., Ltd.) are used for delivering four different kinds of reagents and potassium dichromate solution as a titrant. These burettes, which are installed in the main cabinet as shown in Fig.2, can deliver with an accuracy of 0.02 mL at 20 mL. At the pretreatment position the following reagents are successively added by a signal from the control unit as shown in Fig.3.

1. 20 mL 85% phosphoric acid stored in a 2000-mL reservoir;
2. 2 mL 1M iron(II) sulphate-0.4M sulphamic acid-0.5M sulphuric acid mixture stored in a 250-mL reservoir;
3. 5 mL 4M nitric acid-1% ammonium molybdate-0.1M sulphamic acid mixture stored in a 500-mL reservoir;
4. 20 mL 0.1% vanadium(IV) oxide sulphate-0.5M sulphuric acid stored in a 2000-mL reservoir.
FIG. 2. Main cabinet of the automated uranium titration system. (1)–(5) burette and reagent reservoir; (6) pretreatment and titration station; (7) electrode washing vessel; (8) water pump; (9) A/D converter; (10) waste solution reservoir; (11) water reservoir.

At the titration position, 0.01M standard potassium dichromate titrant is added for titration. The titrant is stored in a 2000-mL reservoir surrounded by a thermal insulator to keep the temperature constant during the system operation. In addition, a platinum resistance thermometer is installed in the reservoir to monitor titrant temperature, which is printed out for each titration. A photo-diode sensor is attached to the outlet of each reservoir to detect reagent shortage.

Pretreatment and titration station

A pretreatment and titration station, installed on supporting rods above the sample changer, has two arms driven by two motors and a chain linkage. Both arms move not only up and down but also back and forth, respectively, for the reagent addition, titration and electrode washing by signals from the control unit. They always stay at the rest position when not in operation. The arm for pretreatment has four delivery tips, and the other for titration has a delivery tip and a combined platinum electrode – Metrohm model EA259.
FIG. 3. Valency adjustment and titration time sequence.


(1) If titration (2) is completed earlier than (1), then (4) succeeds (1) immediately.
(2) Titration period is variable. (3) Electrode washing is carried out for 90 s.

**Electrode washing vessel**

After each titration is completed, the arm with the titrant delivery tip and the combined electrode moves forward and down to an electrode washing vessel, which is a glass double-wall coaxial beaker. The combined electrode is rinsed three times with running fresh water. During the rinsing process, the titrant delivery tip stays between glass beaker walls so as not to get wet. By rinsing the electrode the uranium contaminated liquid waste is collected into a waste reservoir.
Balance

An electronic balance, model AK-160 (Mettler Instrument Corp.), is used for weight base aliquoting of sample solution. The balance has a maximum capacity of 160 g, and a readability of 0.1 mg. The digital signals of weight measurement are sent to the computer memory through the interface.

2.2. Software

The interactive system software with the operator has been developed for the acquisition and storage of sample identification and weight information. This software also controls the titration process — operation of sample changer, reagent addition for pretreatment, titration and electrode washing; and then processes the titration data to obtain the analytical results and prints out tables of analytical results and titration curves. In addition to these major roles, an analytical data quality assurance program and a fault-monitoring program are incorporated in this system software.

The system software consists of a main program and six sub-programs. The main program allows the operator to specify the job options. The first sub-program, OP1, includes three different jobs, such as “Learn titration”, “Test titration” and “Titration curve”. The first one is designed to determine various titration parameters based on some titration results of known samples and the titration curve; the second is for a preliminary test of titration performance and also for conditioning the electrode; and the third is for preparing the titration curve of each sample after the whole process is complete.

The second sub-program, OP3, is the fundamental program for automatic titration operation in which titrant delivery is controlled as follows: the first 1.00 mL titrant is added stepwise, with a 2-second interval after the continuous addition of 4.00 mL titrant. If the potential difference exceeds 3 mV, 0.12 mL titrant is added stepwise with a 2-second interval until the potential difference becomes 7 mV. After changing the increment of titrant to 0.02 mL with 7-second intervals, the titration is added stepwise with a delay of 7 seconds at each step until the maximal potential difference is detected. Total titrant volume added at the end-point is calculated from the three titrant volumes at the maximum potential difference and those before and after the maximum.

The third sub-program, OP5, is prepared for data quality assurance based on the X-R control chart method, in which control limits and mean bias are calculated from the last 40 consistent measurements (Fig.4).

The other three sub-programs with many files are prepared for controlling titration parameters, various messages and so on, and also for displaying the processing status on the CRT.
2.3. Fault-monitoring system

The developed automated system has a simple fault-monitoring system consisting of hardware sensors and software. When twelve kinds of fault (e.g. reagent shortage, abnormal movement of a sample changer, abnormalities of electrode potential and reagent temperature, read error of weight measurement etc.) are detected during automatic operation, the computer gives error
messages on the CRT and terminates sample processing except for read error of weight measurement and cassette access error. For the latter, the computer requests actions to be taken by the operator, and the process restarts once correct action has been performed.

3. SYSTEM OPERATION

The standard routine operation procedure of this system taken by a government inspector for analysing uranium concentration in samples such as uranium(IV) oxide powder and pellets is as follows.

After SRD measurement, about 2 g of the sample is taken and dissolved with 20 mL 3–7M nitric acid, then diluted to 100 mL, or to about 100 g solution on weight base. Three aliquots (5 mL or 5 g) of each sample solution are transferred into three titration cells. The standard uranium solution is prepared by dissolving high-purity uranium metal (JAERI-U4, 99.99%) and 5 mL, or 5 g, of the solution is also put into the titration cell three times. These cells are placed at position Nos 1, 2 and 3 of the first magazine, and two laboratory control samples containing about 100 mg uranium are placed at positions Nos 4 and 5 on the same magazine. The forty-eight cells containing the samples to be analysed are placed from position No.6 to No.53 in the second to eleventh magazines.

At position Nos 54 and 55 two laboratory control samples, the same as those at positions 4 and 5, are placed. If the total number of cells containing samples is less than 48, two laboratory control samples are always placed at the last two positions of the sample to be analysed.

At first the operator loads the system program from a program cassette into the computer memory, and selects the job number to be performed. The computer CRT shows a table with items to be checked before starting the system. To log sample information such as sample identification, material description and so on, the operator exchanges the program cassette for an information cassette which is issued from the host computer (HP-9845T) after the titration system has been checked. Subsequently, the operator enters the date of analysis, the operator code and number of cells and selects the aliquoting mode — volume base or weight base. If the automatic weighing mode is selected for aliquoting, a series of aliquot weights are automatically entered from the electronic balance.

For the standard solutions and titrant, three values are entered: the purity and weight of uranium metal and potassium dichromate which are taken, and the weight or volume of solutions resulting from the dissolution. For unknown samples, the weight of the sub-sample, the weight or the volume of the solution resulting from the dissolution, and the volume of each aliquot are entered. After
entering the required data into the computer memory, the operator can check and correct all the data entered, then load the magazines in which five cells are placed on the sample changer, and exchange the information cassette for a data cassette. By typing the “Cont” key of the computer, the system now starts to process 55 sample titrations.

The standard valency adjustment and titration time sequence for the routine operation is shown in Fig.3, which illustrates how the valency adjustment of the second sample is carried out while titration of the first sample is proceeding. After three cells containing uranium standard sample are processed,

<table>
<thead>
<tr>
<th>Rack Bottle</th>
<th>Material Item</th>
<th>Initial Endpoint</th>
<th>End Point</th>
<th>Temper.</th>
<th>Found</th>
<th>Relative standard deviation</th>
</tr>
</thead>
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<tr>
<td>No. No. descipt. code</td>
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<td>(mV)</td>
<td>(mL)</td>
<td>(us)</td>
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Mean of uranium content: 87.54 ut%
Relative standard deviation: 0.09%
three determined titrant concentrations are printed out with a mean value and bias from the theoretical value based on data previously entered. Two laboratory control samples are then titrated and the results obtained are compared with the control limits of an \( \bar{X} \)-R control chart. A system halt is initiated if the results are beyond the control limits. Once every three titrations of the same sample are completed, the analytical results with sample identification, the potential of electrode, the titrant volume at the end-point and the temperature of titrant, are printed out as shown in Table I, and these data are stored in the data cassette. The other samples are processed in a similar manner as above. Finally, two other laboratory control samples are titrated, and the upper and lower control limits updated and printed out. The whole process — 55 titrations — is accomplished in about 10 hours. The operator exchanges the data cassette for the program cassette to update the data file for the data quality assurance.

Whole data obtained in an analytical cycle are logged through the data cassette into a data base in the host computer system. The host computer re-calculate the uranium concentration of each sample by using atomic weight obtained by mass-spectrometric analysis and makes the final report for national safeguards purposes using the titration and mass-spectrometry results.

A titration curve (Fig.5) sometimes provides diagnostic information if the analytical results obtained are unusual compared with that expected. On this system, a titration curve for each measurement can be printed out if necessary after the whole titration process has been completed.
4. SYSTEM RELIABILITY

Various kinds of uranium-bearing samples have been analysed with this system and representative results are given in Table I. Those results, obtained on approximately 100 mg uranium, gave a precision of 0.2% (RSD, n = 3) and an accuracy of better than 0.1%.

This automated uranium titration system started operation in September 1981. After one year's routine operation, no serious problem has arisen with the hardware components except for unreliable action of switches for positioning the magazine on the sample changer. After the positioning mechanism was modified, this problem was solved. In one year's operation of this system, about 400 uranium-bearing samples have been analysed as routine work.

In conclusion, our laboratory has greatly succeeded not only in saving manpower for uranium analysis, but also in increasing the reliability of analytical results by the development of this system.

ACKNOWLEDGEMENTS

The authors wish to thank Mr. H. Kawabata, Kyoto Electronics Manufacturing Co., Ltd. and colleagues for their practical contribution in developing this system.

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CONTAINMENT AND SURVEILLANCE IN IAEA SAFEGUARDS

(Session 5)
Chairman

A.J. STIRLING
PENETRATION MONITORING AS A POTENTIAL SAFEGUARDS MEASURE
Review of present status

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Abstract

PENETRATION MONITORING AS A POTENTIAL SAFEGUARDS MEASURE: REVIEW OF PRESENT STATUS.

Penetration Monitoring is a form of extended Containment/Surveillance which is designed to detect bypassing of flow Key Measurement Points. This would be achieved in practice by providing a containment structure surrounding the Material Balance Area and monitoring all penetrations of the containment which present realistic diversion routes. The performance of a penetration monitoring system may be quantified in terms of detection probability and false alarm probability. The validity of the quantification of system performance depends in turn on: the quantifiability of the performance of individual detection devices; the independence of performance between individual devices; the completeness of coverage of all realistic diversion routes by the penetration monitoring system; the assurance that the integrity of the containment itself remains unimpaired. A sub-group of the International Working Group on Reprocessing Plant Safeguards carried out an appreciation of penetration monitoring as applied to this type of facility, discussing in some detail aspects of system performance including quantification of performance of types of detection devices. Since this study little further work has been reported. Problem areas remaining to be tackled include: (a) An investigation to ascertain which facilities or parts of facilities would lend themselves to the application of penetration monitoring, bearing in mind the needs of the Agency. (b) Design of penetration monitoring systems which do not involve an excessive number of detection devices; the solution of this problem depends very much on the choice of containment boundaries and the penetrations thereof which constitute realistic diversion routes. (c) Specification of performance requirements for detection devices, including those for demonstrating the integrity of the containment itself. (d) Formulation of appropriate response procedures to anomalies generated by a penetration monitoring system.

1. INTRODUCTION

The technical conclusion of the Agency's NPT verification activities is a statement in respect of each material balance area (MBA) of the amount of material unaccounted for (MUF) over a specified period, giving the limits of accuracy of the amounts stated. Statistical procedures including sampling plans are used to provide assurance that the required values for non-detection probability and false alarm probability are met for each MBA. For a given MUF, there is a trade-off
between these parameters which is determined by the material balance uncertainty as measured by the operator and verified by the inspector. In bulk handling facilities the material balance uncertainty expressed in absolute terms tends to increase with increasing throughput and this makes the detection of diversion of a given quantity of nuclear material more difficult to achieve with high probability by conventional safeguards approaches alone. It has therefore been suggested [1] that additional safeguards measures might be employed in support of conventional methods. The International Working Group on Reprocessing Plant Safeguards (IWG-RPS) included in their study an examination of two advanced safeguards measures, near-real-time material accountancy and extended containment/surveillance (C/S), which might be suitable for this purpose.

Conventional C/S measures are defined as those which give direct support to the nuclear material accountancy system. They either provide assurance of the validity of measurements of declared transfers at flow Key Measurement Points (KMPs) (e.g. by cameras or human surveillance) or maintain continuity of knowledge of a fixed inventory (e.g. by cameras or seals). Extended C/S measures seek to ensure, by surveillance of certain locations and conditions associated with diversion scenarios of concern to safeguards, that the inventory within the area is only changing, according to the declared transfers, at the defined flow KMPs. In this way it is claimed that extended C/S can provide additional support to the nuclear material accountancy system. Penetration monitoring involves the surveillance of diversion routes judged to be credible in diversion scenarios which require the removal of material through containment boundaries. In other words, penetration monitoring is designed to detect the bypassing of flow KMPs. It is envisaged that this could be achieved in practice by providing a containment structure around the MBA and monitoring with suitable devices all penetrations of the containment which represent technically realistic diversion routes.

The purpose of this paper is to review the present status of the safeguards potential of penetration monitoring and to indicate the areas where extra study and development are required.

2. THE CONCEPT OF PENETRATION MONITORING

The conventional use of seals on containers or access points to a store is very similar in safeguards concept to
that of penetration monitoring and a comparison of the two is instructive.

In the majority of cases seals are used to provide evidence as to whether or not a particular containment may have been penetrated between attachment and removal of the seal. Upon re-examination of the seal there are two possibilities:

(a) The seal is intact
(b) The seal is not intact.

If (a) applies, then assurance of non-diversion of the contents of the container is obtained, provided that:

(i) there has been no substitution of the seal
(ii) there has been no tampering with the seal (including opening and reclosing), and
(iii) the containment remains intact.

If one or more of these three conditions cannot be satisfied then the situation is as in (b); in which case diversion may have occurred and follow-up measures are required to confirm or discount this possibility.

There are however two major differences between the conventional use of seals and the penetration monitoring concept:

Where seals are used, there is usually only one credible route into or out of the contained space; this is covered by the seal itself. Where penetration monitoring is applied, there may be many credible routes into or out of the contained space, each of which must be covered by a seal, camera, TV system, nuclear material monitor or other device.

A sealed container or store is a static system, that is to say there can be no legitimate transfer of nuclear material into or out of the contained space while the seal is in place. On the other hand, where penetration monitoring is applied, e.g. to the process area of a bulk handling facility, legitimate transfers of input material, product and waste into and out of the monitored space do take place, hence the contained inventory can and usually does change from time to time.
If penetration monitoring is proposed for safeguards purposes, then the Agency would seek assurance on the following points [2]:

- that the monitoring of penetrations of containment is effective
- that the tamper protection of instruments is effective
- that nuclear material cannot credibly be used to further the manufacture of nuclear explosive devices [3] within the containment boundaries without detection. Examples where this condition applies include those zones in nuclear facilities which present a particularly hostile environment, such as inside a reactor primary containment vessel and the early stages of a spent fuel reprocessing facility.

3. THE APPLICATION OF PENETRATION MONITORING

There are a number of factors that would need to be borne in mind both in the practical application of penetration monitoring and in any quantification of the probability of detection of diversion which would be provided by such a system.

3.1. The number of surveillance devices required. These must cover all credible penetrations in the boundaries of a given contained space, together with the integrity of the boundaries themselves. In general, a greater number of devices implies greater complexity of operation (but perhaps easier resolution of anomalies). The number of devices is conditioned by:

(a) The selection of containment boundaries. In general, the closer the chosen boundaries are to the process equipment, the more penetrations will need to be monitored; on the other hand the more remote the chosen containment boundaries are from the process equipment, the smaller the number of penetrations, and the simpler the task of monitoring. It has been suggested [4] that the facility design might be reconfigured to reduce the number of penetrations or to redefine the containment boundaries to effect such a reduction. However, it is not clear to what extent it would be possible in practice to make major alterations here, in view of the many other technical and economic constraints. Moreover, it must be pointed out that where containment boundaries are
remote from the process equipment, it may be
difficult to demonstrate the non-credibility of the
use of nuclear material to further the manufacture of
explosive devices within those boundaries, without
detection. However, an interesting example of what
might be termed extended C/S is employed in the
safeguards approach for a Candu 600 reactor, where
the route taken by the irradiated fuel during
transfer from the reactor core to the bays is covered
by TV surveillance systems, photo-surveillance
cameras, radiation yes/no monitors, bundle counters
for spent fuel, an attribute verifier for spent fuel,
and spent fuel sealing equipment [5].

(b) The identification of all penetrations. Hence not
only walls but also the roof and floor must be
included. Problems of accessibility mean that the
inspector may have to verify some penetrations at an
early stage in the construction of the facility and
maintain assurance that no undeclared penetrations
are introduced at a later stage.

(c) The credibility of use of penetrations as diversion
routes. Unless all penetrations are to be monitored,
some decision is necessary as to which constitute
credible diversion routes. Those deemed incredible
would not be included in the penetration monitoring
system. It would be desirable if the subjective
elements of "credibility" could be reduced as far as
possible; the notion of "technical realism" of
diversion routes conveys a more objective picture of
what should be sought here. Subgroup I of the
IWG-RPS proposed [4] a list of technical factors
which affect credibility and which are reasonably
objective in nature. However, a considerable degree
of subjectivity seems inescapable in deciding:
- how to weight the relative importance of the
  factors
- how to combine the factors, and
- how to choose the "cut-off level" for
  credibility.

These problems were recognized by the Subgroup and
seem likely to be severe. If different (but
inherently reasonable) factors lead to widely
differing results the whole approach would be
suspect. This could be the most vulnerable step in
the whole evaluation process for process monitoring
but the complexity of the problem would depend on the facility type.

(d) The possibility of using one surveillance device to cover several penetrations would reduce the total number of devices required. Resolution of anomalies might however be adversely affected. Again, the potential for such reduction would be closely related to the actual design of the facility.

3.2. The type of surveillance devices required. From the point of view of penetration monitoring there are essentially three types of surveillance device:

(I) Those which provide yes/no information about possible undeclared operations. These include seals, cameras and TV systems (when the last two are used in situations which supply information which can be interpreted unambiguously).

(II) Those which provide a graded response depending on the quantity of nuclear material present. These are mainly radiation monitors but also include liquid-in-line detectors.

(III) Those which require essentially subjective interpretation by the inspector. These include cameras and TV systems used as an extension of human presence, together with human presence used directly.

The choice of device has an important bearing on the optimum diversion strategy, the response to an anomaly and the quantification of detection probability for the penetration monitoring system as a whole. Type (I) devices provide no information on the quantity of nuclear material removed. If the diverter chooses to violate such a device he will then remove as much material as possible in one operation since his chances of detection are no higher than if he removed a smaller quantity. Type (II) devices have the characteristic that the diverter's overall probability of being detected is minimized if he removes material in a large number of small diversions. It is doubtful whether the performance of type (III) devices can be analysed in general objective terms and for this reason they should not be included in a penetration monitoring system where a quantified performance is sought.

Although many devices exist or are under development for the detection of movement of nuclear material through
penetrations, the demonstration of containment integrity by instrumental means presents many complex problems which still remain to be investigated.

3.3. The performance requirements for the individual devices. Performance parameters required include:

- Detection probability as a function of material removed (the "assurance function")
- False alarm probability
- Reliability
- Maintainability
- Verifiability
- Data recording capability
- Tamper protection.

The relevance of these parameters will differ depending on the actual device and in particular on whether it belongs to type (I) or type (II). The specifications required for these parameters, including numerical values where appropriate, must be determined under realistic field conditions. Tamper protection must of course be provided for the entire data chain, from the initial measurements to the final recording.

4. THE QUANTIFICATION OF PROBABILITY OF DETECTION AND FALSE ALARM PROBABILITY

For quantification to be possible, each device must have its own quantified detection probability and false alarm probability and these must be combined in the appropriate manner. Thus the assurance function must be known for each device and, as already mentioned, this effectively excludes the incorporation of type (III) devices. For type (I) devices, the detection probability is effectively 1 for all sizes of diversion. Outside the context of nuclear material accountancy the false alarm probability must be defined carefully; here it is taken to mean the probability of statistical anomaly arising, given that no diversion has occurred (or has been attempted) and with proper design and use should approach zero. For type (II) devices the concepts of detection probability and statistical false alarm probability closely parallel the standard usage in nuclear material accountancy.

The detection probability $P_D$ of the penetration monitoring system can be derived from the separate detection probabilities $P_{d_i}$ of $n$ C/S devices by the relationship:
The conditions are that the separate $p_{dj}$ values are
exhaustive (see 3.1(b)) and exclusive, i.e. there is no
dependence such as may result from correlated systematic
errors). Note that the system $P_D$ is the minimum possible
value achieved by assuming that the diverter chooses the
scenario which maximizes his chances of non-detection.

The false alarm probability $P_F$ of the penetration
monitoring system can also be derived from the separate false
alarm probabilities $p_{fi}$ of $n$ devices:

$$P_F = \sum_{i=1}^{i=n} p_{fi}$$

The condition is that the separate $p_{fi}$ values must be
exclusive, i.e. the occurrence of a false alarm from one
device does not influence the occurrence of a false alarm from
any other device. Where the number of type (II) devices is
large, this could lead to a high system false alarm
probability. However, if the optimum diversion scenario
includes a large number of removals of small quantities of
material through type (II) devices, then $p_{dj}$ for a single
small removal can be reduced considerably while still
maintaining a satisfactory value for the system $P_D$. Since
there is a trade-off between $p_{dj}$ and $p_{fi}$, this reduction
in $p_{dj}$ allows a corresponding reduction in $p_{fi}$, resulting
in a lower value for $P_F$. Non-statistical false alarms
(arising, for example, from instrument malfunction) may
however still present a problem. Whereas $P_D$ is determined
only by the sensitivity of those devices involved in the
worst-case strategy, the system false alarm rate is
determined by the false alarm rates of all installed devices.
Also, $P_D$ cannot be smaller than the value of $p_{di}$ for the
single most sensitive device involved and, if a number of
devices are involved in the diversion route, $P_D$ will be
significantly higher. To avoid an excessive number of false
alarms, the aim would be to minimize the total number of
devices and/or the false alarm rate per device. A reduction
in the total number of devices could be achieved by reducing
the total number of penetrations to be monitored. A reduction
in the false alarm rate per device can also be achieved but
only at the penalty of a lower detection sensitivity in that
device. However, this latter need not affect the system
performance $P_D$ since, provided that only devices not
involved in the worst case diversion strategy have their
sensitivities reduced, the overall system performance is not
affected. This of course will only remain true so long as the reduction of sensitivity in a particular device or devices does not create a new, less sensitive worst case strategy.

5. RESPONSES TO ANOMALIES

The type of response depends very much on the type of device and the type of situation in which it is employed. Thus for type (I) devices an anomaly, if confirmed as such, would require re-inventory of the affected area. This could present severe practical problems for bulk handling facilities and in any case might not be particularly meaningful, since a re-inventory would only establish one term in the MUF equation, leaving the possibility of incorrect values for the opening inventory, input flows and output flows as potential sources of apparent MUF. The problems would be eased if redundant devices were employed and response actions initiated only where anomalies were recorded by all such devices. Also, if the affected area were subdivided into a number of subsidiary self-contained areas, each monitored separately, then this would help limit the necessary response actions. For type (II) devices, it might be possible to ascertain the cause of the anomaly within a very short period of time by direct investigation.

6. SUMMARY OF PROBLEM AREAS

The types of facility or parts of a facility which would lend themselves to the application of penetration monitoring is the most important feature which needs investigation, bearing in mind that the Agency seeks assurance that nuclear material cannot credibly be used to further the manufacture of nuclear explosive devices within the containment boundaries, without detection. In addition, the following technical problems present themselves.

6.1. The number of C/S devices required may be large and is determined by three considerations:

- The selection of containment boundaries. In a real facility the choice may be limited by construction features which may require a large number of penetrations to be monitored.
- The need to identify all penetrations of the containment, which raises questions of accessibility for the inspector.
- The need to designate certain penetrations as constituting realistic diversion routes which need to be monitored.
6.2. The performance requirements for Type (II) detection devices are complex. They include (besides detection and statistical false alarm probabilities) the determination of reliability, maintainability, verifiability, tamper protection and data recording capability. Currently available type (II) devices have been designed primarily for process, safety and security roles where the specification is not usually so complex. Non-statistical false alarms (arising, for example, from instrument malfunction) may still give problems. The specification of a suitable device for demonstrating the integrity of containment boundaries remains to be devised.

6.3. The appropriate responses to anomalies generated by a penetration monitoring system remain to be spelled out. A re-inventory of the affected area could present severe practical difficulties and in any case the uncertainty of the final balance would be determined by the capability of the nuclear material accountancy system. Possible approaches to this problem include: the use of back-up containment with detection devices to provide redundancy; and the subdivision of areas into smaller units to reduce re-inventory problems should they arise.

References


EFFICIENCY OF A MATERIAL ACCOUNTANCY
SAFEGUARDS SYSTEM INCLUDING EXTENDED
CONTAINMENT/SURVEILLANCE MEASURES

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Abstract

EFFICIENCY OF A MATERIAL ACCOUNTANCY SAFEGUARDS SYSTEM INCLUDING EXTENDED CONTAINMENT/SURVEILLANCE MEASURES.

In the International Working Group on Reprocessing Plant Safeguards (IWG-RPS) Overview Report extended containment/surveillance (C/S) measures have been defined as those safeguards measures which attempt to provide assurance that all transfers are through key measurement points (KMPs). In that report it has been suggested that further work on the models on material accountancy (MA) and C/S systems in a complete safeguards approach should be continued at least to the point where well-founded and detailed analyses on the individual assurances and their combinations can be carried out. This paper presents the results of the efforts of various research groups, as well as of the authors, to solve some of the problems indicated above. As shown in earlier papers, under appropriate assumptions the MUF-D test is optimal for a safeguards system based on material balance and data verification principles, where material has been diverted by methods the diverter considers most favourable. In this paper the justification for choosing the MUF-D test and applying it also to C/S measures is explained, as well as its use in detecting shipments of special nuclear material (SNM) through access points not initially declared for this purpose. The analyses made up to now are expanded and the generalized test statistics, together with the optimal detection probability, are determined once again for cases where the diverter uses a strategy which is most advantageous to him. This will show the maximum detection probability increase obtainable by this method.

1. Introduction

When the Non-Proliferation Treaty (NPT) /1/ was signed and came into force in 1968, there existed no nuclear facilities with large throughputs of Special Nuclear Material (SNM) which had to be safeguarded under the terms of the Treaty. Thus, it was only natural that the Material Accountancy (MA) methods used by governmental authorities in nuclear weapons

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plants on the one hand and by the IAEA for peaceful uses in accordance with the provisions of INFCIRC/66 /2/ on the other hand were included in the discussions on the implementation of NPT and incorporated in the IAEA's model agreement /3/. However, the originators of this document were far-sighted enough explicitly to include other methods which they termed CONTAINMENT AND SURVEILLANCE (C/S) and considered IMPORTANT COMPLEMENTARY MEASURES.

Safeguarding practices, the amounts of SNM and the magnitude of measurement errors encountered did not create any serious problems during the first decade of safeguarding. With the advent of commercial reprocessing on a large scale the picture began to change. In the controversy which then developed one of the originators of the model agreement continued to stress /4/ that C/S was intentionally built into the agreement as an important complementary measure since it was considered a realistic solution of the problems which the rapid evolution of nuclear technology could be expected to develop. Later, some publications /5, 6/ showed that the C/S measures possess the same legal status as MA measures. In the meantime numerous variations of MA and C/S have emerged and have been proposed for operational use, but except at the Tokai Mura Reprocessing Plant none of them have been applied so far.

In the light of the above we think it useful to consider some details of the historical and theoretical backgrounds before discussing a combination of MA and penetration monitoring (a form of extended C/S) which may overcome the difficulties in safeguarding large bulk facilities, especially reprocessing plants.

2. Special Issues Involved in Safeguarding Large Amounts of SNM

In 1953 the USA introduced its ATOMS FOR PEACE policy and for the first time in the history of international relations the delivery of SNM and equipment was linked to special safeguards measures. This concept was then enshrined into the IAEA Statute /7/, the IAEA being authorized to demand safeguards measures for such SNM and equipment as was made available by them or put under their supervision.

One of the fundamental principles of the NPT was that the peaceful uses of nuclear power should not be unduly obstructed. This was evident from the fact that safeguards measures were no longer concerned with equipment but only with SNM /3/. The mere transfer of the relevant technology became a sensitive issue,
and thus governments had a strong motive to restrict access by the Agency's inspectors. This was the starting point for the work done in the Nuclear Research Center Karlsruhe /8/, the result of which was the realization that by measuring the SNM flow at a few key measurement points (KMP's) sufficient information on possible diversions could be gained.

As indicated above, the technique of MA was already being used by the nuclear powers and the IAEA. This technique was derived from conventional accounting procedures used in business, where only information extracted from a comparison of book values with physical inventories was considered acceptable. The information results in the so-called Material Unaccounted For (MUF) defined by

\[ \text{MUF}_j = I_{j-1} + R_j - S_j - I_j \]

where \( I_{j-1} \) is the beginning physical inventory for period \( j \), \( R_j \) is the sum of increases in inventory (receipts, nuclear production, de-exemption, corrections to receipts, as appropriate), \( S_j \) is the sum of decreases in inventory (shipments, nuclear loss due to radioactive decay or burn-up, exemption, measured discard, accidental loss, as appropriate), and \( I_i \) is the ending physical inventory for period \( j \), which is also the beginning physical inventory period \( j + 1 \). The value of MUF is determined by the measurement errors, and in the case of large throughputs these will be the dominating components. Thus, in bulk handling facilities MUF could "mask the diversion of sizable proportions" of SNM /4/.

Given their declared aim of safeguarding SNM, the NPT /1/ and the model agreement /3/ call for TIMELY DETECTION OF DIVERSION OF SIGNIFICANT QUANTITIES OF NUCLEAR MATERIAL FROM PEACEFUL NUCLEAR ACTIVITIES ... AND DETERRENCE OF SUCH DIVERSION BY THE RISK OF EARLY DETECTION. In designing large reprocessing plants with throughputs of more than 1,000 tons per year¹ - as was the case with the Gorleben plant /9/ - the limitations of conventional MA must inevitably be faced up to. Selecting two to four material balance areas (MBA's) was considered reasonable by the International Working Group on Reprocessing Plant Safeguards (IWG-RPS) convened by the IAEA 1979 - 1981 /10/. In this case the magnitude of MUF would be about 1% of throughput which would amount to 0.1 tons per year of plutonium for the commonly occurring error probabilities. This is several times more than the minimum quantity considered significant /11/ within the framework of the model agreement. Increasing the number of MBA's and the number of inventory periods in a given year would jeopardize the results owing to the increased number of components and the resulting errors /12/. Increasing the number of shut downs per year to shorten the detection time of abrupt diversions would

¹ Throughout this paper tons are metric tons.
seriously limit the availability of the plant; moreover, in the long run this would not solve the problem of protracted diversion. Thus, in pursuing the detection goals mentioned above, especially in the case of large reprocessing plants the conventional MA technique led to collisions with Art. 5-7 + 78 of the Verification Agreement /13/ which provide that
- operation of the facilities should not be unduly hindered;
- optimum cost-effectiveness should be ensured.

To overcome the difficulties enumerated above several methods have been put forward, e.g. Near Real Time Accountancy (NRTA) /14/ and PIPEX /15/, just to give an idea of the range.

In the case of NRTA it became clear at an early stage that this would require a large number of in-process or even in-line instruments that would need verification. The evaluation of the data produced by them would need new computerized methods, which would have to be developed. But it is still an open question to what extent it would be possible to detect protracted diversion. For verification purposes the necessary C/S measures will have to be provided not only to protect all the measurements but also to detect tampering, and the large number of measurement points appears to constitute the major drawback of this method.

PIPEX may be considered an extreme case of extended C/S which - modified in a given way - appears feasible. For radiation protection and safety reasons all personnel leaving the plant will in any case have to be examined, and so the idea of monitoring the individuals leaving the plant gained ground. Taking also into account the access points for material it should be possible to limit their number to a few which have to be kept under surveillance. Nevertheless, there is an international consensus that conventional MA with periodic physical inventory-taking has to be maintained.

3. Evaluation of Radiation Monitor Data

In the following we restrict our discussion of extended C/S measures to only those measures which give a quantitative response to the transfer or presence of SNM. An important group within this category are radiation monitors, which are able to detect the characteristic γ- or n-radiation of SNM. The data provided by these monitors are by nature of the radiation sources randomly distributed. Therefore, it is necessary to use statistical methods for the evaluation of their capability of detecting the transfer or presence of given amounts of SNM.
In an earlier paper /16/ the distributions of radiation monitor data have been examined carefully, which were experimentally determined in the presence of SNM as well as for pure background radiation. Systematic experiments have been performed where small amounts of Pu in different lead shieldings passed a portal monitor equipped with a γ-detection system. The counting rate for each passage during a fixed time interval was stored together with the counting rate of pure background radiation measurements. The difference of the counting rate data for SNM and pure background may be expressed as a quantity of SNM under appropriate assumptions, e.g. thickness of shielding material. In this way the data of C/S measures may be evaluated with the help of the same formalism as commonly used for MA data, and thus the efficiency of these C/S measures as well as their contribution to the overall system efficiency can be determined.

4. Efficiency of a Combined System for a Given MBA

In order to analyse the efficiency of C/S measures within a given combined MA- and C/S-system, let us consider the following simplified safeguards procedure for the SNM flowing through a given MBA: The material enters the MBA through a LEGAL INPUT ACCESS POINT, and it leaves the MBA through a LEGAL OUTPUT ACCESS POINT. In addition, there exist OTHER ACCESS POINTS which are protected by means of C/S measures and through which material does not flow under normal conditions. For simplicity's sake we also describe as other access points the relevant parts of all pipes penetrating the boundaries of the MBA.

Let us further consider one inventory period and assume that all statements, including those derived from C/S measures are made at the end of the inventory period. The test procedures to be discussed are based on the following three test statistics:

- **MUF** for the MA measures,
- **D** for the data verification and
- **S** for the C/S measures;

here MUF is, as already explained, the algebraic sum of initial inventory, receipts, shipments and ending inventory, as reported by the operator; D is the difference between the data derived from the inspector's independent random sample on the one hand, and the operator's data on the other; and S is the sum of the individual C/S measurement data relative to the background measurement data.

In Refs. /17, 18/ an evaluation procedure is considered, whereby individual tests for all single C/S measures
Table I. Numerical example for the diversion of the quantity M during one inventory period:
Case 1 describes a pure data falsification strategy where only shipment data are falsified, case 2 diversion into MUF, and case 3 a mixture of the two foregoing ones. Measurement errors are neglected.

<table>
<thead>
<tr>
<th></th>
<th>Case 1: $M_1 = 2$, $M-M_1 = 0$</th>
<th>Case 2: $M_1 = 0$, $M-M_1 = 2$</th>
<th>Case 3: $M_1 = M-M_1 = 1$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>true data</td>
<td>reported data</td>
<td>true data</td>
</tr>
<tr>
<td>Initial Inventory $I_0$</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Receipts $R$</td>
<td>100</td>
<td>100</td>
<td>100</td>
</tr>
<tr>
<td>Shipments $S$</td>
<td>48</td>
<td>50</td>
<td>50</td>
</tr>
<tr>
<td>$M_1$</td>
<td>2</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>$M-M_1$</td>
<td>0</td>
<td>0</td>
<td>2</td>
</tr>
<tr>
<td>Ending Inventory $I_1$</td>
<td>50</td>
<td>50</td>
<td>48</td>
</tr>
<tr>
<td>$MUF = I_0 + R - S - I_1$</td>
<td>2</td>
<td>0</td>
<td>2</td>
</tr>
<tr>
<td>$D$</td>
<td>-2</td>
<td>0</td>
<td>-1</td>
</tr>
<tr>
<td>Reported MUF - $D$</td>
<td>2</td>
<td>2</td>
<td>2</td>
</tr>
</tbody>
</table>
(e.g., radiation monitors) are constructed under the assumption that the total diversion is allocated such that the detection probability is minimized. Naturally, this procedure can also be extended to the evaluation of D, MUF and C/S test statistics. Here, information on the allocation of the total diversion is obtained, however, at the cost of reducing the total detection probability, as we shall see.

Let us assume, now, that the operator chooses the following strategy in order to divert the quantity M of SNM: He falsifies the shipment and/or ending inventory data by the quantity M₁ and removes this through other access points from the MBA, thus maintaining an appearance of correctness of the material balance.

It will be clear that where shipment and/or ending inventory data have been falsified, the data of subsequent material balance areas and/or inventory periods will have to be falsified likewise. This will be discussed in the next chapter.

The quantity M-M₁ of SNM must be diverted without any data falsification, just by taking advantage of the inaccuracies of the material balance, and removed via other access points. Table I illustrates this diversion strategy, omitting measurement errors for the sake of simplicity.

Under the null hypothesis H₀ (no diversion of SNM) and under the alternative hypothesis H₁ (diversion of the quantity M of SNM) the expected values of D, MUF and S according to our assumptions are

\[ E(D, MUF, S) = \begin{cases} (0, 0, 0) & \text{under } H₀ \\ (-M₁, M-M₁, M) & \text{under } H₁. \end{cases} \]

The Neyman-Pearson test (N-P test) maximizes the total detection probability for a given false alarm probability (see e.g. /19/). It aggregates the three test statistics D, MUF and S into a single one which means that an inspector cannot determine how much of the total quantity diverted is ascribable to data falsification (M₁) and how much to diversion into MUF (M-M₁).

In order to be able to construct the N-P test, we have to know the common probability density \( f(t) \) of the random vector \( t' = (D, MUF, S) \) under the two hypotheses \( H_i, i = 0,1 \). In case of -
MODEL A, i.e. where under $H_1$ all data of all batches are falsified by class specific quantities (see, e.g. /20/), it is given by the following formula

$$f_i(t) = \frac{3}{2\pi} \cdot |\Sigma_i|^{-\frac{1}{2}} \cdot \exp \left(-\frac{1}{2} \cdot (t - E'_i)^\prime \cdot \Sigma^{-1}_i \cdot (t - E'_i) \right)$$

where

$$E'_i = (E_i(D), E_i(MUF), E_i(S)), i = 0,1$$

is the expected vector under $H_1$, $i = 0,1$, where the variances

$$\text{var}(D) = \sigma_D^2, \text{var}(MUF) = \sigma_{M/U}^2, \text{var}(S) = \sigma_S^2$$

are assumed to be known, and where because of

$$\text{cov}(MUF, D) = \sigma_{M/D}^2$$

(see, e.g. /20/) the covariance matrix $\Sigma$ has to following form

$$\Sigma = \begin{pmatrix}
\sigma_D^2 & \sigma_{M/U}^2 & 0 \\
\sigma_{M/U}^2 & \sigma_M^2 & 0 \\
0 & 0 & \sigma_S^2
\end{pmatrix}.$$

The critical region $K$ of the N-P test, i.e. the set of observations $t$ of the random vector $\underline{I}$, which leads to the rejection of $H_0$ is given by

$$K = \left\{ t : \frac{f_1(t)}{f_0(t)} > k \right\} = \left\{ t : t' \cdot \Sigma^{-1} \cdot E_1 > k' \right\},$$

where $k$ and $k'$ are determined by the false alarm probability.
Because of
\[ E (\mathbf{1}' \cdot \Sigma^{-1} \cdot \mathbf{1} - 1) = \begin{cases} 0 & \text{for } H_0 \\ E_1' \cdot \Sigma^{-1} \cdot E_1 & \text{for } H_1 \end{cases} \]

and furthermore,
\[ \text{var} (\mathbf{1}' \cdot \Sigma^{-1} \cdot \mathbf{1}) = E_1' \cdot \Sigma^{-1} \cdot E_1, \]

the detection probability \( 1 - \beta_{\text{NP}} \), defined by
\[ 1 - \beta_{\text{NP}} = \text{prob} (\mathbf{1} \in K | H_1), \]

as a function of the false alarm probability \( \alpha \), defined by
\[ \alpha = \text{prob} (\mathbf{1} \in K | H_0), \]

is given by
\[ 1 - \beta_{\text{NP}} = \phi \left( \sqrt{E_1' \cdot \Sigma^{-1} \cdot E_1 - U_{1-\alpha}} \right), \]

where \( \phi \) is the Gaussian distribution and \( U \) its inverse.

* In order to obtain the guaranteed detection probability \( 1 - \beta_{\text{NP}} \), i.e., the minimum detection probability, with respect to all \( H_i \) of \( H_1 \), we only have to determine the minimum of the quadratic form
\[ E_1' \cdot \Sigma^{-1} \cdot E_1, \]

which leads to
\[ 1 - \beta_{\text{NP}}^* = \phi \left( M \cdot \sqrt{\frac{1}{\sigma_D^2 - \sigma_M^2} + \frac{1}{\sigma_S^2}} - U_{1-\alpha} \right). \]
The optimal strategy from the operator's point of view is

\[ M^*_1 = M, \quad M - M^*_1 = 0, \]

i.e., a pure data falsification strategy, as in the case of MUF and D alone /20/. The statistic pertaining to this guaranteed detection probability, unbiased with respect to M, is

\[
\frac{1}{\sigma_D - \sigma_M^2} \cdot \left( \frac{1}{\sigma_D} \cdot (MUF - D) + \frac{1}{\sigma_S} \cdot S \right).
\]

While it is difficult to interpret the optimal strategy of the operator by intuition, one can understand this test statistic, given \( M^*_1 = M \). In fact, if one determines the linear statistic \( a_1 \cdot D + a_2 \cdot MUF + a_3 \cdot S \) for \( E(D,MUF,S) = (-M,0,M) \), unbiased with respect to M and with minimum variance, one obtains the statistic given above.

This test statistic has the expected value \( M \) for any value of \( M^*_1 \). From this one concludes that the strategy of the inspector, choosing the N-P test, and the strategy \( M^*_1 = M \) of the operator are saddlepoint strategies in the two person zero sum game with the detection probability as a pay-off for the operator.

Naturally, the detection probability \( 1 - \beta^*_\text{NP} \) given above is greater than the corresponding one without C/S measures \( (\sigma_S^2 \to \infty) \); it approaches 1 as \( \sigma_S^2 \) approaches 0. This does not mean, however, that the data verification and MA measures can be neglected: If there were no measures of that kind, then the operator could divert all material through the legal access points without being detected by the inspector.

5. Efficiency of a Combined System for Two MBA's

We have already mentioned that where a falsification of material balance data for one inventory period and one MBA occurs there exists a further possibility of detection by reviewing
the data from an earlier or subsequent inventory period or MBA. In this chapter we shall determine the increase in detection probability offered by this possibility.

Let us consider the following simplified safeguards procedure for the SNM which flows through 2 MBA's in succession (e.g. a reprocessing and a fuel fabrication plant). The input SNM enters the first MBA through a legal input point and leaves it through a legal output point. Thereafter it flows in the same way through the second MBA. We shall assume that in both MBA's there exist other access points which are protected by C/S measures, and through which material does not flow under normal conditions.

We further assume that both MBA's have inventory periods with the same starting dates and duration, and that only one statement for both MBA's is prepared at the end of the inventory period.

In practice there will of course be a separate statement for each MBA. The intention here is to determine the optimal test procedure in the sense of the overall detection probability.

The test procedures now to be discussed are based on the following six test statistics for the two material balance areas:

- $M_{UF1}$ and $M_{UF2}$ for the MA measures,
- $D_1$ and $D_2$ for the data verification and
- $S_1$ and $S_2$ for the C/S measures.

We shall assume that shipments out of the first and receipts into the second MBA are measured and verified independently. Thus, $M_{UF1}$ and $M_{UF2}$ and also $D_1$ and $D_2$ are independent of one another. Therefore, the covariance matrix of these six random variables given above is

$$
\begin{pmatrix}
\Sigma_1 & 0 \\
0 & \Sigma_2
\end{pmatrix}
$$

where 0 is a 3x3 matrix with zero elements, and $\Sigma_1$ and $\Sigma_2$ correspond to the covariance matrix of the foregoing chapter, with the components

$$
\text{var} (D_i) = \sigma_{D_i}^2, \quad \text{var} (M_{UF0}) = \sigma_{M_i}^2, \quad \text{var} (S_i) = \sigma_{S_i}^2, \quad i = 1, 2.
$$
Table II. Optimal distribution of the total diversion $M$, guaranteed probability of detection and unbiased test statistics for two consecutive material balance areas (MBA's) for the case that diversion, data falsification, and actual removal are limited to these two MBA's.

$\text{var} (\text{MUF}_1) = \sigma_{M_1}^2$, $\text{var} (D_i) = \sigma_{D_i}^2$, $\text{var} (S_i) = \sigma_{S_i}^2$, $N_i = \sigma_{D_i}^2 - \sigma_{S_i}^2$, $i = 1,2$.

<table>
<thead>
<tr>
<th>Case</th>
<th>Random Vector X</th>
<th>Expectation Vector under $H_1$</th>
<th>Optimal Distribution of the total Diversion M</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Case A</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>No C/S-Measures; Diversion only in first MBA</td>
<td>$D_1$</td>
<td>$-M_1$</td>
<td>$M_1^* = M - \frac{\sigma_{M_1}^2}{\sigma_{M_1}^2 + \sigma_{M_2}^2}$</td>
</tr>
<tr>
<td></td>
<td>MUF$_1$</td>
<td>$M-M_1$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>$D_2$</td>
<td>$M_1$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>MUF$_2$</td>
<td>$M_1$</td>
<td></td>
</tr>
<tr>
<td><strong>Case B</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>No C/S-Measures; Diversion in both MBA's</td>
<td>$D_1$</td>
<td>$-M_1$</td>
<td>$M_1^* = M - \frac{\sigma_{M_1}^2}{\sigma_{M_1}^2 + \sigma_{M_2}^2}$</td>
</tr>
<tr>
<td></td>
<td>MUF$_1$</td>
<td>$M_2$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>$D_2$</td>
<td>$M_1 + M_3$</td>
<td>$M_2^* = M - \frac{\sigma_{M_1}^2}{\sigma_{M_1}^2 + \sigma_{M_2}^2}$</td>
</tr>
<tr>
<td></td>
<td>MUF$_2$</td>
<td>$M_1$</td>
<td></td>
</tr>
<tr>
<td><strong>Case C</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>C/S-Measures; Diversion only in first MBA</td>
<td>$D_1$</td>
<td>$-M_1$</td>
<td>$M_1^* = M - \frac{\sigma_{M_1}^2}{\sigma_{M_1}^2 + \sigma_{M_2}^2}$</td>
</tr>
<tr>
<td></td>
<td>MUF$_1$</td>
<td>$M-M_1$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>$S_1$</td>
<td>$M$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>$D_2$</td>
<td>$M_1$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>MUF$_2$</td>
<td>$M_1$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>$S_2$</td>
<td>$0$</td>
<td></td>
</tr>
<tr>
<td><strong>Case D</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>C/S-Measures; Diversion in both MBA's</td>
<td>$D_1$</td>
<td>$-M_1$</td>
<td>$M_1^* = M - \frac{\sigma_{M_1}^2}{\sigma_{M_1}^2 + \sigma_{M_2}^2}$</td>
</tr>
<tr>
<td></td>
<td>MUF$_1$</td>
<td>$M_2$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>$S_1$</td>
<td>$x \cdot M_1 + M_2$</td>
<td>$M_2^* = M - \frac{\sigma_{M_1}^2}{\sigma_{M_1}^2 + \sigma_{M_2}^2}$</td>
</tr>
<tr>
<td></td>
<td>$D_2$</td>
<td>$M_1 + M_3$</td>
<td>$x = \frac{\sigma_{S_1}^2 + \sigma_{S_2}^2 - \sigma_{M_1}^2 - \sigma_{M_2}^2}{\sigma_{S_1}^2 + \sigma_{S_2}^2 - \sigma_{M_1}^2 - \sigma_{M_2}^2}$</td>
</tr>
<tr>
<td></td>
<td>MUF$_2$</td>
<td>$M_1$</td>
<td></td>
</tr>
<tr>
<td></td>
<td>$S_2$</td>
<td>$(1-x) \cdot M_1 + M_3$</td>
<td></td>
</tr>
<tr>
<td>Guaranteed Probability of Detection</td>
<td>With respect to total Diversion M unbiased Test Statistics</td>
<td></td>
<td></td>
</tr>
<tr>
<td>------------------------------------</td>
<td>----------------------------------------------------------</td>
<td></td>
<td></td>
</tr>
<tr>
<td>( \Phi \left( M \cdot \sqrt{ \frac{1}{N_1} + \frac{1}{\sigma_{M_1}^2 + \sigma_{M_2}^2} - U_{1-\alpha}} \right) )</td>
<td>( \frac{\sigma_{M_1}^2 + \sigma_{M_2}^2}{\sigma_{M_1}^2 + \sigma_{D_2}^2} \cdot D_1 + \frac{\sigma_{M_1}^2 - \sigma_{M_1}^2}{\sigma_{M_1}^2 - \sigma_{D_2}^2} \cdot MUF_2 )</td>
<td></td>
<td></td>
</tr>
<tr>
<td>( \Phi \left( M \cdot \sqrt{ \frac{1}{N_1 + N_2} + \frac{1}{2 \sigma_{M_1}^2 + \sigma_{N_2}^2} - U_{1-\alpha}} \right) )</td>
<td>( MUF_1 + MUF_2 = \frac{2}{\sigma_{D_1}^2 + \frac{1}{2 \sigma_{D_2}^2}} \cdot (D_1 + D_2) )</td>
<td></td>
<td></td>
</tr>
<tr>
<td>( \Phi \left( M \cdot \sqrt{ \frac{1}{N_1} + \frac{1}{2 \sigma_{M_1}^2 + \sigma_{M_2}^2} + \frac{1}{2 \sigma_{S_1}^2} - U_{1-\alpha}} \right) )</td>
<td>( \left[ \frac{1}{N_1 + \sigma_{M_1}^2 + \sigma_{M_2}^2} \right] \cdot MUF_1 )</td>
<td></td>
<td></td>
</tr>
<tr>
<td>( \frac{1}{\sigma_{M_1}^2 + \sigma_{M_2}^2} + \frac{1}{2 \sigma_{S_1}^2} \cdot MUF_2 )</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>( \Phi \left( M \cdot \sqrt{ \frac{1}{N_1 + N_2} + \frac{1}{\sigma_{M_1}^2 + \sigma_{M_2}^2} + \frac{1}{2 \sigma_{S_1}^2 + \sigma_{S_2}^2} - U_{1-\alpha}} \right) )</td>
<td>( \left[ \frac{1}{N_1 + N_2 + \sigma_{M_1}^2 + \sigma_{M_2}^2} \right] \cdot (MUF_1 + MUF_2) )</td>
<td></td>
<td></td>
</tr>
<tr>
<td>( \frac{1}{\sigma_{M_1}^2 + \sigma_{M_2}^2} + \frac{1}{2 \sigma_{S_1}^2 + \sigma_{S_2}^2} \cdot (S_1 + S_2) )</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Four cases in ascending order of complexity should exemplify the scale of the problem:

A: no C/S measures, diversion only in the first MBA,  
B: no C/S measures, diversion in both MBA's,  
C: C/S measures, diversion only in the first MBA and  
D: C/S measures, diversion in both MBA's.

Contrary to the assumptions in the previous chapter we shall now assume that the data are falsified in such a way that there is an impact neither on the preceding MBA's or the inventory periods nor on the subsequent ones. In other words, we assume that the falsification is limited to the two MBA's and the single inventory period under consideration. This means that only the shipment data of the first and the receipt data of the second MBA is falsified consistently. The reason for this assumption is that we want to analyse the additional possibility of detecting a falsification in the first MBA arising from the operator's consistency problem; this would not be possible if the falsification were carried forward to other MBA's.

As in the previous chapter we construct once again the N-P test and consider only the data falsification model $A$. The results of the analyses for the four cases A, B, C, D are summarized in Table II. The first column contains the test statistics. The second column shows the expected values of these test statistics under the alternative hypotheses $H_1$ (diversion of the total quantity $M$ of SNM); this expected vector represents the diversion strategy chosen by the operator. It is important to realize that falsification of the data from the first MBA can be detected not only by verifying the data in the second MBA, falsified consistently with those of the first one, but also by establishing the material balance for the second MBA, as the falsified receipt data cause a deficit in the latter's balance. In the third column the optimal diversion strategy (i.e. that diversion strategy which minimizes the detection probability) is given, in the fourth column the guaranteed probability of detection, and in the fifth column the resulting test statistic.

Instead of discussing all details of these formulae, a few general comments are offered:

i) As in the previous chapter, the test statistics given in the fifth column have the expected values $M$ for any strategy resulting in the total diversion $M$. Thus, the inspector's strategy of choosing the N-P test, and the diversion strategy given in the third column are SADDLEPOINT STRATEGIES in the two person zero sum game with the detection probability as a pay-off for the inspector.
ii) Contrary to the previous chapter, in none of the four cases is the best test statistic composed of MUF_i-D_i statistics, i = 1,2. Perhaps as a result of this, in none of the four cases is the optimal diversion strategy a pure data falsification strategy.

iii) The detection probability in case A is greater than where only one inventory period is involved. Even though it is nontrivial, this is to be expected because of the additional measurement errors introduced by the second MBA.

iv) Naturally, the detection probability in case B is smaller than that in case A, since the operators have more opportunities to allocate the diversion between the two MBA's. It depends, however, on the parameters, whether or not this detection probability is greater than where only one inventory period is involved.

v) The addition of C/S measures (cases C and D) to those of the cases A and B will give an appreciable increase in guaranteed detection probability. Even though the timeliness criterion enters the analysis via the length of the inventory period only and some idealizing assumptions have been made, the results show quantitatively how the efficiency of a pure material accountancy data verification system can be improved by introducing extended C/S measures.

6. Summary

Conventional MA should be complemented in order to give reliable results in safeguarding large reprocessing plants. Of the two methods thought feasible at the time being, NRTA or extended C/S, the latter one has been investigated in some detail; the approach presented is a means of combining the quantitative results derived from MA and from the special case of radiation monitors. In this sense it may be considered a first step in the direction indicated by the IWG-RPS "... further work on the model of MA and C/S systems, in a complete safeguards approach, should be continued at least to a point where well founded and detailed analyses on the individual assurances and their combinations can be carried out" /10/.

So far the only objective was the overall detection probability, given a certain MBA and inventory period structure. This means that neither the optimal number of MBA's nor the optimal number of inventory periods for one plant and one year has been considered so far. Furthermore, the time needed to
detect the diversion of a given quantity of SNM was no objective; before taking this into account, ideas should be developed on how to reconcile the tradeoffs between safe and early detection which are known to be frequently in conflict with one another.

References

/1/ Treaty on the Non-Proliferation of Nuclear Weapons; available as IAEA Document INFCIRC/140, IAEA, Vienna 1970.


/3/ The Structure and Content of Agreements between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons; INFCIRC/153, IAEA, Vienna 1971.


/13/ Gesetz zu dem Obereinkommen vom 5. April 1973 (Verifikationsabkommen); Bundesgesetzblatt 1974 II, S 794 ff.


SAFEGUARDS SEALS FOR UNDERWATER SPENT-FUEL STORAGE
The principles and the progress towards a practical system for CANDU reactors

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Abstract

SAFEGUARDS SEALS FOR UNDERWATER SPENT-FUEL STORAGE: THE PRINCIPLES AND THE PROGRESS TOWARDS A PRACTICAL SYSTEM FOR CANDU REACTORS.

While seals are in widespread use for safeguards, the development of a completely satisfactory underwater seal for irradiated fuel has not been straightforward. One aspect of the problem is that, to produce a seal which is (ideally) impossible to duplicate requires a non-reproducible identity. Equally challenging is to provide a simple means of accurately comparing an underwater seal with a reference. The Canadian seal programme is producing seals using two different inspection technologies, ultrasonics and optics. Ultrasonics lends itself to relatively simple measuring tools, but accurate comparative measurements are difficult. The design now proposed for the CANDU ultrasonic cap seal uses a stainless-steel wire coil as the identity element. Tools to install and read the coil seal have been produced and the experimental results obtained with laboratory equipment have confirmed that the seal has real potential for IAEA use. Optical inspection techniques are well developed, but simple tools have yet to be demonstrated for the application. Progress on the optical seal has been in the area of selecting an identity which cannot be duplicated but can be readily examined. Millimetre-sized crystals have been grown in pure zirconium. These are irregular in shape, brightly coloured and the patterns can be readily matched.

1. UNDERWATER SEALS

Of the nuclear materials under safeguards today, stored irradiated fuel represents the greatest inventory. Most of this is in water-filled pools. While materials accountancy practices

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can be, and are, applied to this fuel, containment and surveillance (C/S) techniques are also in widespread use. CANDU reactor safeguards systems include redundancy to maximize safeguards effectiveness, and thus the irradiated fuel storage pools permit materials accountancy, optical surveillance and (safeguards) sealed containment (Fig. 1). The C/S system will first be demonstrated in the CANDU 600 MW reactors, which could show the way for other stored fuel facilities.

Although the IAEA has significant experience with seals for safeguards, the development of a completely satisfactory underwater seal has not been easy. Put simply, it is difficult to examine carefully for signs of tampering at a distance of 5 m, underwater and in a radiation environment. It has also become evident that the principles of seals are neither well documented nor widely known, and this has hampered seal development in
This paper therefore presents the current status of the Canadian seal development program, and mentions the important seal principles which are becoming clearer as the subject is more widely researched.

2. PREVIOUS SEAL DEVELOPMENT WORK

Following early work by Euratom (Ispra), techniques borrowed from ultrasonic non-destructive testing technology appeared suitable for underwater sealing systems [1]. They offer the possibility of remotely scanning welds or inclusions in metal which can form a unique seal signature. Moreover, NDT technology seems a good candidate for interrogating the integrity of the container to which the seal is attached.

The first seal for irradiated CANDU fuel was an adaptation of Euratom's mechanically scanned ultrasonic seals [1]. While satisfactory in concept, it proved too difficult to install and read to be considered for use in large storage pools. To eliminate the handling difficulties, AECL and Euratom jointly developed a cap seal [2]. Rather than mechanically driving an ultrasonic transducer across a pattern of inclusions in metal, the seal signature consisted of the time-amplitude reflection pattern from internal inclusions at various depths within the seal body (Fig. 2). Signature repeatability, particularly with different interrogating transducers, was never conclusive. Turning adversity to advantage, Euratom incorporated the variable transducers as part of the signature in an "Integrated Seal" [3]. At the same time, AECL started a broadly-based seal program, including a re-examination of the choice of ultrasonic NDT technology for safeguards seals and the desirability of in-situ verification.

FIG. 2. Cap seal incorporating internal occlusions which provide a unique identity.
3. REQUIREMENTS AND PERFORMANCE TARGETS

Since the level of technology and resources available to an adversary cannot be known with certainty, it is not possible to write an all embracing seal specification that will serve either the immediate or long-term future. In fact it must be assumed that the technology available to an adversary will advance with time. Consequently, seals may have to be improved as time passes, in anticipation of more sophisticated defeat methods. Thus although the requirements and performance of a seal can be stated, they can only be stated as targets.

The requirements of the spent-fuel safeguards seals include those which are generic to all seals, those which stem from IAEA practices and those specific to a particular facility design.

Used CANDU reactor fuel is discharged as radioactive bundles, 50 cm long and 10 cm in diameter. A CANDU 600 MW reactor discharges approximately 100 bundles per week. The bundles are stacked on trays, to be covered with lids which will be sealed. The seal is intended to be attached to a rod which extends from the bottom of the stack, through the space between the bundles, and through the lid. The seal will be approximately 4.5 m below the water surface (Fig. 1).

The radiation dose rate on contact with fuel (after six months in storage) is $10^3$ Gy/h ($10^5$ rad/h). This decays with time so that a total dose of $10^7$ grays (or $10^9$ rads) can be expected at a seal near the fuel after 30 years (the expected fuel storage period).

The water in the pool is held within narrow temperature, pH and purity ranges. Visibility through the water is good, impeded only by surface ripples and heat-induced refractive index fluctuations near very recently discharged fuel.

In addition to requirements imposed by the environment, those which are generic to safeguards seals are:

1. The container and seals must include feature(s) that make repair and duplication as difficult as is reasonably achievable. (It is recognized that seals proposed cannot be judged against absolute standards, only against other seals, and even then on a subjective basis.)

2. The cost of applying and verifying the seal must be less than the cost of verifying the contents. Cost of verifying the contents of a stack of 1500 spent fuel bundles, even on a random basis, would likely exceed $10^6$.

3. Seal verification must ideally be unambiguous, to minimize diversion risk and false alarm rate.
(4) A permanent record of the verification is required.

(5) Verification will take place, at most, once every two months. The seal must therefore withstand at least 60 inspections for a 10 year life.

(6) Verification procedures must be simple enough for non-technical inspectors whose duties frequently change.

(7) The seal should preferably be verifiable in situ, i.e. verifiable without being destroyed.

(8) The method for verifying the seal must be compatible with and complimentary to methods for verifying the integrity of the container.

From these requirements can be deduced certain design principles:

(1) Seals are vulnerable to defeat by either duplication or repair.

(2) To produce a seal which is impossible to duplicate requires a unique non-reproducible identity. Candidates for such fingerprints are signatures based on processes which are extremely difficult to control, and provide an extremely large quantity of information.

(3) Although the seal signature should contain much information, it must be simple to measure and accurately compare with a reference signature. This can be achieved by comparing all the signature information, or a selected part, known only to the inspector.

(4) Signatures which require control over many parameters and those requiring mastery of several technologies will be more difficult to reproduce.

(5) As a consequence of (1) above, it also follows that the integrity and identity should not be capable of repair or duplication independently, and that both should be readily inspectable. To this end, seals in which the identity is destroyed when the integrity is broken provide protection against repairing and duplicating the seal.

(6) The goal of making a seal readily inspectable leads to a further important design principle. For an in-situ verifiable seal, the break-point should be well defined and readily accessible to inspection.
It would be desirable if the performance of a seal could be quantified by a few key specifications. Unfortunately, there seems to be no formally agreed procedure. Canada is collaborating with Euratom (Ispra) on standards for presenting seal specifications and recommends that other seal developers join such efforts. Probably the most that can be done now is to compare various seals (possibly subjectively) on the basis of:

(a) cost of producing a duplicate seal;
(b) time required to produce a duplicate seal;
(c) probability of false positive alarms, requiring inventory reverification;
(d) probability of failure to detect damage to integrity or change in identity;
(e) cost of verifying seal (including manpower, radiation exposure, etc.);
(f) simplicity of installation.

Of these, the difficulty of duplication is clearly very important, as is the probability of false alarms. Consequently, the R&D programs have placed emphasis on making estimates and optimizing these parameters.

The Canadian seal program is producing seals based on these general principles using two different inspection technologies, ultrasonics and optics. The two types of seals may, in fact, be used together, increasing the technological demands on an adversary and using redundancy to protect against possible false alarms. Limiting the program to two technologies has been based on cost constraints, not on technical grounds.

\[ T_1 = \text{TRANSDUCER FOR INTERROGATING IDENTITY AND INTEGRITY} \]

**FIG. 3.** Cap seal incorporating a random coil which provides a unique identity and evidence of its integrity.
4. ULTRASONIC SEAL PROGRAM

The decision to pursue ultrasonics technology was made because there is a large body of ultrasonic seal expertise being built up in the IAEA and its safeguards support programs. Canadian work on ultrasonic seals over the past year has focussed on the random coil seal [4] (Fig. 3). It was developed to eliminate some fundamental difficulties with the Euratom internal-signature cap seal [2], viz.

- signatures produced by reflections from metal inclusions were unsuitable. For many signature elements made there was relatively little information present. It has also been shown that the signatures may be temperature-sensitive.

- signatures produced by secondary (mode-converted) reflections were insufficiently reproducible, particularly if read by different transducers. These signatures are also unacceptably sensitive to the location of the transducer.

- when small diameter interrogating transducers were used to ameliorate the above problem, the break-pin could be removed and replaced without destroying the signature.

The design¹ now proposed for the CANDU ultrasonic cap seal uses a stainless steel wire coil as the identity element. This coil is incorporated in a cavity in the top of the seal. Since it is in water, it is directly accessible from the interrogating transducer. The importance of this is that the signature is produced by primary reflections. The measured signature is a function of ultrasonic frequency, transducer location and timing window. Hence, there is a great deal of information in the signature, as is required. It is intended that the IAEA select specific values for these parameters, but retain the option to use as many as needed to achieve the confidence they desire.

A second important feature is that the coil is attached to the seal body such that it will be distorted if the seal is broken. Thus the identity and integrity features are linked.

Tools to install and read the coil seal have been produced and the experimental results obtained with laboratory equipment show that the seal has real potential for Agency use. The signatures of 36 seals have been measured, in storage-bay conditions, using laboratory equipment [6]. Signatures are recorded as time-amplitude patterns both by photographing an oscilloscope screen, and by direct digitization. Comparisons can be made visually, or by calculating correlation coefficients between the

¹ Patent applied for [5].
Each seal was measured once with each of 3 transducers, but correlation coefficients for signatures measured with different transducers are not included. Total = $3^C_2 \times 36 \times 3 = 1890$ coefficients

Each seal was measured 5 times with 3 different transducers, but correlation coefficients for signatures measured with different transducers are not included. Total = $5^C_2 \times 36 \times 3 = 1080$ lost data points

= 1050 coefficients

**FIG. 4.** Distributions of correlation coefficients of pairs of signatures (from 36 seals).

digitized patterns. Successive measurements of a single seal are highly correlated while signatures of different seals show little correlation. The difference in correlation coefficient between repeated measurements of a single seal and those from non-identical seals is sufficient to make the assessment that there is a large population of measurable different seals (Fig. 4) [4].

A portable non-destructive testing instrument can provide photographs of signatures for visual comparison. While this leaves the inspector with a subjective judgment when comparing seals, it has the advantage of being commercially available.

In order to provide a convenient quantitative method for inspectors to determine signatures, the U.S.A. (Sandia Laboratories) is developing an automated reading instrument. This provides the pulse train to excite the ultrasonic transducer,
and the circuit to process the reflected signals. The instrument stores reference signatures in plug-in bubble memories and the correlation between reference signals and inspection signals is calculated and displayed along with the signature itself. Work has been started on five readers for field trials.

To summarize, the ultrasonic seal appears sufficiently promising to proceed to make tooling and readout units, train IAEA staff and field test the system in a spent fuel pool. Ultrasonics is a technology that lends itself to relatively simple measuring tools, but even with state-of-the-art equipment, accurate comparative measurements remain difficult. Efforts (by the development team) to subvert the seal have failed, but the IAEA has yet to assess the vulnerability of the seal. No attempt has been made to apply ultrasonic technology to the interrogation of parts of the fuel storage racks as yet.

5. OPTICAL SEAL PROGRAM

The incentive for looking at seals which can be interrogated visually is that optical inspection methods are the most promising for verifying the integrity of the fuel storage container. If it is necessary to provide inspectors with a telescope or an underwater video camera, it may also be possible to use these instruments to verify the seal.

Progress on the optical seal has been in the areas of

- selecting an identity which cannot be duplicated;

- designing the seal so that its identity and integrity would be destroyed together and would be visible from the surface of the storage pool, and

- specifying and procuring a telescope and photographic equipment suitable for laboratory evaluation of the seals.

The most promising identity developed is based on growing millimetre-sized crystals in pure zirconium using well-known metallurgical techniques. The crystals have random shapes, and when oxidized, the surface polarizes reflected light to produce a series of different colors, akin to a geo-political map. The identity will be destroyed if the seal is removed, because the break point has been incorporated into the seal face, which can be examined from the water surface (Fig. 5). Photographs, taken from the surface with commercial photographic and lighting equipment, are sufficiently promising to proceed to develop a simple-to-use optical viewing system. The photographs provide vast amounts of information, but by superimposing and subtracting images, the eye easily spots differences.
FIG. 5. Cap seal incorporating random crystal growth in the metal surface to provide a unique identity and evidence of its integrity.

It is, as yet, rather early to predict whether this sealing system will be suitable for routine use by the IAEA, although there is little doubt that the identity is very suitable for seals of various types.

6. OTHER SEAL OPTIONS

The history of seal development shows that there is no shortage of ideas for signatures or non-return mechanisms. It is the techniques for examining seals (in situ and at a distance) that present the challenge. This leads to questioning the requirement for in-situ verification. Designs for single-use seals which are removed from spent fuel stacks for verification are clearly possible and are parallel (in concept) to the traditional Agency E-type seal. Ultimately, these will have to be compared with in-situ verifiable seals, to determine which provides the higher degree of confidence.

7. CONCLUSIONS

A seal is, ultimately, only of use to those who have confidence in it. Once it is in use, its owner must remain aware of advancing technology that may enable divertors to defeat it. It would be quite presumptuous to believe that any seal is invulnerable to defeat and it is therefore highly desirable that the IAEA have a number of seals available for each application.

The AECL Random Coil Ultrasonic Seal and the Euratom Integrated Ultrasonic Seal are now ready for vulnerability testing and trials under field conditions. Tooling for applying and interrogating Random Coil seals in spent fuel pools has been designed. A readout which is suitable for comparing electrical wave forms with reference signatures is also nearing completion. Closely following these developments, alternative seals based on optical verification will likely be available to the IAEA.
ACKNOWLEDGMENT

This work is being undertaken as part of a co-operative programme with the IAEA and the Atomic Energy Control Board of Canada.

REFERENCES


LINKS BETWEEN DIFFERENT ULTRASONIC TECHNIQUES PROPOSED FOR UNIQUE IDENTIFICATION

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Abstract

Over the last two years, several research establishments involved in R&D for international safeguards have been investigating the possibility of using the “ultrasonic signature” principle to identify uniquely such items as containers and fuel elements or bundles. The ways in which general principles are applied always appear to be very different. A careful systems analysis shows that the solution chosen can be inserted in a general scheme resulting in three major approaches involving only two different families of electronic devices for the identity, pick-up, and treatment.

1. INTRODUCTION

Since 1969 ultrasonic techniques, which are in general use for NDT, have been proposed for safeguards purposes. The general objectives of such techniques are:

- The unique identification of items, such as seals or structures containing fissile material; and
- Verification of the integrity of items such as seals and containers.

Most techniques developed up to now involve:

- One (or several) ultrasonic transducer which transmits ultrasonic radiation into the structure to be identified;
- A standard ultrasonic “instrument” or part of one; and
- Various recording devices and data treatment procedures.

The different ways in which the transducer transmits the ultrasonic waves into the structure, the different structures involved, the possible selection of echoes
received by the equipment and variations in the recording and treatment of information, mean that different applications appear essentially as different ultrasonic techniques. However, through harmonization of the data treatment, standardized readout equipment can be used.

2. TECHNIQUES DEVELOPED AND THEIR MAJOR APPLICATIONS

2.1. Ultrasonics: A choice made in 1970

2.1.1. When the problem of the unique identification of items (structures containing fissile material) arose, investigations were carried out in several directions related to non-destructive examinations (NDE).

2.1.2. An X-ray method was the first technique proposed, for the particular case of the Material Testing Reactor fuel plate, the sandwich of which was characteristic (‘dog-boning’ of the fuel).

X-ray fluorescence was also tested and proposed for general-purpose seals, indicating considerable possibilities for identifying a wide variety of designs.

These X-ray techniques in general were not used for safeguards C/S applications because of the bulky nature of the equipment and also because of the danger involved during their application. But today such an attitude would be unjustified.

2.1.3. Electromagnetic techniques were also tested — the standard eddy current instrumentation was used on MTR plates (JRC, Ispra), seals and welds (Sandia Laboratories), but the considerable effect of surface damage (scratch corrosion) on the identity pattern made it unusable for MTR plates. Nuclear plant components control has required further development of eddy current techniques and nowadays the “surface effect” can be subtracted. Multi-frequency systems are used and should be considered for safeguards applications.

Electromagnetic transducers (EMAT) are important for structure interrogation and could replace piezoelectric transducers in several instances.

2.1.4. Visual techniques were of course investigated but external marks do not appear to be tamper-resistant.

Special optical solution and external marks are now being reconsidered.

2.1.5. Ultrasonic techniques appeared to offer the best compromise and the following advantages:

Internal or external detection of marks, and thus those techniques are applicable for identifying internal marks or reading external marks.
Possible combination of unique identification and integrity checks, owing to the specific character of the technique used for NDE of structures.

Easy adaptation of this industrial technique, for which experience exists and the equipment (electronics and sensors) is commercially available.

Portability of the instruments is proven because of industrial development for over 30 years.

A wide range of possible adaptations, in principle:
With or without scanning of the item;
With focused or widespread ultrasonic beams;
At a long distance or on small pieces;
Under water, under sodium or in dry air (contact);
On irradiated material (gamma radiation);
In temperature up to 300°C.

Although many difficulties appear when actual applications are considered, the advantages given above were decisive in giving priority to the ultrasonic technique for item identification and integrity check in 1968 at JRC, Ispra and in other laboratories during the 1970s (Sandia Laboratories, United States of America; AECL, Canada; AERE, Harwell; UKAEA, United Kingdom etc.).

2.2. The principle of major ultrasonic signature techniques

2.2.1. In all cases, the wave reflected by marks, either internal to a matrix or on the surface of a matrix, is used to produce identity patterns such as:

(a) A scan on an oscilloscope screen;
(b) Record of an amplitude as a function of time; and
(c) Digital information resulting from the electric signal in (a) or (b).

These can be a photograph, an analog record or a digital record on most types of recorders existing today.

2.2.2. Method based on external sensor

When a sensor is external to the item to be identified by transmitting ultrasonic waves into the structure, two major possibilities exist:

(a) The item is scanned by an ultrasonic beam which can be focused. This method has the advantages of concentration of energy on the marks to be seen and, in particular, easy interchangeability of transducers.

(b) The transducer is in a fixed position and the beam irradiates the whole zone to be examined. The energy density is often very low (mainly for internal marks...
in steel) and the characteristics of the transducer are very important if reproducible identities are to be obtained. Interchangeability of transducers is possible only under special conditions. The precision in mechanical positioning of the transducer in relation to the structure is important. In this category, several sub-classes can be identified:

(i) Immersion transducers (underwater coupling); and
(ii) Contact transducers.

But the two most important categories are:

(i) Internal marks which can be very sensitive to temperature; and
(ii) External marks. These, as stated above, are more sensitive to possible damage but give better ultrasonic signals.

2.2.3. Method based on integrated sensors

The sensor is included in the seal body or the container structure. This transducer is part of the identity and integrity of the controlled item. A simple electrical connection between the seal and the ultrasonic equipment is sufficient to give the identity/integrity pattern. This method is thus not (or is less) dependent for its reliability in identity reproducibility, on:

- Mechanical positioning;
- Coupling conditions;
- Transducer characteristics; and
- Temperature effect.

2.3. Applications of major ultrasonic techniques

The different techniques are best identified by their applications.

2.3.1. The general-purpose ultrasonic seal (JRC, Ispra) [1, 2] (Fig. 1)

The plexiglass seal comprises a box and a cap, both of which contain bronze inclusions. This seal is rotated in front of an immersion focusing transducer. As the echo due to the inclusions appears and disappears in the ultrasonic beam because of the rotation, it produces a simple record of amplitude as a function of time which can be displayed on all types of chart recorder connected to a commercial ultrasonic apparatus. An example of such a chart is the identity/integrity record.
2.3.2. The LWR cap seal, MTR rivet seal (JRC, Ispra) [3, 4] (Figs 2, 3)

The principle of this technique is as follows — a transducer sonically irradiates a metallic seal through a water column. The echoes, which can be picked up (both for identity and integrity checks) by a commercially available instrument, come from inclusions, discontinuities and wave-propagation mode transformations. A simple Polaroid picture of the cathode tube screen forms the identity record. Digital devices have been built but acceptable repeatability and reproducibility were not obtained. The digital equipment digitized peak values of the "identity" in six gates.

This entire technique has also been used for Candu cap seals [5], but it has been modified by AECL (see Section 2.3.4).

2.3.3. The FAID (Fuel assembly identification device) (Exxon-Sandia Labs., USA) [6] (Fig. 4)

The FAID is a metallic cap seal to be used for LWR fuel assemblies. The ultrasonic transducer transmits waves into the seal through a water column. The signals picked up by the ultrasonic instrument, especially built and enclosed in a portable case, come from the inside of the seal for the integrity check only; and from the surface of the seal (weld) for the unique identity.
FIG. 3. The cap seal for BWR fuel as was proposed by JRC, Ispra in 1970.
The signals are directly treated by the equipment and correlated with a reference signal contained in a bubble memory. The result of the identity verification is a correlation factor; this is therefore an automatic operation.

2.3.4. **The Candu cap seal, redesigned by AECL (AECL, Canada) [7] (Fig. 5)**

The cap seal is used for underwater surveillance of many irradiated Candu fuel bundles. It was proposed and designed by JRC (Ispra) in 1979 using the techniques described in Section 2.3.2. It has been greatly modified by AECL in its marking and data treatment aspects; a surface-type signal (echo of reflection between the marking and the coupling water) and the Sandia Laboratory equipment for on-line correlation (as in Section 2.3.3) are used.

2.3.5. **The raw material can identification at AERE Harwell (UKAEA, UK) [8]**

Using water coupling, precise specific transducer positioning and rotation of the can around its axis, the weld of the can cover can be “seen” through the
cylindrical walls in a particular mode of wave propagation. The ultrasonic signal obtained from the weld is processed by a laboratory computer to produce correlation coefficients. Graphic records (amplitude as a function of angle of rotation of the can) are also given as an identity chart.

2.3.6. The integrated sensor technique (JRC, Ispra) [9] (Figs 6, 7)

As explained in Section 2.2.2, the sensor is embedded in the item, and is a low-cost transducer (approximately US$ 3 — June 1982) but still with good ultrasonic characteristics. It is the result of development work carried out at JRC, Ispra and is today being commercially produced in Italy and France. Major applications investigated or proposed, are:
Uranium oxide enrichment standards (joint IAEA, CEC, NBS, ESARDA programme); General-purpose seals (wire seals or specific mechanical seals) for in-situ verification (CEC collaborations with CEA and IAEA); and Candu cap seal (type 3, with integrated sensor).

An important application of the technique is the integrity check of the containers.

2.3.7. Continuous verification of storage (collaboration between JRC, Ispra and CEA, France) [10]

Integrated seals, as described in Section 2.3.6 and placed on fissile material containers, are continuously connected to selectors which interrogate all seals sequentially. The ultrasonic signals are transmitted to a central computer which compares certain selected peak values with their values obtained during the preceding interrogation (within a few seconds).

3. MAJOR POSSIBILITIES

3.1. Major features of the techniques proposed

From Section 2.2 and from the short descriptions of applications given, it appears that the major features concern the transducer; the scanning procedure of the seal; the echo selection (natural or artificial); the signal processing; the record; and the data treatment.

The transducer always has specific characteristics. In the case of the integrated seals it is part of the identity.

Scanning either takes place (rotation) as for the plexiglass seal and the cans, or is not present.

"Echo selection" is an important feature of the technique — echoes from an inclusion or discontinuity in all different wave propagation modes; and echoes from a surface marking in the longitudinal mode. Signal processing is always used, e.g. smoothing or rectification, or more specific methods such as modulus computation. The record, when it exists, is either a chart (when scanners exist); Polaroid photography; or a record in a bubble memory or on magnetic tape.

Data treatment using correlation techniques is used by Sandia laboratories, AECL, AERE, and also by JRC, Ispra (but starting from Polaroid pictures). Cap seals and rivet seals, as well as integrated sensor seals, are tested with instruments utilizing digital treatment of part of the identity, e.g. six maximum amplitudes at precise time intervals on an A scan.
3.2. Need for signal processing

Even though differences appear in seal scanning and in transducer characteristics, all the techniques are linked. The most important differences concern the "echo selection". This selection determines the need for complex signal processing or the possibility of straight recording. When echoes which come from inclusions or discontinuities in the material are used, the sensitivity is often considerable to external parameters such as mechanical positioning; temperature of the seal; temperature of the coupling; and characteristics of the ultrasonic transducer. In such cases, signal processing is necessary to compensate for, or eliminate such influences — phase shift suppression by computation of the signal modulus as a function of time (mathematical transformation).

3.3. Major techniques

All techniques, once perfected, can give results stable enough to be processed in the same way, either because the ideal case exists or as a result of signal processing (if the ideal case is not feasible). The important difference thus remains at the level of signal processing, which is complicated if internal marks
are to be used because of environmental conditions or other constraints. Figure 8 illustrates the links between techniques at the level of the ultrasonic features themselves, and at the level of data-processing.

4. CONCLUSIONS

4.1. All techniques start, of course, from the same physical principles. Their differentiation lies in the way in which the general features of the “ultrasonic signature” method are applied. The transducer and the marking systems create differences which make the resulting techniques differ greatly from one another.

A large part of these techniques might be considered as common and the discussion has to be pinpointed on the harmonization, leaving the choice of method to experience — results of field tests.
FIG. 8. Three major groups of techniques and their possibilities of automation.
4.2. Harmonization between different ultrasonic techniques should start from the "bottom line" — the first criteria to be defined should concern the way in which automatic verification of the identities can be performed to the satisfaction of the safeguards authorities.

Such guidelines will easily lead to standardization of the read-out experiment; this result could be obtained in the near future for most techniques, as shown in Fig. 8.

4.3. The second part of the read-out equipment — the ultrasonic part — can be standardized only in a few cases as this type of electronic apparatus should be matched with the ultrasonic transducer. In most cases the electronics could derive from a standardized device produced by a commercial firm.

4.4. The record, if present, should be put on one standardized carrier medium (e.g. magnetic tape) which is also the standard proposed for NDA measurement records.

4.5. Harmonization on the markings of seals could take place in cases where external constraints do not impose special features. A clear example can be given — to avoid problems due to deposits on the seals in LWR reactors, cap seals have been designed by JRC, Ispra with internal marks and not with external marks. But in many cases external marks could be the ideal standard, e.g. a small weld thread as proposed by Sandia/Exxon for the FAID.

4.6. In future, after further development of low-cost transducers to be used under all environmental conditions, the integrated system may prove to be the best standard for convenient seal and identification tools for providing effective in-situ verification or even continuous verification. When the integrated system is not appropriate, the specifications of the transducers could be standardized to give the best possible standard in stability and interchangeability.

REFERENCES


DEVELOPMENT AND PERFORMANCE OF THE ADVANCED CONTAINMENT AND SURVEILLANCE SYSTEM AT THE FAST CRITICAL FACILITY FCA

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Abstract

DEVELOPMENT AND PERFORMANCE OF THE ADVANCED CONTAINMENT AND SURVEILLANCE SYSTEM AT THE FAST CRITICAL FACILITY FCA.

A personnel portal monitor was developed for IAEA safeguards at the fast critical facility FCA. The main components of the portal are (1) a walk-through metal detector, (2) a visual surveillance system with closed-circuit television and video-tape recorders, (3) tamper-indication system, and (4) remote monitoring capability through the RECOVER system. The metal detector can detect a single coupon of metallic nuclear fuel plate (2 in. x 2 in. x 1/16 in.) regardless of the orientation of a plate relative to the direction of travel inside the metal detector. The field tests of Phase I and Phase II were conducted jointly by the IAEA and JAERI to evaluate the various features and performance of the portal.

INTRODUCTION

Most of the fast critical facilities are characterized by high inventory of fissile material in relatively pure form from the safeguards point of view. Therefore, international safeguards for these facilities are very severe, and manpower and radiation exposure problems caused by frequent inspections are quite a burden for both the facility operator and the IAEA.

Containment and surveillance (C/S) measures can complement the material accountancy and can be used for timely detection of illicit activities involving nuclear materials. The application of C/S techniques to fast critical facilities is expected to alleviate significantly heavy burdens of frequent inspections aimed at non-destructive assay of nuclear material.
At FCA, the fast critical facility of the Japan Atomic Energy Research Institute at Tokai, a personnel portal monitor has been developed to support international safeguards and to stimulate the reduction of inspection frequency. This portal represents the first unattended equipment of this kind being proposed for use in an international safeguards role. As may be expected, a very comprehensive evaluation of the system is required.

This programme is carried out within the framework of the Japan Support Programme for Agency Safeguards (JASPAS).

1. PORTAL MONITOR DEVELOPMENT

1.1 Containment and Surveillance at FCA

Most of the nuclear material in the critical assembly (a zero power reactor) is located inside the reactor building. The rest of the material is contained in the birdcages which are stored in the fuel storage vaults outside the reactor building. The birdcages are secured by Agency seals and by State seals. It is almost impossible to apply seals to the nuclear material in the reactor because of experimental procedures. Therefore, the main effort of the Agency inspection is directed towards verification of the inventory of non-sealed material in the reactor by NDA techniques.

The reactor building provides almost an ideal containment because of its explosion-tough and air-tight structure. However, there is a limited number of penetrations through the reactor building. These penetrations (except for the personnel doorway) can be secured with the Agency approved seals. The personnel doorway cannot be sealed because of passages for daily maintenance and experimental work. The portal monitor is thought to be one of the most practical measures to detect the diversion through this doorway and to keep the material accountancy in the reactor room simple and easy.

The concept of the C/S system at FCA is shown in Fig. 1.

1.2 Containment and Surveillance in IAEA Safeguards

To become acceptable IAEA safeguards equipment, a portal monitor must demonstrate the capability of meeting a number of basic functional requirements and conditions including the following:
- High confidence in detecting diversion
- Timeliness in detecting and reporting diversion
- Low false alarm rates
- Unattended operation capability for a certain period of time
- Tamper resistance
- High reliability and low maintenance
- Non-availability of information on operating status and event data to facility personnel
- Acceptably low level of interference with facility operations
- Acceptable cost.

A number of these requirements tend to be mutually exclusive. At FCA, development efforts have been expended to meet these requirements as much as possible.

1.3 FCA Portal Monitor [1]

The major characteristics of the FCA portal monitor are the following:
(a) Walk-through metal detector for detecting metallic nuclear material: Nuclear materials of FCA consist of metallic plates of several sizes, nominally 2in.x2in.x1/16in. and 1/8in.\(^1\) Radiation shielding of nuclear material makes radiation detection unattainable as a diversion detection technique in the personnel portal. Metal detection is the only realistic method for detecting nuclear material of metallic form.

The metal detector developed has a special coil arrangement that can detect a single metal coupon of the size of 2in.x2in.x1/16in. regardless of its orientation relative to the direction of passage.

(b) Visual surveillance system to support the unattended operation: The scene inside the portal is recorded on video tapes by means of closed circuit television (CCTV) and video tape recorder (VTR). Recording is triggered by the start of passage of personnel, or by a detector alarm at fixed time intervals. Time, date and, in the event of an alarm activation, a code, are superimposed on the TV scene.

(c) Tamper-indication system and tamper resistance: Intrusion sensors and video sensors detect tamper attempts to break the ceiling and walls of the portal structure or to tamper with the CCTV. Multiplexer transmission of signals maintains signal transmission between the controller and the control/recording console. Micro-switches on the controller box and control/recording box detect attempts to open these boxes.

(d) Activation of an alarm: Altogether six different situations can activate the portal monitor trigger alarms. These are Metal Detection, Tamper, Equipment/Emergency Door Open, Time-Over, Trouble and Multiple Occupancy. For example, without an alarm when the equipment/emergency door is opened, personnel could use the exit from the reactor building before passing through the metal detector. Or, exceeding the limited time period in the portal provides a chance for tampering. Also, more than two persons in the portal may take a chance to throw and catch nuclear coupons through the metal detector without being detected.

(e) Data recording of events using paper print-out and VTR: Direction of passage (from or into the reactor building) and any alarm status (together with time/date of an event) is printed out.

\(^1\) 1 in. = 2.54 cm.
**FIG. 2.** Structure of FCA portal monitor (top view).

**FIG. 3.** Exterior of FCA portal monitor.
(f) **Compatibility with the RECOVER system to timely report anomalies:** The portal is equipped with the interface to the monitoring unit of the RECOVER (Remote Continual Verification) [2] system by means of which the Agency can interrogate the operating status of the system and its data from its headquarters at any time.

An illustration of the FCA portal monitor is shown in Figures 2 and 3. In these figures, MD1 is the IAEA metal detector and MD2 is the operator's metal detector used for preliminary checks of carelessly carried metal objects. The auto-door at the exit of the portal is locked up upon metal detection alarm, in which case personnel has to return to the reactor building and leave the metal there. The second reason for locking this door is not to let a second person enter the portal when the first one is already there. Since this door-lock-up feature provides state-of-health data to facility personnel, it is to be separated from the Agency part of the portal and will be used as a traffic control device for the facility operator.

2. **JOINT FIELD TEST OF FCA PORTAL MONITOR BY IAEA AND JAERI/FCA**

2.1 **Purpose and Procedures of Field Tests**

To evaluate the performance of the portal, IAEA and JAERI/FCA agreed to conduct field tests in three steps. The Phase I field test was conducted in 1980 to evaluate the system features and performance factors. Several system modifications were agreed upon during the Phase I test and later completed in 1981. The Phase II test was the long-term test of the portal when operated under facility conditions as normal as practical. Emphasis was placed on accumulation of data on the metal detection capability, tamper indicating performance, false alarm rates, overall system reliability and on the impact on the facility operation during the routine use of the portal.

2.2 **Phase II Test on Mark I Version of FCA Portal**

The test on the Mark I version of the portal (i.e. the portal with automatic door lock-up upon metal detection) was initiated in April 1982 and was continued for five months until August 1982. It was agreed to separate the automatic door from the IAEA signal and to use the door only as the entry controller for the operator in order to avoid the multiple occupancy alarm. This version of the portal is known as Mark II.
2.2.1 Special Tests to Verify System Performance Characteristics

During the Phase II test on the Mark I, the IAEA inspectors visited the facility regularly, viewed printed data and verified recorded video tapes. There was also a number of special tests performed by inspectors during these visits designed to exercise and verify repeatedly over a long period of time all features of the portal. Special attention was paid to sensitivity and reliability of the metal detector, simulated component failures and tamper events. During the special tests, the correlation of the visual scene and alarm displays recorded on video tapes with printed data was confirmed. The following are the procedures and results of special tests:

(a) **Metal detection** - A person without any metal objects on him was requested to carry a metallic coupon (stainless steel plate of 2in.x2in.x1/16in.) through the metal detector, using the different plate orientation against the direction of passage at different walking speeds and holding the plate in different positions. All these passes through the metal detector resulted in an alarm printed out on the printer and recorded on video tape.

(b) **Tamper indication** - Various attempts were made to simulate possible tamper scenarios. These were designed to activate intrusion sensors or the motion detector, tampering with the visual surveillance system or with the controller box. All these resulted in recorded tamper alarms, as expected.

(c) **Alarm activation other than metal detection and tamper indication** - Time-over (exceeding the time allowed in the portal), Multiple Occupancy (existence of more than two persons in the portal), Equipment/Emergency Door Opening and Trouble (certain equipment component faults) are designated as alarm events. All attempts related to all these alarms resulted in an alarm indication.

These special tests were performed repeatedly through Phase II tests and the results were always successful.

2.2.2 Data Review and Analysis

A team consisting of IAEA development and operation division staff and JAERI/FCA staff endeavoured at the end
of the five-month test period to analyse a large number of data.

(a) Terminology for statistics - Necessary definitions were established, which were then applied to data statistics, in order to obtain results based on safeguards criteria.

i) False Alarm

- False alarm of metal detector - An alarm caused in spite of there being no metal in the metal detector, or the multiple trigger caused by one large metal object.
- False alarm of Trouble, Tamper, Time-Over or Multiple Occupancy - An alarm triggered without evident cause.
- False alarm explained - A false alarm which meets the above definition and the cause of which can be explained later by a recorded video tape or by some other method.
- False alarm unexplained - A false alarm whose cause cannot be explained.
- False alarm unresolved - An unexplained false alarm where there is no chance of confirming by other means that the nuclear material has not been diverted.

ii) False Alarm Rate

- False alarm rate no. 1 - Ratio of total number of false alarms (explained and unexplained) to the total number of normal passages (total number of passages into and out of the reactor room) including number of returns.
- False alarm rate no. 2 - Ratio of total number of unexplained false alarms to that of normal passages including returns.

iii) System Failure

- Failure of the metal detector, which is the most vital part of the portal.

(b) Summary of data analysis - The printed data and the recorded video tapes were reviewed for consistency and summarised in day-by-day event tables. A typical data summary is shown in Table I. These data were accumulated during a two-week period.
Table I. TYPICAL DATA OF DAY-BY-DAY LOG DURING PHASE II TEST ON FCA PORTAL MONITOR, MARK I VERSION 1)

<table>
<thead>
<tr>
<th>Date</th>
<th>Total Events</th>
<th>No. of passages</th>
<th>No. of alarms</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>DIR-A 2)</td>
<td>DIR-B 3) RET 4)</td>
</tr>
<tr>
<td>May 30 Sun</td>
<td>0</td>
<td>3 3 1</td>
<td>1 0 0 0 0 0 0</td>
</tr>
<tr>
<td>31 Mon</td>
<td>8</td>
<td>6 6 1</td>
<td>1 0 0 0 0 0 0</td>
</tr>
<tr>
<td>June 1 Tue</td>
<td>14</td>
<td>8 8 1</td>
<td>1 0 0 0 0 0 0</td>
</tr>
<tr>
<td>2 Wed</td>
<td>18</td>
<td>4 3 0</td>
<td>0 0 1* 1* 0 0</td>
</tr>
<tr>
<td>3 Thu</td>
<td>9</td>
<td>16 15 4</td>
<td>3 0 1* 1* 2 1</td>
</tr>
<tr>
<td>4 Fri 5 Sat</td>
<td>43</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* Equipments transfer
1) The portal with door-lock up feature upon metal detection,
2), 3) Passage into and out of reactor room, relatively,
4) Return to reactor room side during passage from reactor room,
5) Metal detection
6) Trouble
7) Equipment/Emergency door-opened
8) Tamper attempts
9) Time over
10) Multiple occupancy

Altogether, data of 137 days from 4 April to 18 August 1982 were analysed. Out of 137 days' data, 41 were omitted because the equipment/emergency door was opened on these days, which would make data analysis rather complicated. These 41 days include 11 days for IAEA routine inspections and Phase II special tests. On the occasion of visitor tours into the reactor building, this door was opened five times. Only three openings of the equipment door were really necessary during this test period; the rest would not have been needed if previous arrangements had been made.

For the 96 days analysed during this exercise, 1492 passages including returns were recorded.

The summary of analysis is shown in Table II for ten two-week periods. There were a total of 42 false alarms, 36 explained and six unexplained. Out of 36 false alarms explained, five alarms in the third period were tamper alarms caused by flickering of illumination outside the portal. Because of this relatively large number of false alarms and of the exceptionally low frequency of passages in this period, false alarm rate no. 1 was very high.
There were two big earthquakes, one in the 8th and one in the 9th period. These caused four metal detection alarms in the 8th period, and one alarm in the 9th period. The large number of 13 false alarms out of 14 in the 10th period were metal detection alarms caused by heavy thunderbolts occurring within 30 minutes. These causes were very clearly and easily identified but made false alarm rate no. 1 very high. The rest of the 13 false alarms out of 36 alarms were metal detection alarms multiple triggered by one large metal object. Therefore, one half of the explained false alarms were caused by natural phenomena.

Of the six unexplained false alarms, five were metal detection alarms and one was a multiple occupancy alarm. Recorded video tapes were reviewed for these six alarms and no personnel were found in the monitor at the time of their occurrence. Therefore there was no chance of diversion at the time. The team then concluded that these unexplained false alarms were caused, probably, by component malfunction.

During the special tests, the metal detector was repeatedly tested for its sensitivity and reliability using a test piece under various conditions. The metal detector never missed detecting the test piece and over the Phase II test period it did not fail. The team, therefore, concluded that there was no system failure.

Comparing the explained and unexplained false alarms, the former is mostly a false alarm of some system component (its cause can usually be identified by other components of the system) and therefore it can be classified as a component false alarm. The unexplained false alarm is usually a false alarm of some system component, but its cause cannot be explained even with the help of the overlapping function of other individual components and therefore it can be classified as a system false alarm. As a result, false alarm rate (FRA) no. 1 is the sum of the false alarm rates of individual components and FRA no. 2 is the system false alarm rate. In Table II, the average FRA no. 1 over the test period is 2.8%. As was explained, half of this is due to natural phenomena and the rest of it can probably be reduced by modifications in the electric
Table II. SUMMARY OF DATA ANALYSIS FOR PHASE II FIELD TEST ON FCA PORTAL MONITOR MARK I VERSION

<table>
<thead>
<tr>
<th>Period in 1982</th>
<th>No. of passages</th>
<th>No. of False Alarms</th>
<th>False Alarm Rate(%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Explained</td>
<td>Un-explained</td>
</tr>
<tr>
<td>Apr 4 to Apr 17</td>
<td>166</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>Apr 18 to May 1</td>
<td>205</td>
<td>2</td>
<td>0</td>
</tr>
<tr>
<td>May 2 to May 15</td>
<td>38</td>
<td>5</td>
<td>0</td>
</tr>
<tr>
<td>May 16 to May 29</td>
<td>95</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>May 30 to Jun 12</td>
<td>117</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Jun 13 to Jun 26</td>
<td>312</td>
<td>6</td>
<td>0</td>
</tr>
<tr>
<td>Jun 27 to Jul 10</td>
<td>145</td>
<td>1</td>
<td>0</td>
</tr>
<tr>
<td>Jul 11 to Jul 24</td>
<td>158</td>
<td>6</td>
<td>3</td>
</tr>
<tr>
<td>Jul 25 to Aug 7</td>
<td>91</td>
<td>1</td>
<td>0</td>
</tr>
<tr>
<td>Aug 8 to Aug 18</td>
<td>165</td>
<td>14</td>
<td>2</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>1492</strong></td>
<td><strong>36</strong></td>
<td><strong>6</strong></td>
</tr>
</tbody>
</table>

circuitry and in the portal monitor housing structure. The system false alarm rate (FRA no. 2) of a two-week period is between 2% to 0% with the average value of 0.4% over the whole test period. This low rate is not surprising when one takes into account that the system is redundant with overlapping functions of individual components.

It also became clear that video tape review forms an essential part of the data verification procedure, namely in cases of complicated scenarios presented on the print-out. It should be noted that the portal monitor has been designed as a surveillance system with redundant or overlapping individual components and therefore, in most cases, it was relatively easy to find out the cause of anomalies.

2.2.3 Maintenance Data

The FCA portal monitor was constructed in late 1979. Since then the portal has been operated continuously for more than two years with several intermissions for minor modifications and some testing. The portal experienced some troubles of the metal detector during the initial testing period. The troubles were the comparatively frequent false
alarms of metal detection. The noise filter of the electric
circuitry and housing structure of metal detector were
modified in order to reduce the false alarm frequency. After
this modification, the frequency was reduced to about once per
month (average value during the Phase II test period).

The other detected failures were mechanical failures
of video tape recorders at the final stage of the Phase II
test. The video tape recorders were operated for more than
two years and no maintenance was performed except for cleaning
the video tape heads. Failures of the mechanical parts of
tape shuttling occurred but these were not wear-out failures.

Except for the failures of the VTRs, the portal
system has not experienced any other failures and can be
considered as equipment with low frequency of maintenance.

2.3 Future Plan for Development and Evaluation

As mentioned in Section 2.2, the portal monitor Mark I
version was modified as Mark II after the Phase II tests. For
the Mark II, the automatic door lock-up feature and any
indications that will give information on operating status to
facility personnel were excluded from the device. The field
tests on the Mark II version will start in November 1982, for
three to five months, jointly by the Agency and JAERI/FCA.

Data analysis and long-term experience with the use of
the portal monitor brought some further requirements on addi-
tional system modifications and more tests. Consequently, the
portal will be redesigned after tests on the Mark II to incor-
porate several significant improvements, namely microprocessor
control and modular system construction. These refinements
aim to improve the Agency's safeguards performance in the
field in providing high quality man-machine interface with a
computer and presenting the possibility of establishing a good
regular maintenance and repair programme based on exchange of
modules.

Concurrently, a comprehensive system of containment
penetration monitoring will be established and constructed,
thus completing the application of up-to-date C/S methods to
the Fast Critical Facility. This comprehensive system will
consist of a combination of C/S equipment such as CCTV in and
between the double-structured reactor building, remotely veri-
fiable seals, intrusion sensors, gas detectors and portal
monitor. The objective of gas detectors and CCTV in the reac-
tor room is to detect gases which will be present during the
desolution process of metallic uranium and to watch for such illicit activities.

3. CONCLUSIONS

The Phase II field test has lead to a number of conclusions which are summarized below.

(1) The portal monitor performed very well over the period of the Phase II test and demonstrated the capability of meeting most of the basic functional requirements and conditions which are required for acceptable IAEA safeguards equipment.

(2) Among these requirements, those of high confidence in detecting diversion, low false alarm rate, unattended operation capability, tamper resistance, high reliability and low maintenance and timeliness in detecting diversion were met. The requirement for timeliness in reporting diversion was verified during the RECOVER field test in 1980 [3].

(3) Interference with facility operations is a rather severe problem from the facility management point of view. However, facility personnel will prefer this interference to the frequent inspections accompanied by non-destructive assay of nuclear material.

(4) High reliability was verified as a whole, but VTRs were recognised as one of the more unreliable elements of this system because of their complicated mechanical parts for frequent tape shuttling. The maintenance programme of the VTRs should be established before the instrumentation of the portal as IAEA equipment.

Further modification, improvement and evaluation will be performed. The role of containment and surveillance measures becomes increasingly important in the course of pursuing the cost-benefit aspect of international safeguards. The factors which should be taken into account for cost-benefit consideration are (a) operation shut-down cost, (b) manpower costs as direct costs, and (c) radiation exposure problems as indirect costs.

Combining C/S measures and periodic inspection will lower the frequency of NDA verification and will be the practical measure of international safeguards to limit manpower requirements and decrease its intrusiveness.
ACKNOWLEDGEMENT

The authors gratefully acknowledge the help of Mr. H. Ogawa (JAERI) in performing the successful Phase II test and data analysis. Thanks are gratefully extended to Mr. Y. Odama (Shimadzu Corp) for his efforts in constructing the portal and for his valuable discussions during the Phase II test. Appreciation should be expressed to the members of FCA for their helpful co-operation.

REFERENCES


DEVELOPMENT OF AN ELECTRONIC SEAL AND REMOTE MONITORING SYSTEM

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Abstract

DEVELOPMENT OF AN ELECTRONIC SEAL AND REMOTE MONITORING SYSTEM.
A real-time centralized remote monitoring system with electronic seals was developed at the Nuclear Material Control Center within the framework of the JASPAS programme. This system should facilitate safeguards inspections and verifications. The system consists of two major components, the remote monitoring unit (RMU) and the electronic seal unit (ESU). These components are connected by the transmission line (LINE). A fibre optic cable is used as a sealing wire in the ESU, and the function of tamper-indication is applied to the ESUs and LINEs. The information of the ESU in operation is monitored centrally by the RMU, and if an anomaly occurs in the ESU the anomaly information is recorded with the date and time in the RMU. A demonstration-type model was completed at the end of November 1981, and an on-site demonstration was performed.

1. INTRODUCTION

When a large-scale facility with a large annual throughput of nuclear material is considered, it is forecast that the safeguards work in the storage area will be increased because of the increase in the inventory. Generally, the nuclear material stored in the storage area is controlled or managed by the application of seals. To ensure the integrity of the material it is necessary for inspectors who have access to the seals to ensure their identity and integrity. But it is almost impossible to ensure these during one inspection. Therefore, the random sampling method for identifying many applied seals was introduced. The main purpose of the project is to develop a real-time centralized remote monitoring system with small and light-weight electronic seals which can ensure the identity and integrity of all applied seals.

The functions needed to ensure the integrity and to identify the seals are tamper-indication and the detectability of any counterfeit for each of them.
The integrity of a container and a flask, to which the seals are applied, can be assured only after these two functions are satisfied and also after the integrity and identify of the seals are confirmed. The examination of these two functions and the verification of information from the seals have been practised separately by inspectors and by headquarters personnel. The former verify any tamper-indication in each facility, and the latter examine the possibility of seal counterfeiting. In this method, the integrity of the sealed unit (container or flask) cannot be assured until absence of counterfeiting has been confirmed in headquarters. Accordingly, the timely detection of an anomaly cannot be done in these cases. The new sealing system developed is able to verify both tamper-indication and counterfeit at the time of inspection.

Recently, a seal with a built-in microcomputer appeared. It can verify tamper-indication automatically with programmed functions. Furthermore, it is able to record the date and time when an anomaly occurs. Because of this recording function, the safeguards effectiveness is expected to increase from the standpoint of recognizing the time when the opening and tampering of the seal took place, and the period from the anomaly occurrence to the next inspection. But this seal has an inner battery source and reduction of battery power could cause operational troubles during long-term operation. Therefore, the newly developed sealing system has no battery source, but the ability of remote verification and centralized monitoring.

Models of three types were manufactured and tested during the project. Model I was used to establish the electronic circuit and Model II to organize the system components. Model III, which is described in this paper, was demonstrated at the nuclear facility, and the applicability, usability and reliability were evaluated by the NMCC staff and IAEA personnel.

2. CONCEPTS AND DESIGN

The development concepts of the electronic seals and the remote monitoring system are as follows:

(1) The system must be able to monitor many seals by a monitoring device in real-time;
(2) Both sealing wire and seal unit must indicate any tampering;
(3) The system must monitor tampering and other anomalous events for each seal and must indicate the details of such an anomaly;
(4) The system must record the date and time when an anomaly occurs and its details;
(5) The system must detect a seal counterfeit;
(6) The seal must be re-usable; and
(7) The verification activities must be simple.
The specifications we considered during design are as follows:

1. We introduced a centralized monitoring technique with a microcomputer system to verify operational status and anomaly in a seal.
2. We regard the seal as a sensor in the remote monitoring system. It always transmits its own operational or anomaly status to the remote monitoring unit (RMU).
3. Each seal is connected to the RMU by a long-line cable. The sealing status signals are sent back to the RMU through the cable.
4. There must be no battery power source in a seal unit.
5. The sealing status signals must be encrypted.
6. The optical fibre, which is highly tamper-proof, must be used as a sealing wire.
7. The seal unit must be small and light in weight in order to reduce intrusive-ness into the facility where the seals are applied.
8. High reliability of the whole system must be assured.

3. SYSTEM DETAILS

3.1. System

This system consists of the remote monitoring unit (RMU) with a built-in microcomputer system and the electronic seal unit (ESU) connected by transmission lines as shown in Figs 1 and 2. The RMU has four transmission lines (LINE), and 32 ESUs can be connected to each LINE. Therefore, the RMU is able to monitor 128 ESUs. The maximum allowable length of a LINE is 2 km and that of a
FIG. 2. (a) Remote monitoring unit (RMU).
(b) Electronic seal unit (ESU).
sealing wire is 2 m. The size of an ESU is 70 × 80 × 28 mm and it weighs about 250 g.

Environmental requirements are:
(i) Place of use: indoors
(ii) Temperature: 5 to 35°C for RMU
     -10 to 50°C for ESU
(iii) Humidity: 20 to 80% RH for RMU
      10 to 90% RH for ESU

3.2. Function

Each ESU has its own data set number and address number in order to distinguish it from the others, and the RMU interrogates each seal sequentially. Each ESU usually sends a signal which means ‘normal operation’ to the RMU. But once an anomaly occurs at an ESU, the ESU sends a signal which means ‘abnormal event has occurred’ to the RMU. Also, the counterfeit is detected by checking the data set number. The RMU records the details of an anomaly, and the date and time in the event memory when an anomaly occurs. Should the power source fail for either intentional or accidental reasons, the events recorded before the failure happened are maintained by the auxiliary battery source.

Tampering activities with this sealing system were analysed to decide anomalies which must be detected by this system. As a result, the tamper-indication shown in Table I was introduced.

3.3. Hardware

To ensure high reliability of each unit, the United States military specifications, or their equivalent, were adopted. The block diagrams of the two components of this system are shown in Figs 3 and 4.

3.4. Principle

The ESU has no electronic power source in itself. The power is supplied from the RMU in the shape of random-phase signal pulses. Each ESU is interrogated by this random-phase signal pulse sequentially and the signal pulse is sent back to the RMU only when the interrogating signal coincides with the address number of the seal.

The signal pulse sent back to the RMU consists of two kinds of function bit. One is the anomaly detection bit and another is the data set bit. Tampering is detected and indicated by a combination of the anomaly detection bit, and counterfeiting is detected and indicated by the change of the data set bit.
<table>
<thead>
<tr>
<th>Typical tampering action</th>
<th>System trouble</th>
<th>Tampered</th>
<th>Seal opened</th>
<th>Tampered &amp; opened</th>
<th>False seal</th>
<th>Line short</th>
<th>Line opened</th>
<th>Power failure</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Sealing wire was cut</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
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<td></td>
<td></td>
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<tr>
<td>2. Sealing wire was disconnected from the seal box</td>
<td></td>
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<td></td>
</tr>
<tr>
<td>3. Seal box was opened to find out the dataset number and address set</td>
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<td></td>
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<tr>
<td>4. Seal box was opened and the sealing wire cut or removed from the seal box</td>
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<tr>
<td>5. Seal unit was exchanged for another seal unit</td>
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<tr>
<td>6. Seal box was opened and the dataset number and address set where changed</td>
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<tr>
<td>7. The electronic circuit in the seal box was broken by physical means</td>
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<tr>
<td>8. The electronic circuit in the seal box was broken by a high voltage electric source</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>Typical tampering action</td>
<td>System trouble</td>
<td>Tampered</td>
<td>Seal opened</td>
<td>Tampered &amp; opened</td>
<td>False seal</td>
<td>Line short</td>
<td>Line opened</td>
<td>Power failure</td>
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<tr>
<td>-------------------------------------------------------------</td>
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<tr>
<td>9. Seal box was disconnected from the transmission line</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>10. Transmission line was cut</td>
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<td></td>
<td></td>
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<td></td>
<td></td>
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<tr>
<td>11. Transmission line was short-circuited</td>
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<td></td>
<td></td>
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<tr>
<td>12. An electronic device for stealing information was connected to the transmission line</td>
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<td></td>
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<tr>
<td>13. The AC power source was cut out</td>
<td></td>
<td></td>
<td></td>
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</tr>
</tbody>
</table>
The six anomalies (TAMPERED, SEAL OPENED, TAMPERED and OPENED, LINE SHORT, LINE OPENED, SYSTEM TROUBLE) are detected by checking the anomaly detection bit patterns in the pulses sent back. Counterfeiting (FALSE SEAL) is detected by comparison between the data set bit pattern and the registered one in the event memory.

To avoid any false alarm, the returned signal is checked three times in the RMU. Only when the three returned signals are identical is the signal processed as correct data.
4. DEMONSTRATION TEST

A demonstration test of Model III was carried out in the bulk-handling facility (PNC Plutonium Fabrication Facility in Tokai) from the beginning of December 1981 to the end of February 1982. During this period an IAEA specialist was invited to Japan to participate in the test. Model III was evaluated for its applicability, usability and reliability in the demonstration test.

4.1. Applicability

4.1.1. Tamper-indication

This system requires 0.375 seconds to interrogate 128 seals three times. Accordingly, to confirm what can be done in 0.375 seconds, successive opening and shutting of the seal unit cover, and the removal of the sealing wire or LINE, were done. As a result, the following conclusions were reached:

(a) Even if the cover of a seal unit is opened in 0.375 s, it is impossible for a divertor to read the data set number and address number of the seal in the remaining time.
(b) A divertor cannot remove and reset the sealing wire or transmission line in 0.375 s.

4.1.2. Detection of false seals

False seals can occur in two phases — the removal of the ES followed by the installation of a new one and the change of the data set number and address number. It was confirmed that "LINE OPENED" and "FALSE SEAL" were indicated in pairs in the former case, and "TAMPERED" and FALSE SEAL" also in pairs in the latter, both in the RSU.

4.1.3. Vulnerability

Sealing systems which have been developed so far were designed to function with the sealing unit only. But this system is designed to function with the ESUs, RMU and LINEs. Therefore, in this system, tamper-protection must be considered for the three above parts. Vulnerability of this system will only be eliminated when tamper-protection, or the function of tamper-indication for the three above parts, has been solved. In this project, tamper-indication for the ESU (sealing wire & sealing unit) is achieved by electronic circuits, and by the introduction of encryption for tamper-protection of the LINE. However, tamper-protection or tamper-indication function for the RMU, is not enough, even if the electronic seal is applied to the RMU cover.
This is the problem that must be settled for the remote monitoring or verification system which has the sensor and monitoring unit separately.

4.1.4. Application

It is expected that this sealing system will be applied to large-scale storage areas in bulk-handling facilities.

4.2. Usability

4.2.1. Operation

The operation of the system for verification is easy because the ESU verification procedure can be performed automatically by putting certain commands into the RMU keyboard.

4.3. Reliability

The demonstration run of the system was carried out continuously from 7 December 1981 to 25 February 1982. The operation period was about 1920 hours 56 minutes. During this period no trouble with the system was observed.

5. FUTURE PLAN

A field test of this system will be carried out at the PNC facilities by national inspectors of the Science and Technology Agency, Japan.
A REMOTE VERIFICATION SYSTEM FOR INTERNATIONAL SAFEGUARDS: STATUS OF THE RECOVER PROGRAMME IN THE IAEA

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Abstract

A REMOTE VERIFICATION SYSTEM FOR INTERNATIONAL SAFEGUARDS: STATUS OF THE RECOVER PROGRAMME IN THE IAEA.

The paper describes the current status of the Remote Continual Verification (RECOVER) programme being conducted in the Agency. The programme which started in 1979 with support from the U.S. Arms Control and Disarmament Agency has, as participants, several other Member States besides the U.S.A., namely Australia, Bulgaria, Canada, the Federal Republic of Germany, Japan and the United Kingdom, the aim being to demonstrate a new system designed to improve the efficiency and effectiveness of international safeguards. The paper briefly explains the RECOVER concept and the functions of the various components of the system such as the Monitoring Unit (MU), On-Site Multiplexer (OSM) and the Resident Verification Unit (RVU). Results obtained since the installation of the RVU at IAEA Headquarters at the beginning of 1980 are outlined. The feasibility of secure transmission data via public telephone lines was demonstrated. In the initial design these data were supposed to be status-data of containment/surveillance devices such as cameras, seals, CCTV. A later modification permitted the transmission of alpha-numerical texts such as certain inspection forms via RECOVER. With the progress of testing and demonstrating the RECOVER system it was felt necessary to evaluate the cost and benefits of implementing RECOVER for IAEA safeguards. A draft report prepared by the Brookhaven National Laboratory on this subject was the main topic of discussion at the last meeting of RECOVER participants held in June 1982. The paper concludes by identifying certain tasks that need to be successfully completed before RECOVER is ready for routine use by the Agency.

1. Introduction

Containment/surveillance (C/S) instruments such as cameras, closed circuit television (CCTV), seals and monitors play a crucial role in the implementation of safeguards by the IAEA. These devices, installed in nuclear facilities around the world, are meant to work unattended for extended periods of time and provide the necessary assurance of non-diversion of nuclear material from areas "guarded" by them. If a device fails, the fault is detected only at the time of the next...
inspector visit. Failure of C/S devices could in some cases result in considerable effort towards reverification of inventories.

To improve this situation, the U.S. Arms Control and Disarmament Agency (ACDA) proposed the development of a secure system for remote verification of the status of C/S devices from a central station. This station could be the IAEA Headquarters, or one of its regional offices. The system was given the acronym RECOVER (REmote CONTinual VERification) [1,2]. To investigate the feasibility and potential benefits of RECOVER to safeguards, the IAEA concluded in 1979 a research agreement with ACDA under which ACDA provided the necessary equipment and a cost-free expert. This agreement is still in force. Several Member States (Australia, Bulgaria, Canada, Japan, Federal Republic of Germany, United Kingdom, United States of America) joined in the cooperative R & D project and persons nominated by these countries attend periodical meetings to review progress and to plan future action. The last meeting was held in Vienna in June 1982.

2. The RECOVER Concept

In the RECOVER system one or more monitoring units (MUs) are attached to a surveillance device which registers various parameters such as battery voltage, film advance, light level (for electronic seal loop), door security (of camera housing) of the device. Eight bits of information are transmitted by each MU to an on-site multiplexer (OSM). The OSM periodically interrogates all MUs attached to it, stores data on their status and, on demand, encodes and transmits these data to the resident verification unit (RVU) located at IAEA HQ over the international telephone system. The RVU is a microprocessor-based device with a display screen and a keyboard, which is attached to the telephone system via a modem and an automatic dialler. The RVU interrogates the OSMs at predetermined frequencies, receives the coded transmissions, decodes them, stores them in its memory and detects whether any predefined "alert" status exists and then activates an alarm display. At present, the RVU is capable of monitoring up to 40 devices (MUs plus OSMs).

3. Results Obtained So Far

Since the installation of the RVU at IAEA HQ in January 1980, a large amount of practical experience has been gained on the hardware and software of the prototype system supplied to the Agency on loan from the U.S. Experience has also been
gained on telephone communication between Vienna and participating facilities in Member States.

In November 1980 a carefully planned three-week test of the RECOVER system was carried out which involved accelerated rates of calling the participating facilities. About 3000 calls were placed in this period to 10 installations. A 40-50% success rate was observed for the calls; if "busy lines" are excluded, the rate of success would be about 85%. The test demonstrated the feasibility of the RECOVER concept, in particular with respect to the secure and authenticated transmission of information via public telephone lines, but revealed at the same time some technical problems mainly related to the interface to the Minolta camera system. This shortcoming is now being remedied.

From September 1981 to March 1982 a long-term reliability test was carried out which involved calling the participating facilities at a reduced rate - about once every two weeks. This test showed a slightly reduced success rate of calls attempted; however, it also showed that once a "call connection" had been established between the RVU and a given OSM, the success rate was 90% or greater. As a follow-up, experiments on error rates in high speed data communication between Vienna on the one hand and Australia, the Federal Republic of Germany and the United Kingdom on the other, have been carried out (March/April 1982). These tests indicated that data could be transmitted over commercial telephone lines at speeds of up to 1200 baud (compared to the speed of 300 baud currently used in RECOVER) with negligible errors.

The switch simulator box was used to monitor planned and recorded switch closures. The purpose of this test was to check for false alarm generated by site equipment. There were no false alarms detected by the system.

At one site, Tokai-mura, a locally developed sensor equipment, the Portal Monitor, was configured with two MUs. This site has been in operation for more than a year and a half. In the near future, the Portal Monitor equipment will be upgraded to be monitored by a third and then a fourth MU.

4. Transmission of Alpha-Numeric Text by RECOVER

As originally conceived, RECOVER was supposed to transmit only status information of C/S devices. In 1981, a new feature was developed and successfully demonstrated. This related to the transmission of alpha-numeric data such as certain inspection forms or free texts up to a maximum length of 2000 characters.
In the early part of 1982, a successful demonstration of alpha-numeric data transmission was carried out between Tokai-mura, Japan and Vienna.

To transmit alpha-numeric text by RECOVER, one needs a specially modified OSM, a data-entry equipment with keyboard, such as a Heathkit terminal, an activator key and a few diskettes, smaller in size than the floppy disks used for the RVU, to store data. Besides this hardware, certain changes in the RVU software are also required. Currently, form types PO2 and N32 and also free text can be transmitted by RECOVER. As an example, transmission of a PO2 form from Japan to Vienna took about 11 minutes, costing about $50 in telephone charges.

5. Cost-Benefit Evaluation

Although RECOVER was started as a pure R & D project fully supported by the U.S.A., the need for evaluating its potential benefits to Agency safeguards was uppermost in everybody's mind. Internal studies and discussions were held in the Agency during 1980/81. The concept appeared attractive, but there appeared to be too many unknowns with regard to practical implementation of the new technique and it was felt that a further system analysis study of the whole question was necessary.

In 1981, the Brookhaven National Laboratory (BNL) was assigned the task of examining all aspects of cost and benefits of RECOVER for IAEA safeguards and produce a report by the end of the year.

The draft report ("An Evaluation of a Remote Continual Verification System, RECOVER, for International Safeguards" by E.V. Weinstock and J.B. Sanborn) was prepared early in 1982 and it was discussed in depth at the meeting of RECOVER participants held in Vienna in June 1982. The BNL report concludes that for certain types of facility, considerable savings in cost of inspection could be achieved by using RECOVER if certain conditions were fulfilled. The BNL report also states that the use of RECOVER to transmit safeguards information such as inspection summary reports could be economical if combined with other uses of RECOVER at a facility such as remote monitoring of C/S devices. Transmission of inspection reports via telephones might also improve administrative efficiency.

The RECOVER meeting of June 1982, while paying tribute to the authors' valuable work, noted certain shortcomings of the report and recommended a revision of it. For example, it was
pointed out that a re-examination of the case of a light water reactor (LWR) would be desirable. It was proposed that additional meetings with the authors and other experts be held to discuss a revision of the report in the light of comments expressed at the meeting. Arrangements for this action are now under way.

6. Current Status and Future Plans

During the last 2-3 years, the RECOVER system installed in the IAEA has been tested with the co-operation of the participating Member States. The feasibility of the concept, in particular accurate and timely collection of data in a secure way using public telephone lines, has been proven. The development of interfaces between C/S sensors such as cameras and seals on the one hand and the RECOVER network on the other has not been satisfactory yet and further work is required in this direction. Development and demonstration of new sensors, such as intrusion monitors, electronic seals and bundle counters, which can be connected to RECOVER, are also urgently required.

Once the reliable operation of the entire RECOVER system, starting from the C/S sensor up to the RVU has been demonstrated, the next step would be a prolonged field trial under realistic conditions in a nuclear facility where RECOVER holds promise of yielding positive benefits. Facilities such as an on-load refuelling power reactor or a fast critical assembly (FCA) appear to be leading candidates, but possibly reprocessing plants and LWRs could be chosen for a field test.

In parallel to the work outlined above, a system analysis study on the lines of the BNL report needs to be pursued to reach a definitive conclusion on the benefits of RECOVER for all major facility types under Agency safeguards. As has been indicated, much work remains to be done before RECOVER is ready for routine use by the Agency.

7. Conclusions

The present RECOVER system is a demonstration system which has served its purpose well, but its capacity is almost saturated with the connection of 30-odd OSMs and MUs. It has attracted much attention from the media as a powerful new tool for "early warning".
For an eventual operational use of RECOVER for Agency safeguards, many modifications of the hardware and software are required. Some preliminary work has already been done in this direction, but it appears to be the consensus of the group of experts that met in June 1982, that before a major effort towards design of an operational RECOVER system, certain other tasks need to be successfully completed. These are:

- Timely supply of reliable hardware components of RECOVER. Reliability of RECOVER components is of paramount importance. Experience with the current version of the portable verification unit (PVU), an essential component of the present system, has not been satisfactory and this piece of equipment is being redesigned and reduced in size. Also OSMs are being refurbished to function better.

- Construction and testing of interfaces for C/S sensors. While new interfaces for the Minota dual camera have been built, interfaces for electronic seals, bundle counters, intrusion monitors, etc. still remain to be developed, especially as the sensors themselves are not quite ready yet.

- Improvement in communication protocols so that the line utilization is improved from the current low value (10-20% of time).

- Revision of the draft BNL cost-benefit analysis to include actual situations encountered by the Agency in order to assess more realistically the cost and benefits of RECOVER. It might be mentioned here that benefits of RECOVER could be other than economical, such as improving the credibility of safeguards.

In a recent letter to Ambassador Kirk (U.S.A.), the Director General wrote: "We feel that the (RECOVER) technology is not sufficiently developed for this (routine) application. The Agency is certainly interested in the further development of the RECOVER technology and will consider its use in Agency safeguards once the system is sufficiently developed." For a successful development, strong support by Member States for the programme is a necessary, although not a sufficient, precondition.
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Identification of highly radioactive irradiated fuel kept under water is a rather complicated task, and it is even more difficult if the integrity is also considered\(^1\). Identification marks, such as letters and/or numbers are not easily recognized in every case. If it is possible the operator’s underwater CCTV monitor is also used by the inspector, but this is not always satisfactory because of poor image quality, inaccessibility and other technical difficulties, particularly where numbers are small (e.g. 1.6 mm in height).

Furthermore, the real identification of fuel (i.e. detection of dummies) requires sophisticated measurements, e.g. Cherenkov radiation observation, \(\gamma\)-radiation measurement, ultrasonic methods.

The poster presents a method, which enables identification and integrity evaluation by direct and detailed visual observation and successive photography of the total external surface of objects kept under water.

Applicability of the method depends on two conditions — first, the surface must carry such permanent unintentional or intentional topographical marks, by the aid of which identification is possible and, second, remote viewing equipment is necessary to provide high-resolution stabilized image of objects kept under water. The first condition is fulfilled by the existence of welding, milling and grinding tracks on the surfaces, but even the numbers and letters engraved or indented intentionally (i.e. surface disturbances can be considered as real and unique identification marks in the same sense as the random tracks. The handling of items can obliterate original identifiers and wear marks appear, permitting time sequence assessment. The second condition can be fulfilled by any suitable device; here we refer to the underwater telescope developed in our institute, providing a stabilized image with 0.03 mm resolution even from 20 m depth of water.

\(^*\) National Atomic Energy Commission, Budapest, Hungary.

\(^1\) SANATANI, S., IAEA-STR-29 (1973).
The integrity of a fuel element, assembly or a capsule can be checked by visual observation, including photography of weldings that exhibit random surface patterns. By comparing successive photographs, counterfeits can be detected.

The surface topography of a large number of irradiated fuel assemblies has been observed, and subjected to evaluation. Results confirm the feasibility of the method.

AN UNDERWATER TELESCOPE DESIGNED FOR VERIFYING NUCLEAR FUEL IDENTITY

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As a part of safeguards inspection at nuclear reactor facilities spent fuel assemblies or elements are visually identified by the serial numbers engraved either on the top or on the side wall, with the aid of periscopes or by underwater cameras\(^1\). A new optical device has been developed for use in the core and in spent fuel ponds to identify fuel assemblies.

The telescope is a high-resolution device with continuously variable magnification that allows remote underwater viewing of fuel assemblies (elements) vertically and horizontally. The unit penetrates the water level to a depth of 20 m while water fills up the tube, providing complete radiation shielding for the viewer. Since no radiation-sensitive optical element is built in at the lower end of the unit, the diminishing of optical image quality because of browning, reflection losses and distortion is eliminated. To facilitate acquisition of the object and detailed observation, magnification can be continuously controlled.

The telescope is designed for use in a vibrating environment to allow high-resolution photography for safeguards verification. It has been tested and used at three research reactors. Figure 1 shows the serial number on the spent fuel assembly photographed under water from a distance of 6 m.

\(^1\) SANATANI, S., IAEA-STR-29 (1973).
FIG. 1. Identification number on a spent fuel assembly.
Number size: 1.80 × 1.25 mm
Shooting range: 6 m (water depth 3.5 m)
Exposure: 8 s.
Magnification: 172X

IAEA-SM-260/38P

SECTION-IMAGING OF REACTOR FUEL BUNDLES

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The first results of a research project are presented in this poster. The work explored the technical feasibility of examining bundles of nuclear fuel by using a powerful recently developed technique known as computed tomography, which makes it possible to measure the three-dimensional distribution of important parameters (density, activities of selected isotopes etc.) inside the bundle.
Conventional X-rays, or neutron-imaging, process produce a two-dimensional projection of a three-dimensional object, while a section image is displayed as if it had been possible to cut and view the bundle over a selected plane.

The basic principles of transmission computed tomography are outlined and summarized as follows — measuring a set of projections at different angles of view, an image of a selected layer can be reconstructed by mathematical methods. The greater the number of measured projections the more accurate is the reconstruction. The method allows complete exclusion of sections not under study.
An experimental apparatus has been set up which comprises three processes:

(i) Recording a set of projections by detecting the intensity distribution of radiation transmitted through the bundle in different directions (scanning process);
(ii) Image reconstruction from measured projections using back-projection algorithm;
(iii) Filtering the back-projected image (two-dimensional convolution).

X-ray films are used to detect and analog mathematical methods for back-projection and filtering. During scanning the bundles are continuously rotated over an angle $\Pi$. A sketch of the demonstration fuel bundle of the nuclear reactor of the Technical University, Budapest, and its X-ray transmission section image, as reconstructed by the experimental apparatus, can be seen in Fig.1.

The first experimental results are promising and indicate several approaches for which studies can be made. For imaging irradiated fuel background fogging can be reduced by applying the moving-slit techniques, or (in principle) epithermal neutron-imaging can be used. The best solution is to apply the reconstruction principle for emission-imaging by which section-imaging with isotope differentiation can be realized and no external radiation source is necessary.

All these methods can be considered as installed systems for verification and (in the case of emission imaging) burnup distribution determination.

ACKNOWLEDGEMENT

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One of the projects included in the United Kingdom Safeguards Programme has been the development of a method of establishing the identity and integrity of rolled seam cans, i.e. finger-printing. This technique can also be applied to other containers.

The requirement was to mark permanently the surfaces of the containers with a “signature” which was not easily forged, even with sophisticated equipment. It was proposed to position the unique “signature(s)” strategically so that they would be damaged by any attempt to enter, change the contents or to reseal the container. Therefore, on reading the “signature” the damage would be detected and the auditors alerted to the need to examine the container’s contents.

The poster shows how the technique was evolved from the initial investigations, when conventional metrology equipment was used to produce surface finish traces of containers. Although these traces did not contain sufficient distinctive features to provide viable finger-printing, similar traces taken through serial numbers vibro-etched by hand, produced unique surface texture traces, i.e. signatures which can only be reproduced when retracing the same path through the original etched serial number. The system has the advantage that the serial number can be visually read, yet contains a unique non-forgeable signature which is traced on to a paper chart, the original paper trace being filed and used for comparing with subsequent traces, which are made by re-scanning the serial numbers whenever verification is required.

The unique character of the traces was demonstrated by manually analysing many traces taken from the same and different vibro-etched serial numbers. This hand-etching was undertaken by several people deliberately trying to copy the original numbers. Machine-engraved numbers did not produce unique signatures.

Progress on developing the system is reported, including the results of large-scale tests using automatic signature-recognition techniques to confirm statistically the uniqueness of the technique and to lay down the operating parameters for “field equipment”.

IAEA-SM-260/37P

DEVELOPMENT OF A VERIFICATION TECHNIQUE FOR ROLLED SEAMED CAN AND OTHER CONTAINERS

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This includes a surface scanner instrument and signal analyser which will reduce the electrical representation of each trace over the vibro-etched serial number (i.e. the signature) to a simple numerical code. When interfaced to a signature verification unit the computed codes from the initial and re-trace over the same vibro-etched numbers will be compared, thus establishing both the identity and the security of the container's contents. Should the signatures differ the operator is alerted to the possibility to forging or of damage to the container.

While developing the technique it became apparent that, by vibro-etching similar numbers on the surfaces of other nuclear instruments, equipment and cabinets, their identity could also be established even when the conventional identity labels had been removed or deliberately switched. This satisfies a further safeguard requirement.

IAEA-SM-260/81P

A CORE-INPUT MONITOR FOR BRUCE-A CANDU REACTORS

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The safeguards scheme being considered for multi-unit Candu reactor electric-generating stations employs item accountancy methods, complemented by containment and surveillance measures. Unattended counting devices will be used to verify the flow of fuel bundles into and out of the reactor core. This poster describes the development of a core-input monitor designed to be mounted on the fresh-fuel loading mechanism, to count and record unirradiated fuel bundles charged into the fuelling machines. These data will provide a verification of the declared fuel flow to the reactor core materials balance area. The output will be determined by independent irradiated fuel counters.

The monitor is a small, battery-powered unit designed to meet the need for a simple inexpensive device that can easily be retrofitted to the existing reactor-fuel loading system. The monitor design was largely set by the existing fuel loading system characteristics, and is based upon three electro-mechanical switches and associated electronic logic circuits. The switches are actuated by each fuel bundle as it is passed into the fresh-fuel loading system. The number of bundles loaded, and the number withdrawn in the
FIG. 1. Core-input monitor: (a) with cover installed; and (b) with cover removed.
reverse direction, are recorded in separate digital registers. The direction of bundle movement is detected from the sequence in which the switches are closed.

Other registers record the number of “unusual events”. An unusual event is one that does not meet the switch timing sequence for a normal bundle-loading operation conducted at the normal speed. For example, if the switches were closed by hand, an unusual event would be registered, unless the switches were both closed and opened in precisely the correct time sequence for a true bundle loading. The duration of the unusual event is recorded in a digital elapsed-time register.

The data registers are enclosed in a sealed box and are accessible only to an inspector; thus, the results of tampering attempts with the switches are not visible to the operator. Other tamper-resistant and tamper-indicating features are included.

The prototype monitor is shown in Fig.1. It has been tested in the laboratory, and will be demonstrated on a reactor at the Bruce Generating Station in the near future.

IAEA-SM-260/82P

CERENKOV LIGHT MEASUREMENT FOR CANDU IRRADIATED FUEL VERIFICATION

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The observation and measurement of the Cerenkov glow from irradiated fuel stored in water offer a potentially simple means for safeguards attribute verification of irradiated fuel assemblies. Work by Dowdy and Nicholson¹, in which the Cerenkov light was observed by using a night-viewing device, has illustrated the applicability of the technique for examining light-water reactor fuels in fuel storage bays in which the bay lighting was extinguished.

At the Whiteshell Nuclear Research Establishment (WNRE), the Cerenkov method is being developed for attribute verification of Candu fuel bundles in

artificially illuminated storage bays. For routine safeguards inspections it is desirable to make measurements in the presence of the normal fuel-bay lighting so as not to compromise operator safety and routine operations.

Experiments have been carried out with the apparatus shown in Fig.1. An air-filled collimator tube extends below the fuel-bay water surface, and on it is mounted a night-viewing device (NVD). The tube serves to eliminate water surface reflections, and a colour filter is placed over the NVD lens to absorb a large fraction of the artificial light, of wavelengths greater than 390 nm, reflected from the walls and other surfaces in the fuel bay. The Cerenkov spectrum, as measured through 4.5 m of water with a monochromator, reaches a maximum at about 345 nm. The colour filter has a relatively low absorbance in this region, and the majority of the Cerenkov light is therefore selectively transmitted to the NVD.

Tests of this equipment have been made in the fuel bays of the WR-1 reactor at WNRE, and the Bruce A Generating Station. To achieve a high Cerenkov measurement sensitivity, a NVD sensitive to UV light, and fitted with a quartz lens, was used. The quartz lens was selected because of its better transmission efficiency for Cerenkov light compared with a standard glass lens.
higher neutron flux during irradiation than the three-month-cooled specimen.)

(a) Mercury-vapour light; (b) in the dark; (c) filtered mercury-vapour light.

FIG. 2. Candu fuel cooled for 3, 6, 12, 17 and 24 months (left to right), viewed with the Cerenkov measurement apparatus under the lighting conditions stated. (The brighter six-month-cooled bundle is believed to have experienced a higher neutron flux during irradiation than the three-month-cooled specimen.)

(a) Mercury-vapour light; (b) in the dark; (c) filtered mercury-vapour light.
The measurements were done under high artificial light levels of 150 lux, provided by a mercury-vapour lamp placed directly above the water surface. To minimize interference from this illumination a UV light filter covered the lamp. The filter reduced the UV component from the light source by a factor of one thousand, with no noticeable effect on the visible light levels in the bay. Under these conditions the Cerenkov glow from fuel cooled for two years could readily be seen and photographed. Some test photographs are shown in Fig.2. The results obtained in the presence of filtered artificial light (Fig.2(c)) and in total darkness (Fig.2(b)) are essentially identical.

IAEA-SM-260/12P

PRINCIPAL FEATURES OF A DESIGN FOR A PORTAL MONITOR FOR NUCLEAR SAFEGUARDS

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A portal monitor for nuclear safeguards aims to detect the removal by personnel of significant amounts of special nuclear materials from a plant. Unlike similar devices intended for security use, it is not specifically intended to identify the individuals attempting such a removal [1]. Since diversion may be with the connivance of the plant operator, the monitor must be automatic, difficult to evade, and both resist and record any determined attempt at tampering. Its output would normally be a log of events available only to an accredited inspector.

Passive detection methods must be used since irradiation of personnel would be unacceptable. To minimize the risk of evasion, both gamma and neutron emissions are detected. Gamma-detection systems are very sensitive for detection of both $^{239}$Pu and $^{235}$U, but this sensitivity is reduced enormously by shielding which could be non-conducting, portable and compact and hence difficult to detect. In comparison, quite bulky shields are needed to affect neutron sensitivity. Their maximum dimensions can be limited by physical limits on doorway access and IR beams may be used to detect their presence. For plutonium detection, adequate neutron detection sensitivity is not difficult to achieve, but not so for
uranium except in hexafluoride form owing to the $^{19}$F ($\alpha$-n) reaction. The achievement of adequate sensitivity at an acceptable monetary cost is an important consideration for neutron detection.

The mass $W$ of SNM detectable at a single pass through a monitor is typically

$$W = \frac{mB^{1/2}}{NE^{1/2}}$$

where $N$ is the specific neutron emission $n/s\cdot g$.

The detection efficiency $E$ must be made high by good geometry and high intrinsic efficiency. The measurement time $\tau$ defines the peak traffic density and is probably limited to 10 seconds. The background $B$ may be reduced by shielding which also reduces the monitor’s susceptibility to any interfering radiation. Factor $m$ affects the false-alarm probability and may be minimized by appropriate data processing.

The effective sensitivity may be further increased by multiple-pass analysis in which assessment is made over many passes through the monitor. If $n_t$ is the total number of passes considered, and of these $n_s$ involves the removal of SNM, then the detection sensitivity is effectively improved over the single-pass condition by

$$F = \left(\frac{n_s}{n_t}\right)^{1/2}$$

Where such high sensitivities are achieved the monitor may respond to effects not normally noticed such as shielding [2] and scatter of the background radiation by the body of a person entering the monitor. The ultimate sensitivity is determined by the effectiveness of methods of compensating for these effects.

A practical monitor based on these principles is envisaged as employing a novel "swinging gate" system to control access, define radiation geometry and residence time. Detection of neutrons would be by $BF_3$ detectors, and gamma rays by NaI (Tl) scintillators (but this may change). A microcomputer [3] carries out the data processing and fault diagnosis with output to an "inspectors' interface" preferably one of an agreed standard design. IR sensors would detect occupancy and bulky shielding whilst other sensors respond to attempted evasion and tampering by activation of a surveillance camera. Power would be from a float-charged battery giving several hours' operation in the event of mains interruption. It is assumed that the plant operator would provide mains power with a guaranteed maximum period of interruption of say half an hour with the monitor recording all such interrupts.

The monitor enclosure is likely to be of mild steel covered with a tamper-indicating thin stove enamel finish which is also insensitive to damage from normal handling. The layout of components should enable an inspector to detect tampering by a rapid visual examination of all surfaces. Modular
construction is planned to help bring most servicing within the capabilities of
the inspector and access would be by an unpickable combination lock. Reliability
would be as high as costs allow with enough redundancy to give a low probability
of complete failure of the system.

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IAEA-SM-260/31P

EXPERIENCES ACCUMULATED TO DATE
IN THE FUEL-ASSEMBLY SEAL
DEMONSTRATION EXPERIMENT AT
THE KAHL EXPERIMENTAL NUCLEAR
POWER STATION

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Within the framework of the IAEA support programme of the Federal
Republic of Germany, a demonstration programme on fuel assembly sealing
systems is being carried out, which is at present under way at the Kahl Experi­
mental Nuclear Power Station. It is to demonstrate, in the field, the sealing of
LWR fuel assemblies from the aspect of safeguards requirements, handling during
operation, safety, and to plan for the final development of existing sealing systems.
Work began in 1978 with the use of ultrasonic seals developed at the Ispra Joint Research Centre. After the seals were adapted and the licensing procedure completed with the responsible national authorities, three fuel assemblies fabricated by Kraftwerk Union/RBU were sealed and loaded into the Kahl boiling-water reactor in May 1980. Visual inspection in May 1981 revealed no defects. Mechanical inspection and re-verification of the seals were carried out in November 1981. The required strength data were attained. However, one problem requiring further examination occurred in the shift of one line in the ultrasonic identification signal.

Sealing with the Ispra seal of fuel assemblies fabricated by Exxon Nuclear Corp. (ENC) had been prepared for reactor loading in November 1981. The seal was adapted to the special design features of the fuel-rod attachment to the upper tie-plate of the fuel assembly by means of a spacer ring. After successfully passing the licensing procedure, three seals each were attached to eight fuel assemblies, which were loaded into the reactor after primary pattern recording of the identification signal had been made.

While work was proceeding with respect to sealing the ENC fuel assemblies a parallel effort was directed at drafting the criteria to be met by an in-situ re-verification procedure, in which the ultrasonic signal of a seal is compared with the primary pattern by the calculation of a correlation factor. An approach along these lines also incorporates developments by the Sandia National Laboratories, USA, and includes further development of the seal proper so as to improve the certain re-identifiability of a seal and the ability to distinguish it from other seals.

The seals developed at Sandia National Laboratories have been ear-marked for incorporation into the demonstration experiment.

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A new ultrasonically verifiable identity/integrity feature for safeguards seals has been invented. This feature is suitable for use in the existing cap seal designed for underwater applications in Candu irradiated fuel storage pools.

The novel identity element of this seal is a stainless steel wire coil incorporated in a cavity in the top of the seal (see Fig.1). The characteristic signature is produced by primary reflections from the metal coils. This design overcomes a fundamental limitation of the original design which was based on metal inclusions embedded in the body of the seal [1]. These inclusions generate irreproducible mode-converted waves which contribute significantly to the measured signature, making repeatable measurements difficult.

A drawing of a cap seal mounted on a stud with the cap pin secured in the split collet of the stud is shown in Fig.1. The randomly coiled wire is located in a well, drilled in the centre of the cap. One end of the wire is passed through a hole in the shouldered pin and welded to the bottom face. The other end is passed through a radial hole at the top of the cap and is secured by a buried weld. If the cap is removed from the stud, the securing pin breaks at the shoulder and the spring is extended beyond its elastic limit causing the coil to distort in an irreversible fashion. In this way the identity and integrity functions are integrated and can be verified by a single measurement.

The identity of the seal is established and subsequently measured using a reading tool containing an ultrasonic transducer which is placed directly on the cap seal. The signatures measured for different seals are different since it has been impossible in practice to make identically reflecting coils. Several methods of recording and analysing the signatures have been investigated. Visual comparison of photographs taken from oscilloscope traces of individual signatures has proved to be acceptable, if somewhat subjective. A more promising approach has been to use the correlation coefficients obtained by comparing digitized signatures in a desk top computer [2].

A histogram of correlation coefficients obtained from pairs of signatures taken from five sets of measurements of fourteen seals is shown in Fig.2.

* AECL, Sheridan Park, Ontario, Canada.
FIG. 1. Random coil seal with its reading tool.
Comparisons of repeated measurements of the same seal produced consistently high coefficients (0.8) as shown on the right hand side of the histogram. Such data can be used to specify a threshold level of acceptance for inspection of seals using a correlation method of analysis. The balance of the histogram was generated from comparisons of pairs of signatures from different seals. These data are used to estimate the statistical number of different seals that could be manufactured and also the probability that two different signatures could be mistaken as similar using the correlation coefficient criterion.

All these analytical techniques have confirmed that the new seal meets the requirements for use in Candu spent fuel storage pools. Consequently, a fully engineered sealing system based on this concept is now being developed.

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ISOTOPE CORRELATION TECHNIQUES

(Session 10)
Chairman

D. RUNDQUIST
CORRELATIONS OF BURNUP WITH THE $^{134}\text{Cs}/^{137}\text{Cs}$ AND $^{154}\text{Eu}/^{137}\text{Cs}$ CONCENTRATION RATIOS FOR IRRADIATED LWR FUEL*

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Abstract

CORRELATIONS OF BURNUP WITH THE $^{134}\text{Cs}/^{137}\text{Cs}$ AND $^{154}\text{Eu}/^{137}\text{Cs}$ CONCENTRATION RATIOS FOR IRRADIATED LWR FUEL.

The paper presents results of experimental and theoretical investigations on correlations between burnup and the concentration ratios $^{134}\text{Cs}/^{137}\text{Cs}$ and $^{154}\text{Eu}/^{137}\text{Cs}$. Gamma-spectrometric measurements have been performed on assemblies of the Rheinsberg Nuclear Power Station (WWER-type reactor) with different irradiation conditions and cooling times. The results of these measurements show that quite linear and biunique correlations between burnup and the $^{134}\text{Cs}/^{137}\text{Cs}$ concentration ratio (corrected for losses during irradiation) as well as burnup and the uncorrected $^{154}\text{Eu}/^{137}\text{Cs}$ ratio exist for this assembly type up to a burnup of 2.6% fima. These statements are confirmed by calculations performed for different WWER assemblies. In detail the dependences of the correlations on neutron flux, the $^{235}\text{U}$ initial enrichment, and the neutron spectrum hardness were investigated. A comparison of the calculated dependences with experimental data from the authors' own measurements or other publications shows good agreement within the experimental error limits for WWER assemblies with initial enrichments in the range between 1.5 and 3.6%. A valuation of the present status in the field of correlations with fission product concentration ratios results in the demand for a uniform evaluation and presentation of the experimental data in order to reach a systematization of the correlations.

1. INTRODUCTION

The correlation technique is concerned with the application of relations between several parameters of spent nuclear fuel, e.g. burnup, concentrations of fissionable nuclides and fission products. Correlations with fission product concentration ratios are of special interest in the field of safeguards, since their experimental determination may be performed without calibrating the equipment by standards (self-calibration method).

Application of the correlation technique in nuclear safeguards requires a precise knowledge of the correlations for different types of reactor and fuel assembly. The present paper

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describes results of measurements on burnt fuel assemblies of
the Rheinsberg Nuclear Power Station (NPS) performed within the
framework of an IAEA Research Contract [1]. A special aim of
these experiments was to investigate the influence of different
irradiation parameters on the correlation between burnup and
$^{134}$Cs/$^{137}$Cs or $^{154}$Eu/$^{137}$Cs concentration ratios.
Calculations have been performed for different assemblies
of the pressurized water reactor type WWER to investigate
theoretically characteristic dependences of the correlations
between burnup or the Pu/U atomic ratio and the concentration
ratios mentioned on the irradiation history and the $^{235}$U
initial enrichment.
The report is completed by a survey and a comparison of
published experimental data concerning the correlations for
different WWER-type assemblies.

2. EXPERIMENTAL

Measurements on spent fuel assemblies were performed at the
Rheinsberg NPS, which operates a 70 MW(e) power reactor of the
Soviet WWER series. The assemblies are hexagonal ones and
consist of 90 cylindrical fuel rods. Some figures on the
assemblies are listed in Table I together with WWER-440 assembly
data.

Five assemblies with 2.0% initial enrichment but different
irradiation histories and cooling times were selected for the
measurements. The choice was especially determined by the
following criteria:
- Number of operating cycles
- Burnup range
- Power density during the individual cycles
- Different core positions.

Some operational data of the assemblies investigated are given in
Table II.

The experimental equipment [2] consists of a measurement
vessel, the refuelling machine, a collimator system, a Ge(Li)
detector and a Silena multichannel analyser.

Gamma-ray spectra have been recorded at 10 axial positions of
the assemblies at distances of 250 mm just midway between the
spacers. To obtain representative mean values for the assembly
cross-sections, gamma spectra have been recorded cumulatively by
measuring at the flat sides of the assemblies.¹ Net peak

¹ Because of the different positions of the assemblies in the core and of the influence of
neighbouring control assemblies, diverse distributions of fission product concentrations across
the assembly sections had to be expected. To obtain information on these distributions, the
most intensive peaks of the fission products have been analysed from the assembly sides
separately. As a result of these measurements, differences at two opposite sides were found.
They go up to ± 20% for $^{134}$Cs. The smallest quantities relate to assembly No. 1.
### TABLE I. DATA ON WWER-TYPE ASSEMBLIES

<table>
<thead>
<tr>
<th>Assemblies</th>
<th>WWER-70</th>
<th>WWER-440</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel length (mm)</td>
<td>2500</td>
<td>2420</td>
</tr>
<tr>
<td>Width across the flats (mm)</td>
<td>144</td>
<td>144</td>
</tr>
<tr>
<td>Number of fuel rods</td>
<td>90</td>
<td>126</td>
</tr>
<tr>
<td>Water/fuel volume ratio</td>
<td>1.8</td>
<td>1.7</td>
</tr>
</tbody>
</table>

### Fuel rods

<table>
<thead>
<tr>
<th></th>
<th>WWER-70</th>
<th>WWER-440</th>
</tr>
</thead>
<tbody>
<tr>
<td>Lattice pitch (mm)</td>
<td>14.3</td>
<td>12.2</td>
</tr>
<tr>
<td>Diameter of rods (mm)</td>
<td>10.2</td>
<td>9.1</td>
</tr>
<tr>
<td>Diameter of pellets (mm)</td>
<td>8.6</td>
<td>7.55</td>
</tr>
<tr>
<td>Central hole diameter of pellets (mm)</td>
<td>-</td>
<td>1.4</td>
</tr>
<tr>
<td>235U initial enrichment(%)</td>
<td>2.0 (0.72, 1.5)</td>
<td>3.6 (1.6, 2.4)</td>
</tr>
</tbody>
</table>

### TABLE II. OPERATIONAL DATA OF THE MEASURED ASSEMBLIES

<table>
<thead>
<tr>
<th>Assembly number</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of cycles</td>
<td>3</td>
<td>4</td>
<td>3</td>
<td>5</td>
<td>7</td>
</tr>
<tr>
<td>Theoretical burnup (% fima)</td>
<td>1.56</td>
<td>1.68</td>
<td>1.38</td>
<td>1.87</td>
<td>1.87</td>
</tr>
<tr>
<td>Irradiation time (d)</td>
<td>701</td>
<td>938</td>
<td>701</td>
<td>1181</td>
<td>1738</td>
</tr>
<tr>
<td>Cooling time (d)</td>
<td>808</td>
<td>809</td>
<td>1104</td>
<td>805</td>
<td>810</td>
</tr>
</tbody>
</table>

Areas of the gamma spectra were calculated using the computer unit for spectra integration of the Silena multichannel analyser.

With the 134Cs/137Cs ratio the concentrations measured were corrected for losses by radioactive decay and neutron capture during irradiation. To calculate the correction factors the irradiation history was divided into time intervals (normally one operational cycle) with constant neutron fluxes.

The overall relative efficiency of the equipment was determined by the self-calibration method [3] using gamma transitions of 134Cs with well-known gamma quantum yields.
FIG. 1. Experimental correlations between burnup and the $^{134}$Cs/$^{137}$Cs concentration ratio (with and without correction for irradiation history).

FIG. 2. Experimental correlation between burnup and the $^{154}$Eu/$^{137}$Cs concentration ratio (without correction for irradiation history).
The concentration of $^{137}\text{Cs}$ was used as a monitor for the total burnup. The experimental mean burnup of assembly No. 1 was normalized to the corresponding theoretical value. The choice of this assembly as a standard is based on the fact that this assembly had a short and simple irradiation history (three cycles only, undisturbed core positions). For this case the uncertainty of the theoretical value is less than 5%.

3. RESULTS OF MEASUREMENTS ON WWER-70 ASSEMBLIES

The dependence of burnup on the $^{134}\text{Cs}/^{137}\text{Cs}$ concentration ratio at reactor shut-down (without/with correction for irradiation history) is shown in Fig. 1 for all assemblies investigated. Obviously, the correlation with uncorrected ratios depends on the irradiation history. A further splitting-up of the correlation between burnup and the uncorrected $^{134}\text{Cs}/^{137}\text{Cs}$ ratio relating to upper and lower measurement positions of one and the same assembly had been found for assemblies of the Rheinsberg NPS [4]. This fact is mainly caused by changes in the axial neutron flux profile.

The relationship of burnup $Y$ with the corrected $^{134}\text{Cs}/^{137}\text{Cs}$ ratio $X$ is also shown in Fig. 1 for all measured data. From an analysis of regression with a linear statement, as shown in the figure, the following conclusions may be drawn:

1. The correction for losses during the irradiation yields a unique correlation between burnup and the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio.
2. There exists an excellent linearity of the correlation up to a burnup of 2.6% fima for assemblies with an initial enrichment of 2.0%.
3. The residual variance corresponds to the statistic error of the concentration ratio determination. There are no significant discrepancies between measurement data belonging to different assemblies.
4. The total error of burnup determination is composed of the residual variance plus the assumed burnup error of the assembly used as a standard.

The relationship of burnup with the uncorrected $^{154}\text{Eu}/^{137}\text{Cs}$ ratio is illustrated in Fig. 2. Clearly, the significant difference in comparison with the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio consists in the fact that the burnup depends linearly on the $^{154}\text{Eu}/^{137}\text{Cs}$ ratio without any correction for irradiation history - this is a great advantage. However, the gamma-spectrometric determination is difficult at cooling times shorter than one year because of the low intensity.

From an analysis of regression, as shown in Fig. 2, virtually the same statements hold true as were made in items
(2), (3) and (4) above with regard to, respectively, linearity, range of validity, and residual variance for the $^{134}\text{Cs}/^{137}\text{Cs}$ case.

4. CALCULATION OF CORRELATIONS

The theoretical consideration of correlations for WWER fuel was performed for the following reasons:
- to confirm quantitatively the results of measurements
- to confirm the statements on the correction of losses during irradiation
- to analyse the influence of the $^{235}\text{U}$ enrichment
- to estimate the dependence of correlations on the neutron spectrum
- to obtain information on the relations between Pu/U atomic ratio and fission product concentration ratios.

The calculations were carried out with constant neutron flux and effective neutron cross-sections.

Figure 3 shows the influence of neutron flux on the relationship between burnup and $^{134}\text{Cs}/^{137}\text{Cs}$ ratio in the case of a 2.0% enriched WWER-70 assembly. The dependence is neither a unique nor a linear one if only concentrations existing at reactor shut-down are used. With the concentrations produced during irradiation the correlations are identical up to a neutron flux of about $5 \times 10^{13} \text{ cm}^{-2} \cdot \text{s}^{-1}$. This is also valid in the case of a time-dependent neutron flux.

Figure 4 shows the correlation between burnup and the uncorrected $^{154}\text{Eu}/^{137}\text{Cs}$ ratio. Contrary to the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio, the influence of the neutron flux level on the concentration ratio existing at reactor shut-down is very small. Differences in the correlations are lower than 0.05% fima for neutron fluxes between $10^{13}$ and $10^{14} \text{ cm}^{-2} \cdot \text{s}^{-1}$. The reason for the quite linear dependence of burnup on the uncorrected $^{154}\text{Eu}/^{137}\text{Cs}$ ratio is the formation of $^{154}\text{Eu}$ by neutron capture and radioactive decay of fission products with mass numbers between 147 and 153.

Correlations obtained with correction for irradiation history in the neutron flux range between $10^{13}$ and $10^{14} \text{ cm}^{-2} \cdot \text{s}^{-1}$ are very close to each other but non-linear. Therefore, the uncorrected $^{154}\text{Eu}/^{137}\text{Cs}$ ratio existing at reactor shut-down is in favour for the linearity and, moreover, for the fact that a correction for irradiation history is not necessary.

Figure 5 shows the relationship between burnup and the corrected concentration ratio $^{134}\text{Cs}/^{137}\text{Cs}$ for WWER fuel assemblies with typical values for the $^{235}\text{U}$ initial enrichments. The curves clearly indicate the strong influence of the enrichment on the correlation. Rather linear dependences
FIG. 3. Calculated influence of neutron flux density on the correlation between burnup and the $^{134}\text{Cs}/^{137}\text{Cs}$ concentration ratio.

FIG. 4. Calculated influence of neutron flux density on the correlation between burnup and the $^{154}\text{Eu}/^{137}\text{Cs}$ concentration ratio.
TABLE III. CORRELATIONS FOR DIFFERENT WWER-TYPE ASSEMBLIES

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Object of measurements</th>
<th>Enrichment (%)</th>
<th>Burnup range (% fima)</th>
<th>Regression equation</th>
<th>$s/\sigma$ (%) (b)</th>
<th>Ref.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Y is burnup (% fima); X is $^{134}$Cs/$^{137}$Cs concentration ratio (corrected for irradiation history)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>WWER-70</td>
<td>1 assembly</td>
<td>0.72</td>
<td>0.1 - 0.6</td>
<td>$Y = 12.1 X + 0.01$</td>
<td>2.0</td>
<td>[4]</td>
</tr>
<tr>
<td>3 assemblies</td>
<td>1.5</td>
<td>0.1 - 1.0</td>
<td>$Y = 17.6 X + 0.05$</td>
<td>1.3</td>
<td>[4]</td>
<td></td>
</tr>
<tr>
<td>5 assemblies</td>
<td>2.0</td>
<td>0.7 - 2.6</td>
<td>$Y = 19.8 X + 0.15$</td>
<td>2.8</td>
<td>This work</td>
<td></td>
</tr>
<tr>
<td>WWER-365</td>
<td>1 assembly</td>
<td>2.4</td>
<td>2.5 - 4.5</td>
<td>$Y = 22.0 X - 0.21(a)$</td>
<td>2.0</td>
<td>[8]</td>
</tr>
<tr>
<td>3 assemblies</td>
<td>3.0</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>10 samples</td>
<td>3.0</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>WWER-440</td>
<td>3 assemblies</td>
<td>3.3, 3.6</td>
<td>0.7 - 4.2</td>
<td>$Y = 22.9 X + 0.05(a)$</td>
<td>4.0</td>
<td>[8]</td>
</tr>
<tr>
<td>12 samples</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Y is burnup (% fima); X is $^{154}$Eu/$^{137}$Cs concentration ratio (uncorrected)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>WWER-70</td>
<td>5 assemblies</td>
<td>2.0</td>
<td>0.7 - 2.6</td>
<td>$Y = 190 X$</td>
<td>4.4</td>
<td>This work</td>
</tr>
<tr>
<td>Y is Pu/U atomic ratio; X is $^{134}$Cs/$^{137}$Cs concentration ratio (corrected for irradiation history)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>WWER-70</td>
<td>8 samples</td>
<td>2.0</td>
<td>0.3 - 1.6</td>
<td>$Y = 72.9 X + 1.03$</td>
<td>3.1</td>
<td>[9]</td>
</tr>
<tr>
<td>Y is Pu/U atomic ratio, X is $^{154}$Eu/$^{137}$Cs concentration ratio (uncorrected)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>WWER-70</td>
<td>8 samples</td>
<td>2.0</td>
<td>0.3 - 1.6</td>
<td>$Y = 574 X + 0.981(c)$</td>
<td>2.3</td>
<td>[9]</td>
</tr>
</tbody>
</table>

(a) Original regression equation converted by authors.
(b) $s$ is the residual variance.
(c) Converted by means of a half-life of $^{154}$Eu: $T = (8.57 + 0.07) a.$
6. CONCLUSIONS

The calculations have confirmed the statements gained from experimental investigations, that for a unique presentation of correlations a correction for losses during irradiation is necessary in the case of the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio but redundant for the $^{154}\text{Eu}/^{137}\text{Cs}$ ratio.

Taking into account the experimental data available from the authors' own measurements and literature, and regarding the comparison of experimental and theoretical results, it may be stated that reliable correlations between burnup and the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio exist for WWER assemblies of different enrichments.

In view of the total amount of experimental data at present available, and with regard to their application for safeguards purposes, an essential progress point might be reached in this field if a uniform evaluation and presentation of experimental data could be achieved in future. To meet this aim the following measures are recommended:

- Instructions for measurement procedures with fuel elements in order to get representative experimental data.
- Compilation of uniform nuclear data for the evaluation of measurements.
- Presentation of measured fission product concentrations corrected for the real irradiation history in the case of $^{134}\text{Cs}/^{137}\text{Cs}$ and for spectrum hardness variations in general.

This way, the gamma-spectrometric correlation technique could become an effective method of general use for checking declared data on irradiated fuel elements within the framework of a system of safeguards methods.

ACKNOWLEDGEMENTS

The authors are greatly obliged to the International Atomic Energy Agency, Vienna, and the Academy of Sciences of the GDR for sponsoring this work. They express thanks to the VEB Kombinat Kernkraftwerke "Bruno Leuschner" Greifswald for good cooperation in preparing and performing the measurements at the Rheinsberg NPS. They thank especially A. Keddar, Project Officer, of the IAEA, Vienna, as well as H.-C. Mehner and S. Nagel of the CINR Rossendorf for valuable collaboration.
REFERENCES


FOUR YEARS OF EXPERIENCE WITH THE USE OF CALCULATED ISOTOPIC CORRELATIONS IN ESTABLISHING INPUT BALANCES AT THE LA HAGUE PLANT.

For more than four years the La Hague reprocessing plant has been using calculated isotopic correlations to establish and check its input balances. The masses of uranium and plutonium entering the plant are determined by the gravimetric balance method, which utilizes the burnup obtained by calculated isotopic correlation as well as the Pu/U ratio measured at the dissolver after cross-checking with the values obtained by correlation. Further, a verification of all the parameters needed to establish these balances — whether physical or chemical in origin — is carried out systematically by means of internal coherence constants which make it possible to detect any anomalies in the dissolution data. The calculated isotopic correlations were evaluated when the analyses of numerous representative samples of irradiated fuel and experimental results of separated isotopic irradiation in water reactor spectra had been interpreted. The accuracy achieved was improved by allowing in the neutron calculations for effects inherent in the first reactor core and by selecting a set of calculation functions which attenuates (by compensation effects) the various perturbations in the irradiation history. The results obtained at La Hague with calculated isotopic correlations on nearly 600 t of reprocessed UO₂, because of their large number and above all their high quality, suggest that it be proposed extending the method to other reprocessing plants. This could be done by the operator himself or by national or international control bodies within the framework of a safeguards arrangement.

QUATRE ANS D'EXPERIENCE D'UTILISATION DES CORRELATIONS ISOTOPIQUES CALCULEES DANS L'ETABLISSEMENT DU BILAN D'ENTREE A L'USINE DE LA HAGUE.

Les corrélations isotopiques calculées sont utilisées depuis plus de quatre ans par l'usine de retraitement de La Hague pour l'établissement et le contrôle de ses bilans d'entrée. Les masses d'uranium et de plutonium entrant dans l'usine sont déterminées par la méthode du bilan gravimétrique. Celui-ci utilise le taux de combustion obtenu par corrélation.
isotopique calculée ainsi que le rapport Pu/U mesuré au dissolveur après une vérification par recoupement avec les valeurs obtenues par corrélation. De plus, une vérification de tous les paramètres nécessaires à l’établissement de ces bilans, qu’ils soient d’origine physique ou chimique, est systématiquement effectuée à l’aide d’un jeu de constantes de cohérence interne permettant la détection d’éventuelles anomalies dans les données de dissolution. La qualification des corrélations isotopiques calculées a été effectuée lors de l’interprétation des analyses de nombreux échantillons représentatifs de combustibles irradiés ainsi que des résultats d’expérience d’irradiation d’isotopes séparés dans des spectres de réacteurs à eau. La précision obtenue a été améliorée en prenant en compte dans les calculs neutroniques les effets inhérents au premier cœur du réacteur et en choisissant un jeu de fonction de corrélation qui atténue par effets de compensation les diverses perturbations de l’historique d’irradiation. Les résultats obtenus à l’usine de La Hague sur près de 600 tonnes d’UO₂ traitées lors de l’utilisation des corrélations isotopiques calculées incitent, par leur grand nombre et surtout par leur qualité élevée, à proposer l’extension de cette méthode à d’autres usines de traitement. Celle-ci peut être mise en œuvre par l’exploitant lui-même ou par des organismes de contrôle nationaux ou internationaux dans le cadre d’un contrôle de garanties.

INTRODUCTION

L’utilisation dans les usines de retraitement de corrélations isotopiques calculées est un moyen efficace pour l’établissement ou le contrôle des bilans d’entrée.

L’usine de retraitement de La Hague effectuant un bilan d’entrée contractuel de type gravimétrique, les corrélations y ont été développées dans l’esprit d’une aide à son exploitant.

Les corrélations isotopiques calculées ont été mises en œuvre à l’usine de La Hague dès les premières campagnes de retraitement de combustible à eau légère. Les premiers résultats obtenus ont fait l’objet d’une précédente communication [1].

Le but principal alors visé résidait dans la détermination du taux de combustion par corrélation isotopique en remplacement de la méthode du néodyme qui s’avérait longue, coûteuse et non adaptée à une exploitation industrielle.

On proposait également une détermination du rapport Pu/U par corrélation isotopique en tant que valeur de recoupement des résultats analytiques.

Les excellents résultats obtenus alors ont rendu systématique l’usage des corrélations isotopiques calculées. De plus, leurs utilisations ont été étendues au contrôle des principales données permettant l’établissement des bilans d’entrée, ceci par l’utilisation d’un jeu de constantes de cohérence interne également calculées.

On présente dans cette communication le bilan des quatre années d’expérience acquise lors de l’utilisation des corrélations isotopiques calculées; les résultats exposés concernent près de 600 tonnes d’UO₂ traitées.
1. OBJECTIF

L’objectif visé est différent selon que la procédure de contrôle basée sur les corrélations isotopiques calculées est utilisée par l’exploitant de l’usine de retraitement, en tant que vérification interne du procédé, ou par un organisme de contrôle extérieur à celui-ci, en tant que contrôle des garanties.

Le choix est en réalité imposé par le type de bilan d’entrée contractuel de l’usine de retraitement.

Dans le cas d’un bilan d’entrée de type gravimétrique, l’objectif consiste à effectuer des bilans de référence en les faisant précéder d’une vérification des paramètres nécessaires à leur établissement. Cette procédure est mise en œuvre par les laboratoires d’analyse de l’usine. En cas d’anomalies lors des tests de cohérence, l’accessibilité aux informations de base et la possibilité de refaire immédiatement des analyses permettront dans la plupart des cas d’en localiser à peu près l’origine et d’y remédier le cas échéant.

Dans le cas d’un bilan d’entrée de type volumétrique, l’objectif consiste à effectuer un bilan d’entrée de contrôle, qu’il soit de référence ou de recoupement, indépendamment de ceux utilisés contractuellement par l’exploitant.

Dans cette optique, un organisme de contrôle national ou international peut procéder à une vérification efficace des valeurs fournies par l’exploitant en effectuant un bilan d’entrée gravimétrique utilisant les corrélations isotopiques calculées. Si celui-ci permet de détecter la présence d’anomalies sans difficulté, les conditions dans lesquelles s’effectuent ces opérations de contrôle par un organisme extérieur à l’usine ne permettent généralement pas d’en remonter à la source.

Dans les deux cas de figure, la procédure utilisée est dans ses grandes lignes la même :

— une vérification de la bonne cohérence des différentes données de dissolution à l’aide d’un jeu de constantes de cohérence interne;
— l’obtention des taux de combustion et du rapport Pu/U à l’aide des fonctions de corrélation isotopique en vue de l’établissement des bilans d’usine ou des bilans de contrôle.

2. PRINCIPES DES CORRÉLATIONS CALCULEES

Le nombre restreint de grandeurs nécessaires à l’établissement des bilans d’entrée et la connaissance acquise en physique des réacteurs sur les lois de formation et de disparition des différents isotopes ont permis la mise en évidence de relations présentant des caractères d’invariance aux fluctuations des conditions d’irradiation du combustible.
Ces relations se divisent en deux catégories:

1) **Les constantes de cohérence interne**

Elles prennent en compte des grandeurs décrivant les caractéristiques physiques des assemblages (comme le type du combustible, la teneur initiale en $^{235}$U, le temps de refroidissement, etc.) ainsi que les résultats des analyses chimiques effectuées après leur dissolution (composition isotopique, rapport Pu/U).

Elles conservent une valeur quasi constante dans les gammes usuelles de fonctionnement. Ainsi, tout écart important existant entre la valeur fixée par le calcul et celle obtenue à l'aide des données de dissolution implique la présence d'une anomalie. Celle-ci peut provenir des caractéristiques physiques ou chimiques de la dissolution examinée. Son origine peut être cernée si l'on dispose, d'une part, d'un jeu étendu de constantes de cohérence interne couvrant la totalité des données de base d'une dissolution et, d'autre part, de la possibilité de refaire certaines analyses.

Notons enfin que les valeurs prises par ces constantes peuvent changer selon le type de combustible considéré.

2) **Les fonctions de corrélation**

Elles consistent en des relations liant les grandeurs que l'on désire mesurer, comme le taux de combustion ou le rapport Pu/U, aux compositions isotopiques du combustible retraité.

Si la forme de ces relations est la même pour tous les combustibles à eau légère, les valeurs prises par ses coefficients sont nettement différentes pour les divers modèles de combustibles ou de réacteurs considérés.

Ils doivent donc être calculés, ainsi que les valeurs des constantes de cohérence, pour chaque type de combustible.

3. **QUALIFICATION ET PRECISION**

La qualification des calculs neutroniques d'assemblages est un des aspects essentiels de la validation d'ensemble, des calculs de corrélation. Elle doit être effectuée, d'une part, pour le modèle simplifié de l'assemblage moyen et, d'autre part, pour les banques de données neutroniques associées.

Cette qualification a été effectuée lors de l'interprétation des analyses de nombreux échantillons de combustibles irradiés situés dans des conditions d'irradiation très différentes et provenant d'une large gamme de types de combustible, de teneurs initiales en $^{235}$U et de taux de combustion [2].
TABLEAU I. ÉCARTS ENTRE LES TENDANCES OBTENUES

<table>
<thead>
<tr>
<th>Rapport isotopique</th>
<th>Ponctuel — Global</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{239}\text{Pu}/^{238}\text{U}$</td>
<td>+ 1,3</td>
</tr>
<tr>
<td>$^{240}\text{Pu}/^{238}\text{U}$</td>
<td>+ 0,3</td>
</tr>
<tr>
<td>$^{241}\text{Pu}/^{238}\text{U}$</td>
<td>+ 2,7</td>
</tr>
</tbody>
</table>

a Ce tableau présente les écarts en % entre d’une part, les écarts moyens calcul-experience mis en évidence sur les principaux rapports isotopiques, par l’interprétation des analyses d’échantillons ponctuels de combustibles irradiés et, d’autre part, les écarts moyens calcul-experience obtenus par l’interprétation des analyses d’assemblages combustibles dissous globalement.

Note:
Le taux de combustion considéré est celui obtenu par l’appauvrissement en $^{235}\text{U}$.
Trois types de réacteurs et de combustibles différents sont couverts par les interprétations des analyses d’échantillons ponctuels.
Dix types de réacteurs et de combustibles différents sont couverts par les interprétations des analyses de dissolutions globales des assemblages.

Ces interprétations effectuées à l’aide de calculs neutroniques très sophistiqués [3], permettant la prise en compte détaillée des effets locaux de spectre, ont permis de mettre en évidence un certain nombre de tendances sur les calculs d’évolution des actinides [4]. Tendances qui ont été confirmées par des études effectuées sur les résultats d’irradiation d’isotopes séparés, dans des spectres de réacteurs à eau [5].

Lors de l’interprétation des analyses de dissolution des assemblages en utilisant le calcul simplifié de l’assemblage moyen, les tendances habituelles sur les évolutions calculées des actinides ont été retrouvées.

Nous donnons pour les principaux rapports isotopiques, dans le tableau I, les écarts entre les tendances obtenues lors de l’interprétation fine des échantillons ponctuels et celles obtenues par l’interprétation d’analyses d’assemblages dissous globalement à l’aide du modèle de l’assemblage moyen.

Le fait que ces écarts soient faibles permet d’affirmer que les approximations effectuées lors des calculs de type assemblage moyen n’introduisent aucune erreur systématique importante sur les prévisions d’évolution des actinides à l’échelle d’assemblages entiers.

L’autre aspect essentiel de la validation des calculs de corrélation réside dans la justification de l’invariance des fonctions de corrélation aux fluctuations des paramètres déterminant l’historique du spectre d’irradiation.
TABLEAU II. EFFETS INDUITS SUR LES FONCTIONS DE CORRELATION PAR UN PREMIER COEUR (EN %)

<table>
<thead>
<tr>
<th>Types de fonctions de corrélations</th>
<th>Cycle 1</th>
<th>Cycle 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Taux de combustion</td>
<td>a 1,1</td>
<td>2,2</td>
</tr>
<tr>
<td></td>
<td>b &gt;1,</td>
<td>&gt;1,</td>
</tr>
<tr>
<td>Pu/U</td>
<td>h 4,1</td>
<td>3,3</td>
</tr>
<tr>
<td></td>
<td>δ 2,1</td>
<td>1,3</td>
</tr>
</tbody>
</table>

TABLEAU III. EFFETS INDUITS SUR LES CONSTANTES DE COHERENCE PAR UN PREMIER COEUR

<table>
<thead>
<tr>
<th>Constantes de cohérence</th>
<th>Cycle 1</th>
<th>Cycle 2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cl</td>
<td>0</td>
<td>-0,6</td>
</tr>
<tr>
<td>C2</td>
<td>-5,0</td>
<td>-3,0</td>
</tr>
<tr>
<td>C3</td>
<td>+3,7</td>
<td>+1,0</td>
</tr>
<tr>
<td>C4</td>
<td>+3,0</td>
<td>+1,4</td>
</tr>
<tr>
<td>C5</td>
<td>+1,1</td>
<td>+1,6</td>
</tr>
<tr>
<td>C6</td>
<td>+0,6</td>
<td>-1,2</td>
</tr>
<tr>
<td>C7</td>
<td>-0,4</td>
<td>-2,3</td>
</tr>
<tr>
<td>C8</td>
<td>-2,8</td>
<td>-2,4</td>
</tr>
</tbody>
</table>

a Ecarts systématiques induits sur les valeurs des constantes de cohérence interne par l’appartenance du combustible considéré aux deux premiers cycles du réacteur (régime transitoire) par rapport aux valeurs de celles-ci pour les cycles suivants (régime d’équilibre). Notons que certaines de ces constantes de cohérence n’en sont encore qu’au stade de développement.

Ces paramètres peuvent se classer en trois catégories:

1) Ceux qui n’interviennent que lors des premiers cycles du réacteur, comme les poisons consommables amovibles, et dont les effets peuvent être ou ne pas être pris en compte dans le calcul de base de l’assemblage moyen selon la précision que l’on recherche au niveau des corrélations.
TABLEAU IV. DISPERSION INDUITE SUR LES PRINCIPALES FONCTIONS DE CORRELATION (A) ET LES VALEURS DES CONSTANTES DE COHERENCE INTERNE (B) PAR LES FLUCTUATIONS DE L'HISTORIQUE D'IRRADIATION\(^a\)

<table>
<thead>
<tr>
<th>Fonctions de corrélations</th>
<th>Dispersion en % (1(\sigma))</th>
<th>Constante de cohérence</th>
<th>Dispersion à l'équilibre en %</th>
</tr>
</thead>
<tbody>
<tr>
<td>h</td>
<td>1,3</td>
<td>C1</td>
<td>4,1</td>
</tr>
<tr>
<td>h</td>
<td>1,3</td>
<td>C2</td>
<td>1,3</td>
</tr>
<tr>
<td>h</td>
<td>1,3</td>
<td>C3</td>
<td>1,8</td>
</tr>
<tr>
<td>C</td>
<td>1,8</td>
<td>C4</td>
<td>2,7</td>
</tr>
<tr>
<td>C</td>
<td>1,8</td>
<td>C5</td>
<td>0,7</td>
</tr>
<tr>
<td>C</td>
<td>1,8</td>
<td>C6</td>
<td>1,8</td>
</tr>
<tr>
<td>C</td>
<td>1,6</td>
<td>C7</td>
<td>1,6</td>
</tr>
<tr>
<td>C</td>
<td>2,3</td>
<td>C8</td>
<td>2,3</td>
</tr>
</tbody>
</table>

\(^a\) En excluant les effets propres aux deux premiers cycles du réacteur.

On donne dans le tableau II les effets moyens induits pour les principales fonctions de corrélation par les perturbations dues aux conditions particulières de fonctionnement d'un premier cœur.

On donne dans le tableau III les effets moyens induits sur les valeurs des constantes de cohérence interne. On remarque que la plupart de celles-ci sont insensibles à ce type de perturbation. De ce fait un décalage sensible affectant seulement certaines constantes confirmera que le combustible considéré provient bien d'un cœur de démarrage.

2) Ceux qui interviennent dans les éléments de base de la description du combustible comme la teneur initiale en \(^{235}\)U ou les caractéristiques physiques du combustible. Ces éléments de base ont été réduits à un nombre minimum, mais leur connaissance exacte, aux tolérances de fabrication près, est nécessaire si l'on ne veut pas introduire des erreurs systématiques importantes lors de l'utilisation des corrélations. Les effets induits par des variations de ces paramètres ont été présentés dans une autre communication [1].

3) Ceux qui représentent l'historique de fonctionnement du réacteur, comme la température moyenne du modérateur, le mouvement des barres, la puissance de fonctionnement, dont les valeurs peuvent varier fortement autour de la
valeur nominale ou moyenne. Ces paramètres sont à l'origine d'erreurs aléatoires qui disparaissent par compensation sur des lots de combustible suffisamment importants. De plus, l'utilisation de plusieurs fonctions de corrélation, dont les sensibilités à ces types de perturbation sont différentes, permet de réduire sensiblement la dispersion induite par ces effets.

Nous donnons dans le tableau IV les dispersions induites sur les principales fonctions de corrélation et sur les valeurs des constantes de cohérence interne par les fluctuations de l'historique d'irradiation. Notons que le dispersion de 4% affectant Cl est due à la présence du $^{238}\text{Pu}$ comme paramètre.

En résumé, en dehors de la précision des mesures de composition isotopique, une bonne qualification du calcul d’évolution des actinides et une connaissance exacte des quelques données de base du combustible et de son irradiation permettent de réduire la partie systématique des erreurs affectant la technique des corrélations à sa plus simple expression.

La composante aléatoire peut également se réduire fortement par le choix judicieux du jeu de fonctions utilisées, intégrant les effets particuliers du premier cœur et atténuant les fluctuations par effet de compensation.

4. MISE EN ŒUVRE DES CORRELATIONS ISOTOPIQUES

Elle comprend deux étapes dans le temps:

_L’étape préliminaire_, qui repose sur l'utilisation de calculs d'évolution neutronique des assemblages. Elle permet de fournir les valeurs des constantes de cohérence interne et les coefficients des fonctions de corrélation [6]. La durée de cette étape préliminaire dépend du délai nécessaire à la compilation des caractéristiques générales du combustible (qui par leur caractère très général et leur nombre restreint sont d'un accès facile vis-à-vis des constructeurs et des exploitants de réacteurs) et du temps nécessaire à la mise en œuvre du calcul de l'assemblage moyen et des codes de corrélation associés.

_L’étape en temps réel_, où l'on collationne tout d'abord toutes les données relatives à la dissolution considérée.

Une partie de ces données, concernant les caractéristiques physiques des assemblages dissous, a une origine extérieure au retraitement qui nécessitera un contrôle pour éviter toute erreur lors des transmissions des données ou de la manipulation des assemblages.

L'autre partie de ces données comprend, d'une part, des paramètres propres au procédé et, d'autre part, les analyses obtenues par les laboratoires de l'usine.
A l'aide de l'ensemble de ces données, des tests de cohérence interne sont effectués. Le fait de considérer, dans un premier temps, les données d'origine interne comme correctes permet d'attribuer, le cas échéant, à un paramètre d'origine externe bien précis la cause de l'anomalie (teneur initiale $^{235}$U et $^{236}$U, temps de désactivation, etc.).

En cas de persistance de celle-ci, le raisonnement inverse est effectué en considérant que l'anomalie provient des données internes et que les données d'origine externe sont correctes. Il est en général possible, soit de vérifier le paramètre en cause, soit d'effectuer une nouvelle analyse. Si l'anomalie subsiste, l'ensemble des données concernant cette dissolution sera à considérer avec une certaine prudence, des erreurs multiples pouvant s'y être introduites.

Remarquons que, dans le cas d'une utilisation de cette procédure par un organisme extérieur à l'usine, aucune hiérarchie ne pouvant être établie dans la validité des données, seul le critère de dissolution présentant certains caractères anormaux pourra être retenu.

Enfin la dernière phase consistera à appliquer les coefficients de corrélation précédemment choisis sur les résultats d'analyses isotopiques pour obtenir les paramètres nécessaires à l'établissement des bilans d'entrée [7] (taux de combustion et rapport Pu/U).

5. DESCRIPTION DES FONCTIONS DE CORRELATION

5.1. Fonctions utilisées pour la détermination du taux de combustion

Le taux de combustion est évalué par deux fonctions de corrélation basées sur la composition isotopique de l'uranium.

La première (a), basée sur l'appauvrissement en $^{235}$U, ne présente aucune incertitude de mesure mais une légère sensibilité aux fluctuations du spectre d'irradiation.

La seconde (b), basée sur l'accroissement en $^{236}$U, est au contraire insensible aux effets de spectre mais par contre plus sujette à la précision analytique.

Les teneurs initiales en $^{235}$U et $^{236}$U sont vérifiées à l'aide des constantes de cohérence interne.

De plus, une troisième fonction de corrélation basée sur la composition isotopique du plutonium nous donne une valeur de recouplement du taux de combustion:

$$I = a_E(I) \times \left[ \frac{U_{50}}{U_0} - \frac{U_5}{U} \right]$$
<table>
<thead>
<tr>
<th>Type de réacteur</th>
<th>Nombre de réacteurs</th>
<th>Nombre de types de combustible</th>
<th>Gamme de variation des taux de combustion</th>
<th>Gamme de variation des enrichissements (%)</th>
<th>Gamme de variation des temps de refroidissement</th>
<th>Nombre de dissolution</th>
</tr>
</thead>
<tbody>
<tr>
<td>PWR</td>
<td>11</td>
<td>6</td>
<td>12 000 à 37 000 MW·d/t</td>
<td>1,9 à 3,4</td>
<td>2 à 7 ans</td>
<td>≈ 500</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BWR</td>
<td>3</td>
<td>3</td>
<td>10 000 à 27 000 MW·d/t</td>
<td>2,2 à 2,8</td>
<td>2 à 7 ans</td>
<td>≈ 200</td>
</tr>
</tbody>
</table>

* Plus de 700 dissolutions.
où $I$ est le taux de combustion massique
et $E$ la teneur initiale en $^{235}$U du combustible.

5.2. Fonctions utilisées pour la détermination du rapport Pu/U

Le rapport Pu/U est évalué par deux fonctions de corrélation basées
respectivement sur les compositions isotopiques de l'uranium et du plutonium,
le taux de combustion obtenu précédemment par corrélation n'intervenant que
comme une correction du deuxième ordre.
La première fonction, notée $h$, basée sur l'appauvrissement en $^{235}$U (dont
la mesure est très précise) est quasi indépendante du taux de combustion mais
présente une certaine sensibilité aux effets de spectre (en particulier, la
présence de poisons lors des premiers cycles d'un réacteur demande un
calcul particulier si l'on désire une précision élevée des résultats).
La deuxième fonction, notée $\delta$, basée sur la composition en isotopes de
période longue du plutonium (dont la mesure de certains isotopes peut se
montrer assez délicate: le $^{242}$Pu par exemple) est quasiment insensible aux
effets de spectre mais présente néanmoins une légère sensibilité au taux de
combustion pour des combustibles peu irradiés.
L'utilisation de la moyenne de $h$ et de $\delta$ permet ainsi de garantir une
stabilité dans la précision des résultats obtenus malgré la présence éventuelle
de perturbations:

\[
\frac{Pu}{U} = h_E(I) \times \left[ \frac{U_{50}}{U} - \frac{U_{5}}{U} \right]
\]

\[
\frac{Pu}{U} = \delta_E(I) \times \left[ \frac{^{239}Pu \times ^{242}Pu}{(^{240}Pu)^2} \right]
\]

6. RESULTATS OBTENUS

La procédure d'application des corrélations isotopiques au contrôle du
bilan d'entrée décrite dans le paragraphe précédent a été appliquée sur l'ensemble
des dissolutions de combustible à eau légère effectuées à l'usine de La Hague
depuis plus de 4 ans. Ce qui représente environ 600 tonnes d'UO₂ retraitées.
Sauf pour la première campagne, qui représente moins de 14% de
l'ensemble du combustible retraité, cette procédure a été utilisée en ligne en
fournissant à priori les jeux de coefficients de corrélation. Naturellement, à
la suite de nombreuses améliorations apportées à la méthode, de nouveaux dépouillements ont pu être effectués sur des résultats provenant d'anciennes campagnes.

L'utilisation, dans une première étape, des constantes de cohérence interne a permis de détecter immédiatement la présence d'erreurs dans la transmission des données utilisées (98% des anomalies détectées), qu'elles concernent les analyses ou les caractéristiques des assemblages dissous.

Il a été possible de détecter également, dans les résultats d'analyses, des anomalies qui ont disparu après de nouvelles mesures. La rareté de ces anomalies ne permet pas leur étude statistique.

Nous présentons dans le tableau V la gamme des caractéristiques des combustibles eau légère dissous. Remarquons qu'elle contient la quasi-totalité des types de combustible PWR actuellement en service.

Notons que l'un des types de combustible BWR dissous contenait du gadolinium.

TABLEAU VI. ECARTS MOYENS ET DISPERSIONS ENTRE LES TAUX DE COMBUSTION OBTENUS PAR LES CORRELATIONS ET CEUX ANNONCES PAR CHAQUE REACTEUR (PAR REACTEUR ET PAR COMPAGNE)\(^a\)

VI-A. COMBUSTIBLES DE BWR

<table>
<thead>
<tr>
<th>Type du combustible</th>
<th>Numéro du réacteur</th>
<th>Numéro de la campagne</th>
<th>Nombre de dissolutions</th>
<th>Ecarts moyen en %</th>
<th>Dispersion dissolution (1 ( \sigma )) en %</th>
</tr>
</thead>
<tbody>
<tr>
<td>6 X 6 (Siemens)</td>
<td>12</td>
<td>1</td>
<td>17</td>
<td>+0,3</td>
<td>2,1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2</td>
<td>33</td>
<td>-2,2</td>
<td>2,2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>3</td>
<td>29</td>
<td>-0,2</td>
<td>3,2</td>
</tr>
<tr>
<td>7 X 7 (GE)</td>
<td>13</td>
<td>1</td>
<td>17</td>
<td>-1,2</td>
<td>2,4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2</td>
<td>39</td>
<td>-2,9</td>
<td>3,2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>14</td>
<td>1</td>
<td>-0,6</td>
<td>2,2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2</td>
<td>32</td>
<td>-2,4</td>
<td>2,1</td>
</tr>
<tr>
<td>8 X 8 (GE)</td>
<td>13</td>
<td>3</td>
<td>34</td>
<td>-4,0</td>
<td>1,9</td>
</tr>
<tr>
<td>Gd</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

\(^a\) \( I_{\text{correl}} - \frac{I_{\text{réacteur}}}{I_{\text{réacteur}}} \) (en %)
### VI-B. COMBUSTIBLES DE PWR

<table>
<thead>
<tr>
<th>Type du combustible</th>
<th>Numéro du réacteur</th>
<th>Numéro de la campagne</th>
<th>Nombre de dissolutions</th>
<th>Écarts moyen en %</th>
<th>Dispersion par dissolution (1 σ) en %</th>
</tr>
</thead>
<tbody>
<tr>
<td>14 × 14 (Westing.)</td>
<td>1</td>
<td>1</td>
<td>14</td>
<td>-9,2</td>
<td>1,6</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>19</td>
<td>-2,7</td>
<td>0,9</td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>1</td>
<td>14</td>
<td>-6,5</td>
<td>3,0</td>
</tr>
<tr>
<td>14 × 14 (KWU)</td>
<td>3</td>
<td>1</td>
<td>29</td>
<td>-0,8</td>
<td>2,0</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>26</td>
<td>-1,1</td>
<td>1,7</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>12</td>
<td>-2,4</td>
<td>1,3</td>
<td></td>
</tr>
<tr>
<td>15 × 15 (KWU)</td>
<td>4</td>
<td>1</td>
<td>81</td>
<td>-2,3</td>
<td>2,5</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>24</td>
<td>-3,3</td>
<td>2,4</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>27</td>
<td>-1,7</td>
<td>1,1</td>
<td></td>
</tr>
<tr>
<td></td>
<td>5</td>
<td>1</td>
<td>52</td>
<td>-4,4</td>
<td>1,7</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>21</td>
<td>-1,8</td>
<td>1,6</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>18</td>
<td>-2,3</td>
<td>1,6</td>
<td></td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>1</td>
<td>19</td>
<td>+1,4</td>
<td>1,3</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>25</td>
<td>-2,5</td>
<td>1,8</td>
<td></td>
</tr>
<tr>
<td>15 × 15 (Westing.)</td>
<td>7</td>
<td>1</td>
<td>26</td>
<td>-2,6</td>
<td>2,0</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>23</td>
<td>-3,5</td>
<td>1,8</td>
<td></td>
</tr>
<tr>
<td></td>
<td>8</td>
<td>1</td>
<td>16</td>
<td>-7,0</td>
<td>1,0</td>
</tr>
<tr>
<td></td>
<td>9</td>
<td>1</td>
<td>7</td>
<td>-5,4</td>
<td>0,6</td>
</tr>
<tr>
<td>15 × 15 spécial (inox)</td>
<td>10</td>
<td>1</td>
<td>35</td>
<td>-2,5</td>
<td>1,2</td>
</tr>
<tr>
<td>17 × 17 (Westing.)</td>
<td>11</td>
<td>1</td>
<td>17</td>
<td>-3,2</td>
<td>1,7</td>
</tr>
</tbody>
</table>

### 6.1. Interprétation des taux de combustion obtenus par corrélation

En prenant comme référence les taux de combustion annoncés par les différentes centrales électronucléaires, nous avons calculé les écarts existants avec les taux de combustion obtenus par corrélation isotopique.

Nous présentons ces écarts dans les tableaux VI-A (pour les BWR) et VI-B (pour les PWR).
On y constate un écart systématique avec les valeurs annoncées par les exploitants de réacteurs: de $-1,7\%$ pour les BWR et de $-3,2\%$ pour les PWR.

Une partie de celui-ci pourrait provenir d'une mauvaise valeur des énergies moyennes libérées par fission.

6.2. Interprétation des rapports Pu/U obtenus par corrélation

En prenant comme référence les rapports Pu/U mesurés à l'usine de La Hague, nous avons calculé les écarts existants avec le rapport Pu/U obtenu par les corrélations isotopiques $(h + 5/2)$.

Nous présentons ces écarts dans les tableaux VII-A (pour les BWR) et VII-B (pour les PWR).

On y constate des écarts systématiques très faibles: $-0,5\%$ pour les BWR et $-0,3\%$ pour les PWR.

1 La pondération a été effectuée par type de combustible et par réacteur, le nombre de dissolutions n'ayant pas été pris en compte.

TABLEAU VII. ECARTS MOYENS ET DISPERSIONS ENTRE LES RAPPORTS Pu/U OBTENUS PAR LES CORRELATIONS ET CEUX MESURES AU DISSOLVEUR (PAR REACTEUR ET PAR CAMPAGNE)$^a$

VII-A. COMBUSTIBLES DE BWR

<table>
<thead>
<tr>
<th>Type du combustible</th>
<th>Numéro du réacteur</th>
<th>Numéro de la campagne</th>
<th>Nombre de dissolutions</th>
<th>Écart moyen en %</th>
<th>Dispersion par dissolution (1σ) en %</th>
</tr>
</thead>
<tbody>
<tr>
<td>6 X 6 (Siemens)</td>
<td>12</td>
<td>1</td>
<td>17</td>
<td>$-0,7$</td>
<td>1,4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2</td>
<td>33</td>
<td>$-1,4$</td>
<td>1,4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>3</td>
<td>29</td>
<td>$-0,2$</td>
<td>3,0</td>
</tr>
<tr>
<td>7 X 7 (GE)</td>
<td>13</td>
<td>1</td>
<td>17</td>
<td>$-0,3$</td>
<td>2,2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2</td>
<td>39</td>
<td>$-1,1$</td>
<td>2,7</td>
</tr>
<tr>
<td></td>
<td>14</td>
<td>1</td>
<td>23</td>
<td>$-0,4$</td>
<td>1,4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2</td>
<td>32</td>
<td>$-0,9$</td>
<td>2,1</td>
</tr>
<tr>
<td>8 X 8 Gd (GE)</td>
<td>13</td>
<td>3</td>
<td>34</td>
<td>$+1,4$</td>
<td>2,2</td>
</tr>
</tbody>
</table>

$^a$ $(\text{Pu/U})_{\text{corr}} - (\text{Pu/U})_{\text{analyses}} / (\text{Pu/U})_{\text{analyses}}$ (en %).
VII-B. COMBUSTIBLES DE PWR

<table>
<thead>
<tr>
<th>Type du combustible</th>
<th>Numéro du réacteur</th>
<th>Numéro de la campagne</th>
<th>Nombre de dissolutions</th>
<th>Écart moyen en %</th>
<th>Dispersion par dissolution (1 σ) en %</th>
</tr>
</thead>
<tbody>
<tr>
<td>14 X 14 (Westing.)</td>
<td>1</td>
<td>1</td>
<td>14</td>
<td>−0,0</td>
<td>0,8</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>19</td>
<td>−0,1</td>
<td>1,9</td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>1</td>
<td>14</td>
<td>−0,9</td>
<td>1,0</td>
</tr>
<tr>
<td>14 X 14 (KWU)</td>
<td>3</td>
<td>1</td>
<td>29</td>
<td>+1,1</td>
<td>1,0</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>26</td>
<td>−0,4</td>
<td>0,7</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>12</td>
<td>−0,5</td>
<td>0,8</td>
<td></td>
</tr>
<tr>
<td>15 X 15 (KWU)</td>
<td>4</td>
<td>1</td>
<td>81</td>
<td>+0,1</td>
<td>1,7</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>24</td>
<td>−0,2</td>
<td>1,1</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>27</td>
<td>−0,4</td>
<td>1,6</td>
<td></td>
</tr>
<tr>
<td></td>
<td>5</td>
<td>1</td>
<td>52</td>
<td>−0,2</td>
<td>2,4</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>21</td>
<td>+0,1</td>
<td>1,4</td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>18</td>
<td>−0,6</td>
<td>1,2</td>
<td></td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>1</td>
<td>19</td>
<td>+0,6</td>
<td>1,3</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>25</td>
<td>−0,3</td>
<td>1,3</td>
<td></td>
</tr>
<tr>
<td>15 X 15 (Westing.)</td>
<td>7</td>
<td>1</td>
<td>26</td>
<td>−0,6</td>
<td>1,1</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>23</td>
<td>+0,1</td>
<td>0,9</td>
<td></td>
</tr>
<tr>
<td></td>
<td>8</td>
<td>1</td>
<td>16</td>
<td>−0,7</td>
<td>0,6</td>
</tr>
<tr>
<td></td>
<td>9</td>
<td>1</td>
<td>7</td>
<td>−0,1</td>
<td>0,4</td>
</tr>
<tr>
<td>15 X 15 spécial (inox)</td>
<td>10</td>
<td>1</td>
<td>35</td>
<td>−2,8</td>
<td>1,1</td>
</tr>
<tr>
<td>17 X 17 (Westing.)</td>
<td>11</td>
<td>1</td>
<td>17</td>
<td>−0,9</td>
<td>0,9</td>
</tr>
</tbody>
</table>

Une partie de ces écarts systématiques provient des premières campagnes, pour lesquelles les effets de premier cycle ont été en partie négligés.

La précision espérée par dissolution unitaire sur l'obtention du rapport Pu/U est d'environ 2% pour du combustible BWR et de 1,4% pour du combustible PWR si l'on utilise les jeux de corrélation intégrant les effets d'empoisonnement du combustible pendant les premiers cycles.
TABLEAU VIII. ECART PAR CAMPAGNE ENTRE LES MASSE
DE PLUTONIUM OBTENUES A L'AIDE DU BILAN D'ENTREE
DE RECOUPEMENT ET LE BILAN DE REFERENCE (EN %)

<table>
<thead>
<tr>
<th>Campagne</th>
<th>Ecart global en %</th>
<th>Tonnage d’U métal</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>+0,05</td>
<td>70</td>
</tr>
<tr>
<td>2</td>
<td>-0,31</td>
<td>40</td>
</tr>
<tr>
<td>3</td>
<td>-0,52</td>
<td>140</td>
</tr>
<tr>
<td>4</td>
<td>+0,28</td>
<td>90</td>
</tr>
<tr>
<td>5</td>
<td>+0,16</td>
<td>170</td>
</tr>
<tr>
<td>Total</td>
<td>-0,04</td>
<td>≈510</td>
</tr>
</tbody>
</table>

6.3. Résultats obtenus lors de l’établissement d’un bilan d’entrée plutonium de recoupe
ment


Dans le tableau VIII, nous donnons, par campagne et globalement, les écarts obtenus entre les masses de plutonium données par le bilan d’entrée de référence de l’usine et celles calculées par corrélation isotopique.

Nous voyons que la moyenne des écarts est inférieure à 0,3% par campagne et à 0,04% globalement.

7. CONCLUSION

Depuis quatre ans, l’usine de retraitement de La Hague utilise la technique des corrélations isotopiques calculées pour la vérification et l’obtention des paramètres nécessaires à l’établissement des bilans d’entrée de type dit gravimétrique. Ce bilan étant contractuel pour l’usine, le but poursuivi consiste en une aide à l’exploitation afin d’obtenir et de contrôler des paramètres qui ne sont pas directement accessibles au niveau du procédé ou de recouper de façon indépendante des méthodes de mesure standard.
Dans le contexte d’une usine de retraitement dont le bilan d’entrée contractuel ne serait pas du type gravimétrique (volumétrique par exemple), la technique des corrélations isotopiques calculées permettrait d’obtenir, soit dans le cadre même de l’usine, soit dans le cadre d’un contrôle des garanties par un organisme extérieur, un bilan d’entrée de recoupement qui serait basé sur une méthode gravimétrique indépendante de celle utilisée par le retraiteur.

Dans cet esprit une application a été effectuée dans le cadre des tâches K et L de TASTEX sur plusieurs campagnes de retraitement de l’usine de Tokai-Mura [8].

Les résultats obtenus sont très encourageants, que ce soit dans le cadre de la détection d’anomalies parmi les données de dissolution ou pour l’obtention d’un bilan d’entrée de recoupement.

En conclusion, nous pensons que l’expérience acquise à l’usine de La Hague permet d’envisager pour d’autres usines l’application en temps réel des techniques de corrélation isotopique calculée au contrôle du bilan d’entrée.

Cette procédure simple et efficace serait appliquée dans le cadre d’un contrôle des garanties.

REFERENCES

ISOTOPE CORRELATION TECHNIQUES
FOR VERIFYING INPUT ACCOUNTABILITY
MEASUREMENTS AT A REPROCESSING PLANT

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Tokai Research Establishment,
Tokai-mura, Ibaraki-ken,
Japan

Abstract

ISOTOPE CORRELATION TECHNIQUES FOR VERIFYING INPUT ACCOUNTABILITY
MEASUREMENTS AT A REPROCESSING PLANT.

Isotope correlation techniques were studied to verify input accountability measurements
at a reprocessing plant. On the basis of a historical data bank, correlation between plutonium-
to-uranium ratio and isotopic variables was derived as a function of bumup. The bumup was
determined from the isotopic ratios of uranium and plutonium, too. Data treatment was there­
fore made in an iterative manner. The isotopic variables were defined to cover a wide spectrum
of isotopes of uranium and plutonium. The isotope correlation techniques evaluated important
parameters such as the fuel burnup, the most probable ratio of plutonium to uranium, and the
amounts of uranium and plutonium in reprocessing batches in connection with fresh fuel fabri­
cation data. In addition, the most probable values of isotope abundance of plutonium and
uranium could be estimated from the plutonium-to-uranium ratio determined, being compared
with the reported data for verification. A pocket-computer-based system was developed to
enable inspectors to collect and evaluate data in a timely fashion at the input accountability
measurement point by the isotope correlation techniques. The device is supported by battery
power and completely independent of the operator's system. The software of the system was
written in BASIC. The data input can be stored in a cassette tape and transferred into a higher
level computer. The correlations used for the analysis were given as a form of analytical
function. Coefficients for the function were provided relevant to the type of reactor and the
initial enrichment of fuel.

1. INTRODUCTION

Measurements of nuclear material input to a reprocessing plant, in particular of plutonium, are of great importance,
since it is the starting point of plutonium accountancy based on direct measurements. Isotope correlation techniques (ICTs)
may be applied to verification of the input measurement data. The present study is to develop a method of analysing data
from the input accountability measurements by means of the isotope correlation of heavy nuclides and to make a system
that can be operated in a very small computer and used by
safeguards inspectors to collect data generated at the key measurement point (KMP) and evaluate promptly the data in the field.

2. UNDERLYING CONDITIONS

2.1. Determination of uranium and plutonium masses

The input masses of uranium and plutonium to the reprocessing plant are usually determined from measurements of the total volume of the solution of spent fuel dissolved in nitric acid and the concentration of uranium and plutonium in the solution. On the other hand, these masses may be determined by the so-called plutonium-to-uranium ratio method without the need of any measurement of the solution volume and the absolute concentration of the nuclear materials. The basic relations are derived below.

The material balance of heavy metals in a nuclear fuel before and after irradiation is given by:

\[ U^0 = U + Pu + Tu + F \]  

where \( U^0 \) is the number of atoms of initially present uranium; \( U, Pu \) and \( Tu \) the number of atoms of uranium, plutonium and the other transuranium elements, respectively, after discharge from the reactor; and \( F \) the number of fissions occurring during irradiation.

The term \( Tu \) amounts generally to the order of 100ppm and may be neglected. Introducing definitions of burnup, \( B \), in fissions per initial metal atom (FIMA) units and the atomic ratio of plutonium to uranium, \( R: \) (a) \( B = F/U^0 \), (b) \( R = Pu/U \), the amount of uranium remaining in the fuel is given by:

\[ U = U^0 (1 - B)/(1 + R) \]  

Then the amount of plutonium is obtained by:

\[ Pu = U \cdot R \]  

2.2. Burnup determination

The established method of determining burnup of spent fuel from light water reactors (LWRS) requires one to measure the amounts of \(^{148}\text{Nd}\) formed as fission product by the isotope dilution mass spectrometry as well as those of uranium and plutonium. In the reprocessing plant, however,
measurements are often limited to uranium and plutonium, so that burnup has to be evaluated from those data only.

Burnup in FIMA units is given by

\[ B = F_{25}/U^0 + F_{28}/U^0 + F_{49}/U^0 + F_{41}/U^0 \]  

(4)

where

\[ F_{25}/U^0 = A_{28}^0 [R_{25}^8/28 + R_{26}^8/28 - \delta(R_{25}^8/28 + R_{26}^8/28)] \]

\[ F_{28}/U^0 = p(\delta/\alpha_{28})[1 + (A_{40} + A_{41} + A_{42}(1 + \alpha_{49} + \alpha_{41})/\alpha_{49})] \]

\[ F_{49}/U^0 = p(\delta/\alpha_{49})[A_{40} + A_{41} + A_{42}(1 + 1/\alpha_{41})] \]

\[ F_{41}/U^0 = p(\delta/\alpha_{41})A_{42} \]

\[ \delta = [1 + (Pu/U)(1 + A_{40}/\alpha_{49} + A_{41}/\alpha_{49} + A_{42}(1 + \alpha_{49} + \alpha_{41})/(\alpha_{49}\alpha_{41})) \]

\[ (1 + 1/\alpha_{28})/A_{28}^{-1} \]

\[ p = (A_{28}^9/A_{28}) (Pu/U) \]

where \( A \) is the isotope fraction in atom unit, \( F \) the number of fissions, \( R \) the atom ratio, \( Pu/U \) the atom ratio of total plutonium to uranium, \( \alpha \) the ratio of capture to fission occurring, and the first digit of the suffix gives the atomic number and the second one the mass number of the nuclides concerned.

The \( \alpha \) values depend upon burnup and type of reactor, and those obtained from burnup calculation of typical LWR fuels were fitted to a function:

\[ \alpha = a \cdot B^b \cdot \exp\{c \cdot B\} \]  

(5)

where \( B \) is the burnup, and \( a \), \( b \) and \( c \) are fitted parameters which are given in Table I.

2.3. Isotopic variables and correlation with plutonium-to-uranium ratio

For evaluating reprocessing batch data, two approaches might be possible. The first is to establish the isotope correlation according to historical data which have been accumulated from measurements on the dissolved spent fuels having similar characteristics to those in question. The
**TABLE I. LIST OF COEFFICIENTS OF $\alpha$ FUNCTION FOR LWR FUEL:**

\[ \alpha = a \cdot B^b \cdot \exp[c \cdot B] \]

<table>
<thead>
<tr>
<th>Reactor</th>
<th>$a$</th>
<th>$b$</th>
<th>$c$</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>6.81620( 0)</td>
<td>-1.53200(-2)</td>
<td>2.52114(-3)</td>
</tr>
<tr>
<td>BWR</td>
<td>5.72490(-1)</td>
<td>-5.57520(-3)</td>
<td>-2.70303(-4)</td>
</tr>
<tr>
<td>A</td>
<td>3.28890(-1)</td>
<td>-5.62480(-3)</td>
<td>4.70984(-4)</td>
</tr>
<tr>
<td>BWR</td>
<td>7.04545( 0)</td>
<td>-2.37940(-2)</td>
<td>-7.10656(-3)</td>
</tr>
<tr>
<td>A</td>
<td>5.69903(-1)</td>
<td>-2.45463(-3)</td>
<td>-1.72642(-3)</td>
</tr>
<tr>
<td>BWR</td>
<td>3.29206(-1)</td>
<td>-6.37420(-3)</td>
<td>-1.35435(-3)</td>
</tr>
</tbody>
</table>

The second approach is to use calculated results. For either historical or calculated data the following isotopic variables are computed:

\begin{align*}
V_1 &= 235U^0 - 235U \\
V_2 &= 236U - 236U^0 \\
V_3 &= \ln[(238U/235U)(235U^0/238U^0)] \\
V_4 &= 100 - 239Pu \\
V_5 &= 240Pu/239Pu \\
V_6 &= 241Pu/240Pu \\
V_7 &= (241Pu + 242Pu)/240Pu \\
V_8 &= 239Pu \cdot 242Pu/240Pu^2
\end{align*}

where the symbol of each isotope denotes its isotopic abundance in atom percent and the symbol $^0$ refers to its initial value. The decay correction of 241Pu is necessary for the cooling period after discharge from the reactor core.

Correlation between the respective isotopic variables and the plutonium-to-uranium ratio was studied with a historical data base compiled in the ICT Data Bank of JAERI. The
TABLE II. LIST OF COEFFICIENTS OF CORRELATION FUNCTION BETWEEN Pu/U RATIO AND ISOTOPIC VARIABLES: Pu/U = (v + μB)VX

<table>
<thead>
<tr>
<th>Type of reactor</th>
<th>Initial enrichment (%)</th>
<th>BWR</th>
<th>PWR</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>1.10</td>
<td>1.47</td>
</tr>
<tr>
<td>V1</td>
<td>v = 8.2149-3, 6.3873-3</td>
<td>5.7912-3, 3.8342-3</td>
<td>6.0746-3, 1.3395-3</td>
</tr>
<tr>
<td>V2</td>
<td>v = 3.5937-2, 3.6655-2</td>
<td>3.4899-2, 2.2125-2</td>
<td>2.5921-2, 4.4190-3</td>
</tr>
<tr>
<td></td>
<td>μ = 8.4051-1, -9.5652-2</td>
<td>-3.5531-1, 2.4853-1</td>
<td>3.9791-1, 7.2338-1</td>
</tr>
<tr>
<td>V3</td>
<td>v = 8.3770-3, 8.9557-3</td>
<td>1.1847-2, 1.0227-2</td>
<td>1.2743-2, 5.6396-3</td>
</tr>
<tr>
<td>V4</td>
<td>v = 1.4005-4, 1.5044-4</td>
<td>1.8334-4, 1.5747-4</td>
<td>2.0816-4, 4.5770-5</td>
</tr>
<tr>
<td></td>
<td>μ = 1.2849-3, -4.8073-4</td>
<td>-1.9589-4, 2.0033-3</td>
<td>-3.6641-4, 6.6163-3</td>
</tr>
<tr>
<td>V5</td>
<td>v = 1.5769-2, 1.7470-2</td>
<td>2.1931-2, 1.7185-2</td>
<td>2.5485-2, 6.8248-3</td>
</tr>
<tr>
<td></td>
<td>μ = -5.1309-2, -2.7616-1</td>
<td>-2.1049-1, 1.9996-1</td>
<td>-2.2545-1, 6.5142-1</td>
</tr>
<tr>
<td>V6</td>
<td>v = 7.5781-3, 6.2794-3</td>
<td>6.9152-3, 8.8753-3</td>
<td>7.1393-3, 2.8870-4</td>
</tr>
<tr>
<td></td>
<td>μ = 5.6580-1, 6.1232-1</td>
<td>4.8844-1, 2.9997-1</td>
<td>4.5836-1, 5.9133-1</td>
</tr>
<tr>
<td></td>
<td>μ = 2.1177-1, 2.8146-1</td>
<td>1.9755-1, 1.3124-1</td>
<td>1.4558-1, 3.7321-1</td>
</tr>
<tr>
<td>V8</td>
<td>v = 1.9200-2, 1.7262-2</td>
<td>1.6645-2, 1.4731-2</td>
<td>1.8839-2, 4.1965-3</td>
</tr>
<tr>
<td></td>
<td>μ = -1.0477-1, 2.2763-2</td>
<td>1.0249-1, 2.3273-1</td>
<td>-6.3891-2, 5.2681-1</td>
</tr>
</tbody>
</table>

correlation appeared to depend on burnup and it was approximately fitted to a first-order linear function within a limited range of burnup:

\[
Pu/U = (v + \mu B)VX \quad ; \quad X = 1, 2, \ldots, 8. \quad (6)
\]

The coefficients determined from the fitting varied in different initial enrichment and in different type of reactor. Table II shows the coefficients of Eq. (6) obtained so far for some spent BWR and PWR fuels. The values would be better established as more data are accumulated.

2.4. Data availability

Data of the input accountability measurements are generated batch by batch and must be timely verified. The measured data on a given reprocessing batch should include the following items:

(1) Total solution volume of relevant batch
(2) Volume of heel of previous batch
(3) Concentration and isotope abundance of uranium in the solution
(4) Concentration and isotopic abundance of plutonium in the solution with information of the measurement date.

In addition, further information should be provided to identify spent fuels in the batch and to verify the measured results:

(5) Reactor specifications - name, type, moderator and output power
(6) Fuel specifications - identification number of relevant fuel assemblies, and initial amounts and isotopic abundance of uranium loaded into the assemblies
(7) Irradiation history of the fuel assemblies - at least date of charge to and discharge from the reactor core.

3. Analysis of the input measurement data by ICTs

The data are analysed batch by batch. Accordingly results of the analysis for the previous batch have always to be kept and referred to to correct for mixing with the heel of the previous batch solution. The input accountability tank is usually different from the dissolver and some intermediate vessels are in between. The material of the dissolved fuels is, in general, not completely transferred into the accountability tank. The remaining part of the material will mix with the next dissolution batch. Correction for such mixing of the remaining part of the relevant batch to the next one in the pre-positioned vessels should also be considered. Because fundamental relations were derived on the atom basis, all data must be converted to the atom basis and are subjected to the analysis.

From the data thus corrected, burnup is obtained taking at first the measured plutonium-to-uranium ratio and a set of initial a values which correspond with those for fresh fuel. After the first estimate of burnup, the a values are computed by the use of Eq. (5) and the coefficients listed in Table I. The computation is continued iteratively to converge on reasonably constant values.

The next step is to calculate the isotopic variables as defined in section 2.3. By choosing an appropriate set of coefficients for Eq. (6) from Table II based on reactor type and initial enrichment, the plutonium-to-uranium ratio can be calculated from each isotopic variable. Consequently the eight data obtained for the ratio are treated in the following simple statistical manner.
First, the mean and its variance are calculated:

\[ m = \frac{1}{n} \sum x_i \]
\[ s_m^2 = \frac{\sum (x_i - m)^2}{n(n - 1)} \]

Then, deviation of each \( x_i \) from the mean \( m \) is compared with a limit - 3\( s_m \) would be a good value as the limit. Data that exceed the limit should be rejected and the same treatment is repeated until the rest of the data remain within the limited range.

The final average value of the plutonium-to-uranium ratio is substituted for the measured value that was taken at first to obtain burnup, and the \( \theta \) values, burnup and the ratio of plutonium to uranium are again calculated iteratively until the plutonium-to-uranium ratios converge within 1% difference.

From the most probable plutonium-to-uranium ratio eventually estimated by the procedures described above, the masses of uranium and plutonium input to the reprocessing batch can be determined by Eqs. (2) and (3) in connection with the initial mass of uranium. At the same time, the most probable isotope abundance of uranium and plutonium may also be computed from the finally determined plutonium-to-uranium ratio and the correlation with the isotopic variables.

4. Development of a small-computer-based system

The analysis described in the previous section can be performed in a small pocketable computer that may be very convenient for safeguards inspectors to bring in-field for examining and collecting the input accountability data generated every day.

To demonstrate the feasibility of such techniques, a software system was developed with a Sharp PC-1500 pocket computer which can be supported by battery power and be operated completely independently from the operator's system.

A flow chart of the system is shown in Fig. 1; an example of the output of the input and corrected data in Fig. 2; and an example of results of the analysis in Fig. 3. Just after printing out the input data one may compare these with original records and immediate correction is possible for input data errors. The input data and the results of batch-by-batch analysis are stored, as well as the cumulative masses of uranium and plutonium through a whole campaign, into a cassette tape. The stored data items are listed in Table III.
FIG. 1. Flow chart of the ICT analysis system using a Sharp PC-1500 pocket computer.
MEASURED \( \text{Pu}\-\text{U} = 4.5621\times10^{-3} \)
CALCULATED \( \text{Pu}\-\text{U} = 4.5623\times10^{-3} \)
BURNUP IN FINAL = 1.1212\times10^{-2}

--- DATA INPUT ---

**A**: CAMPAIGN NAME = C1
**B**: BATCH NAME = FU1-047
**C**: REACTOR TYPE = B

**A**: FINAL U CONCENT (g/L) = 186.36
**B**: FINAL U-235 (g/L) = 1.553
**C**: FINAL U-236 (g/L) = 0.18
**D**: FINAL U-238 (g/L) = 36.239

**E**: FINAL Pu CONCENT (g/L) = 0.018
**F**: FINAL Pu-238 (g/L) = 0.19
**G**: FINAL Pu-239 (g/L) = 75.13
**H**: FINAL Pu-240 (g/L) = 6.17
**I**: FINAL Pu-242 (g/L) = 1.872

**J**: DATE OF PU MEAS. = 800507
**K**: DATE OF DECAY NORMAL = 780831

**L**: TOTAL SOLN U (g) = 2373.37
**M**: HEEL SOLN U (g) = 3.535

**N**: SHIPPERS U (g) = 3733.6
**O**: INIT. U-234 (g) = 3737.6
**P**: INIT. U-235 (g) = 0

**Q**: INIT. U-236 (g) = 186.36
**R**: INIT. U-238 (g) = 1.553

IF FAILED DATA INPUT, CORRECT: e.g., \( \text{F}\times X \)

--- CORRECTED DATA ---

**A**: CAMPAIGN NAME = C1
**B**: BATCH NAME = FU1-047

**C**: REACTOR TYPE = B

**A**: C-TOTAL = 1.7784\times10^{-2}
**B**: U-234 (g) = 1.8904\times10^{-2}
**C**: U-235 (g) = 1.9517\times10^{-2}
**D**: U-236 (g) = 1.8923\times10^{-2}
**E**: U-238 (g) = 9.8235\times10^{-2}

**F**: Pu CONCENT (g/L) = 8.9782\times10^{-1}
**G**: Pu-238 (g/L) = 3.9662\times10^{-1}
**H**: Pu-239 (g/L) = 7.4885\times10^{-1}
**I**: Pu-240 (g/L) = 1.7978\times10^{-1}
**J**: Pu-241 (g/L) = 0.4972\times10^{-1}
**K**: Pu-242 (g/L) = 1.4665\times10^{-1}

READY CASSETTE, KEY-IN <CONT>.

*** END OF ICT PROG. ***

FIG. 2. Output of the input data and corrected data of a given reprocessing batch from the Sharp PC-1500 system.

FIG. 3. Output of results of the ICT analysis of a given batch from the Sharp PC-1500 system.
<table>
<thead>
<tr>
<th>X/W</th>
<th>(w)</th>
<th>DR(X)</th>
<th>ID(X)</th>
<th>CI(W)</th>
<th>CM(X)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1/A</td>
<td>ICT U total,g</td>
<td>BatchSerial No.</td>
<td>f.U concn.g/l</td>
<td>cum.U total,g</td>
<td>cum.Pu total,g</td>
</tr>
<tr>
<td>2/B</td>
<td>ICT 234U, wt%</td>
<td></td>
<td>f.234U, wt%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3/C</td>
<td>ICT 235U, wt%</td>
<td></td>
<td>f.235U, wt%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4/D</td>
<td>ICT 236U, wt%</td>
<td></td>
<td>f.236U, wt%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>5/E</td>
<td>ICT 238U, wt%</td>
<td></td>
<td>f.238U, wt%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>6/F</td>
<td>ICT Pu total,g</td>
<td></td>
<td>f.Pu concn.g/l</td>
<td>cum.Pu total,g</td>
<td>cum.Pu total,g</td>
</tr>
<tr>
<td>7/G</td>
<td>ICT 239Pu, wt%</td>
<td></td>
<td>f.239Pu, wt%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>8/H</td>
<td>ICT 240Pu, wt%</td>
<td></td>
<td>f.240Pu, wt%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>9/I</td>
<td>ICT 241Pu, wt%</td>
<td></td>
<td>f.241Pu, wt%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>10/J</td>
<td>ICT 242Pu, wt%</td>
<td></td>
<td>f.242Pu, wt%</td>
<td></td>
<td></td>
</tr>
<tr>
<td>11/K</td>
<td>ICT 243Pu, wt%</td>
<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>12/L</td>
<td>cum. U shipper,g</td>
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<tr>
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<td>Decay time, a</td>
<td></td>
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<td>Diss. heel: O.U</td>
<td></td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>15/O</td>
<td>UU(OU+Shipper)U</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>16/P</td>
<td>F235U, %</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>17/Q</td>
<td>F236U, %</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>18/R</td>
<td>F237U, %</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>19/S</td>
<td>F238U, %</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>20/T</td>
<td>U°, g</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>21/U</td>
<td>Burnup, FIMA</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>22/V</td>
<td>ICT Pu/U</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>23/W</td>
<td>Time</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>24/X</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>25/Y</td>
<td>BWR(0)/PWR(1)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>26/Z</td>
<td>BatchSerial No.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*) Corrected for heel in accountability tank.
**) Corrected for both heel in dissolver and in accountability tank.
***) Corrected also for decay.

f.-final; init.-initial; cum.-cumulative; ICT-results from ICT analysis; A-Avogadro’s number; at.fracn.-atom fraction.
Correlation between the plutonium-to-uranium ratio and the isotopic data has been studied by several workers in consideration of sensitivity, linearity and neutron spectral effects [1-5]. However, it would be a risk if a particular isotope was regarded as the only parameter to be correlated with the plutonium-to-uranium ratio, because isotopic data generated from routine measurements are often less credible, especially for minor isotopes. The proposed procedure in this work takes into consideration all available isotopic data which provide complementary information. It also makes for estimating the most probable isotopic abundance of uranium and plutonium and comparing with the measured value.

Experience gained so far at the Tokai Reprocessing Plant[6] has shown that cumulative results over a campaign composed of a number of dissolution batches agreed well with shipper's values as well as with measured product masses. Although differences between the masses obtained by the isotope correlation techniques and those measured by the reprocessor scattered more than five percent in batch-by-batch comparison due to incomplete transfer of the solution from dissolver and intermediate vessels, relative random errors associated with the cumulative masses of uranium and plutonium were small enough for one to look for a possible systematic bias. Detection of a 1-2% systematic difference would be practicable, when verification by the present method was applied to a whole campaign comprising several tens of batches. Furthermore, batch-by-batch investigation that can be made by the present system enables one to examine timely the data of masses and isotopic abundance of nuclear material measured at the input accountability tank in reprocessing plants. Cumulative results of the batch data provide the total masses for the whole campaign immediately in-field.

To reinforce the ICTs as real effective measures for safeguards at reprocessing plants, on-site applicability should be demonstrated. The procedures developed in the present work would be easily operated by safeguards inspectors with the software provided in a small portable computer and would enable them effectively to collect the data and to verify them in a timely fashion. Moreover, successive accumulation of the data measured by the safeguards authority as well as by the reprocessor and other laboratories should be continued to elaborate the correlations. Effectiveness and confidence of the ICTs will increase with increasing data base. Such an effort must be continued in the headquarters installing a large-computer-based data bank. At the same
time, burnup calculation with a sophisticated code would be useful for understanding and ensuring the correlations.

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REFERENCES

NEAR-REAL-TIME
MATERIAL ACCOUNTANCY

(Session 11)
Chairman

L. THORNE
NEAR-REAL-TIME MATERIAL ACCOUNTANCY
A technical status report

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Abstract

NEAR-REAL-TIME MATERIAL ACCOUNTANCY: A TECHNICAL STATUS REPORT.
Near-real-time (n.r.t.) material accountancy as applied to reprocessing facilities involves two major elements, the measurement or estimation of the in-process physical inventory at frequent intervals and the statistical evaluation of the resulting sequential material balance data. For most reprocessing facilities the bulk of the in-process inventory is normally in intermediate "buffer" tanks, and is directly measurable with little or no added effort. In contrast, the plutonium inventory in the solvent extraction system does not appear to be directly measurable. Variations in this inventory, however, could cause a reduction in the sensitivity of the sequential data analysis. Although some studies are in progress, to date an acceptable means of accounting for these variations has not been found. Statistical tests for evaluating sequential material balance data are still being studied at several laboratories, but consultants at a meeting in January 1982 agreed that under all circumstances there is some increase in both detection timeliness and detection sensitivity using n.r.t. accountancy. IAEA verification of operator-generated measurement data is an area requiring significantly increased effort. It is hoped that a major demonstration over a period close to a full year can be arranged starting in 1983.

1. INTRODUCTION

In late 1977 the IAEA Department of Safeguards adopted a set of significant quantity and detection timeliness criteria for provisional use in the planning and evaluation of IAEA
safeguards inspections. For separated plutonium, especially at larger reprocessing facilities, these criteria clearly were more stringent than could at that time be achieved on the basis of conventional material accountancy with cleanout physical inventories at frequencies of one to four times per year. Indeed it is generally agreed that measurement uncertainties, in particular those associated with measurement calibrations, are such that the significant quantity goal is unlikely ever to be achieved, using conventional material accountancy, for any but the smaller reprocessing facilities. It is also agreed that there is no conceivable possibility of taking cleanout physical inventories on the one to three week schedule mandated by the timeliness criterion.

Various alternatives have been suggested, and research in a number of areas continues. The alternative of interest in this paper is now generally referred to as near-real-time (n.r.t.) material accountancy. In January 1982 the IAEA invited consultants from ten Member States plus the European Community to a meeting to discuss the "Current Technical Status of Near-Real-Time Materials Accountancy" [1]. Discussion was expressly limited to an analysis, from a technical standpoint, of what n.r.t. accountancy potentially could do and what its remaining problems were, and not to any consideration, from a more political standpoint, of what its role in IAEA safeguards should be.

The primary objective of this paper is to report on the results of that meeting. The paper also reviews some of the on-going studies in the field, especially those which have not yet reached a stage where reportable results are available.

2. GENERATION OF N.R.T. ACCOUNTANCY DATA

It is convenient to divide n.r.t. material accountancy into two major sub-divisions. One relates to the practical question of whether it is feasible to generate material balance data, i.e. to calculate an apparent MUF using the material balance equation, on the desired frequency. The other relates to the more theoretical question of how best to evaluate the resulting MUF data. Clearly the concept cannot be used if it is not feasible to generate the required data; equally clearly it should not be used if generating the required data does not lead to a useful improvement in detection sensitivity or timeliness.

Although there has been some work related to the use of n.r.t. material accountancy in MOX fabrication plants, most
studies have related to the process area of a reprocessing plant, using the traditional input accountancy tank measurement data as input and the equally traditional plutonium nitrate product measurement data as output. These data are routinely generated for conventional material accountancy; its use for n.r.t. accountancy should normally occasion no increased operator effort.

N.r.t. accountancy may impose data timeliness requirements, however. Chemical analytical measurements sometimes are delayed to allow time for careful sample preparation and handling, and to allow the laboratory to carefully recheck its work. While this attention to accuracy in preference to speed is certainly desirable, n.r.t. accountancy timeliness considerations normally would require that analytical results be reported within at most a few days or one week. If this timeliness is not achievable using chemical analytical techniques, recourse may be had to the K-edge densitometer for product measurements or the X-ray fluorescence spectrometer for input measurements, either as a substitute for chemical analyses or to provide interim data pending completion of chemical analyses.

Thus as a general statement the question of whether it is feasible to generate material balance data at frequent intervals reduces to the question of whether it is possible to take in-process physical inventories with that frequency.

It is important to emphasize the words 'in-process' physical inventory. To some people the very word 'physical' seems to imply that the inventory must be based on plant shutdown and cleanout, with every effort being made to achieve the maximum possible accuracy in the inventory measurements. No such requirement is included in n.r.t. accountancy. Rather it is expressly understood that inventories will be taken in a manner which does not disturb process operations in any significant manner. Most proposed procedures in fact do not disturb process operations at all. The word 'physical' is retained, however, to call attention to the fact that the measurements or estimates used must nevertheless be independent of any consideration as to how much should be present. Inventory calculations based on the difference between inputs and outputs are no more acceptable in n.r.t. accountancy than they are in conventional material accountancy.

Studies related to the U.S. Barnwell plant, the Japanese PNC Tokai plant, and a reference design developed in the Federal Republic of Germany have all shown that the bulk of the in-process inventory at reprocessing plants will normally
be in intermediate buffer storage tanks. These tanks also are instrumented for process control measurements, and these routine measurements could be used as a basis for in-process inventory determinations.

One or two comments are desirable. The existence of buffer storage tanks is not essential for n.r.t. material accountancy. Tanks are included by designers/operators who want the operational flexibility which such tanks provide. Omission of one or more of these tanks reduces the fraction of the in-process inventory which is in tanks, but the more important effect is that it reduces the total in-process inventory. Decisions as to where to locate intermediate tanks, how big to make them, etc., are purely designer/operator decisions. The n.r.t. accountancy requirement is that where tanks exist they be instrumented for in-process inventory measurement, and the evidence is that logically they will be so instrumented.

While the bulk of the inventory is normally in buffer storage tanks, a significant remaining fraction is in the solvent extraction systems, and studies thus far indicate that inventory estimation in these systems may present a problem. The question relates not to how much plutonium may be in the solvent extraction systems, but to how constant this quantity may be. A truly constant inventory will cancel out between beginning and ending inventories, and have no effect on detection sensitivity. An inventory which varies over a wide range has the same effect as a measurement with an equally wide uncertainty and can seriously reduce the overall detection sensitivity achievable.

One solution to the solvent extraction system inventory problem would be to use centrifugal contactors instead of mixer-settlers or pulse columns. Centrifugal contactors have very low residence times and correspondingly low inventories, and can be omitted from in-process inventory measurements. Unfortunately it is doubtful whether designers would accept this idea, because the chemical dynamics are such that three stages of centrifugal contactors probably would not give adequate fission product removal.

Plant operators are primarily concerned with balancing flow-rates between the two extremes of emulsion formation and system flooding, and have little interest in plutonium inventory quantities. The result is that there are few valid data relating to inventory fluctuations in solvent extraction systems. However, some significant factors can be defined.
a) **Burnup level of spent fuel processed.** The primary component of spent fuel is uranium, which has a solubility limit in nitric acid solution in the range of 200 g/l. At 30 000 MW.d/t the corresponding plutonium concentration is on the order of 2 g/l. At 10 000 MW.d/t it may be only 1 g/l. In the first extraction cycle these variations translate directly into inventory variations. Later extraction cycles may be somewhat less affected, but to a first approximation variations in feed concentration are transmitted through all cycles until the product evaporator is reached.

b) **Dissolver operating conditions.** The exact quantity of fuel charged to a dissolver, the quantity of acid used, and other operating variables are subject to a considerable measure of operator control. Two nominally identical fuel assemblies, one dissolved in the minimum possible quantity of acid and the other dissolved in a larger quantity of acid under slightly different operating conditions, can lead to uranium and plutonium concentration differences in the range of 10-25%. Since solvent extraction system operation is governed largely by total volume flows, these concentration differences again translate directly into corresponding variations in the plutonium inventory in the solvent extraction systems.

c) **Operator-controlled process variables.** The process operator has the ability to vary a number of parameters which will affect the total plutonium inventory. Examples include varying the flow rates of aqueous feed, organic extractant, nitric acid scrub, or aqueous strip solutions. Since a solvent extraction system operates full (to overflow lines) at all times, varying one flow rate does not necessarily require varying a second, and it is difficult to generalize on the effect on inventory.

d) **Non-equilibrium operation.** If the process operator has varied one or more process parameter to too great an extent, any of several process upset conditions may occur. The most common is flooding, which usually is caused by an insufficient pulse stroke relative to input flow rates. The result, in terms of material accountancy, is that the plutonium inventory first increases significantly, then decreases as the accumulated plutonium leaves the system via the aqueous waste stream.

e) **Distribution coefficient changes.** A solvent extraction system that has recently been charged with clean organic extractant has a distribution coefficient which is determined by the nature and concentration of the organic
extractant, the nitric acid concentration, temperature, and the competing influence of fission products. As this system continues to operate, crud formation occurs, leading to a gradual deterioration of the distribution coefficient. Deterioration of the distribution coefficient in turn leads to a flattening of the aqueous/organic distribution curve, and to a gradual increase in the plutonium inventory.

All of this says that there is a problem, and that it cannot be made to disappear through wishful thinking. What to do about it is less clear. Early discussions centered around computer programs which were developed to model the complete chemical dynamics of solvent extraction systems. Because the chemical dynamics of solvent extraction are complex, these computer programs are also complex, and running them on a routine basis would be almost prohibitively expensive. There is also an argument that such programs give inspectors an undesirable access to reprocessing technology. Certainly they produce considerably more technological information than the inspector needs, since the inspector's sole legitimate interest is a single number representing the plutonium inventory totalled across all stages and phases.

A highly simplified model which might be used to estimate the total plutonium inventory has been suggested, but by no means demonstrated. The model consists of two terms, one representing the plutonium inventory under the assumption of equilibrium clean conditions, and the other representing a correction for crud formation, non-equilibrium conditions, etc.

The first term is not difficult to derive, and in fact consists of nothing more than the aqueous volume of the system multiplied by the feed plutonium concentration. The aqueous volume is calculable, given total system volume and relative aqueous and organic flow rates, and should remain relatively constant once the operator has found a set of operating conditions which produce desired results. The feed plutonium concentration, of course, is the same as the input accountancy tank concentration for the first extraction cycle, and is easily obtainable from buffer feed tanks for the other cycles. Thus while there is still some question concerning the constancy of the aqueous volume factor the first term of the simplified model is on relatively solid theoretical and practical ground.

The second term is more questionable. It can be argued that the design ideal equilibrium conditions necessarily correspond to minimum plutonium inventory conditions, and that therefore the second term must in some manner represent an
additional plutonium inventory associated with deviations from ideal equilibrium conditions. It can also be argued that any departure from ideal equilibrium conditions must necessarily shift the tail of the plutonium distribution curve in the direction of an increased Pu concentration in the aqueous waste. Thus the suggestion is that the second term can in some way be represented by a factor, as yet not defined, times the aqueous waste plutonium concentration.

Considerable research is still needed in this area. Despite the theoretical arguments given above, the actual magnitude of variations in solvent extraction system plutonium inventories is not well known, and it may be that the effort to measure or estimate this variation will itself introduce an uncertainty comparable to the magnitude of the variation, resulting in little or no real improvement in detection sensitivity. Measurement uncertainties in the range of 20% or more for aqueous waste concentrations are considered good, and the desired effect may be buried in measurement error noise. Also, solvent extraction systems deliberately include extra stages to ensure good separation in spite of crud formation, and it may be that these extra stages will prevent the predicted effect from being observable. Another disturbing possibility is that the effect on total plutonium inventory may depend on the nature of the process variation which caused the high waste concentration. Some studies are in progress, many more are needed.

3. EVALUATION OF MUF DATA FROM N.R.T. ACCOUNTANCY

In the early development of the use of statistical tests in nuclear material safeguards, it was common to state that diversion 'detection' occurred whenever the test statistic led to rejection of the null hypothesis. It was universally understood that the IAEA would not immediately and automatically report to the Board of Governors on the basis of this single rejection, but nevertheless discussions centered almost totally around selection of the one best statistical test with the desired detection and false alarm probabilities.

In contrast to conventional material accountancy, in which only one or two data points per year exist to be evaluated, an n.r.t. material accountancy system may be expected to provide tens or even one hundred or more data points per year. The analysis and evaluation of such data sequences is a new problem in material accountancy, one requiring the development of new decision processes to take maximum advantage of the additional information available.
Some five statistics have been identified as being of probable value. These are:

- The magnitude of an individual MUF
- The cumulative sum of a sequence of MUF values (CUMUF)
- The cumulative sum of a sequence of residuals computed by subtracting from each MUF value a quantity representing a predicted value based on past data
- An estimated mean value for a sequence of MUF values, based on past data
- The slope of a line describing the cumulative sum of a sequence of MUF values or MUF residuals (the original CUSUM).

These statistics may be evaluated either by comparison with a variance based on recognized measurement uncertainties, the traditional variance/covariance matrix, or by comparison with a variance derived from squared deviations of past data about their own mean. In the absence of diversion these two variances should be the same. Accordingly, in addition to the tests listed above one final test which has been suggested would be a comparison of these two variances with each other.

These various statistics and the possibilities of their evaluation have been the subject of extensive studies and discussions. In most cases it is not possible to write analytically manageable expressions for detection sensitivity, and extensive Monte Carlo calculations have been performed. These studies are still in progress, and the reader is referred to another paper in this symposium [2], or to the references given in the previously referred to consultants' meeting report [1]. Some preliminary conclusions are possible, however.

- For an assumed abrupt diversion during one short-term material balance period, both the reduced quantity of material processed during the short period and the timeliness of the material balance assure that significantly improved detection sensitivity and timeliness will exist.
- For an assumed uniform protracted diversion over a limiting period of one year, the availability of sequential material balance data assures that an improved detection sensitivity will exist. This is especially true for the case where no anomalies are observed. In such cases the assurance of non-diversion can be significantly greater than for a comparable conventional material accountancy system [3].
- For the wide variety of intermediate diversion possibilities that might be assumed it is not possible to
express general conclusions regarding either detection sensitivity or detection timeliness. However, it is clear that in all cases there should be some improvement. In particular, in the limiting case in which anomalies are observed that cannot be resolved without recourse to a clean-out physical inventory, the IAEA will still have some improved detection sensitivity because the material balance will include only the period of time which is in question. Measurement uncertainties will not be inflated through the unnecessary inclusion of data relating to periods during which diversion is not suspected.

As with the frequency of measurement of in-process physical inventories, there is still work to be done. One important problem is the question of whether the two limiting cases, abrupt diversion in one period or uniform diversion over a twelve month period, are indeed limiting in the sense of defining minimum detection probabilities. It may be that all other diversion patterns in fact have a higher detection probability than either of the two limiting cases, or it may be that some other diversion pattern can be defined which represents the limiting case. At the present time all that can be said is that detection sensitivity is dependent on an assumed diversion pattern, and that any given test may not be optimum, indeed probably would not be optimum, if the wrong pattern is assumed.

4. VERIFICATION OF N.R.T. ACCOUNTANCY DATA

If one assumes that material accountancy data have been correctly stated (i.e. have not been falsified), then any diversion must necessarily be reflected in the expectation value of the computed material unaccounted for (MUF), and have a probability of detection defined by the magnitude of the diversion, the magnitude of the uncertainties in the measurements used to close the material balance, and the statistical equations used to combine measurement uncertainties and evaluate MUF. It is agreed, of course, that IAEA safeguards cannot be based on the assumption that material accountancy data have not been falsified. It is important to recognize, however, that with any material accountancy safeguards system the question is not how the diverted material might be removed from the facility, but how the diverted quantity might be prevented from appearing as MUF.

There are in fact very few verification problems in n.r.t. material accountancy that do not already exist in conventional material accountancy. This does not make the
problems any less real, unfortunately. The fundamental problem in reprocessing, the fact that most of the work takes place behind heavy shielding and cannot be verified visually, does not have a simple solution.

There has been relatively little work directed specifically at the question of verification for n.r.t. material accountancy. It can be shown fairly easily that the problem divides into two sub-problems: rapid approximate verification to provide timeliness with regard to the larger falsification needed to conceal abrupt diversion, and highly accurate but not necessarily rapid verifications to provide a detection capability for small biases introduced to conceal protracted diversion. There does not appear to be any need for verifications which are both rapid and highly accurate.

5. A LOOK TO THE FUTURE

A number of future projects are under way, or are being planned. The PNC Tokai work assumed that in-process inventories would be taken at fixed time intervals. A United Kingdom study is looking at in-process inventories taken at fixed process intervals, e.g. immediately prior to input accountancy tank measurements. The argument is that the in-process inventory at that time should be near a minimum and should be more nearly stable. Preliminary results are indeed very interesting, and there should be some publishable data in 1983.

As is reported in Ref. [2], a re-examination of statistical thinking regarding calibrations and associated systematic uncertainties is under way, and also seems likely to lead to useful results in 1983.

An advanced safeguards approach using n.r.t. material accountancy in the process material balance area and incorporating ideas from TASTEX, the IWC-RPS, or elsewhere has been drafted and is currently being evaluated using the safeguards effectiveness assessment methodology. It is hoped that elements of this safeguards approach, especially the n.r.t. material accountancy system, can be demonstrated through a full-scale field test starting sometime in 1983.

In 1977, n.r.t. material accountancy was an idea, a gleam in the eyes of a few people who thought they could make it work, and that doing so would improve the effectiveness of IAEA safeguards. Today, five years later, it is suggested that n.r.t. material accountancy is a concept whose time has
come. There are remaining problems, but there are remaining problems in all areas of safeguards, and most of these problems are of the type that are best investigated and resolved as part of a programme of implementation.

REFERENCES


A NEAR-REAL-TIME MATERIAL ACCOUNTANCY MODEL AND ITS PRELIMINARY DEMONSTRATION IN THE TOKAI REPROCESSING PLANT

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Abstract

A NEAR-REAL-TIME MATERIAL ACCOUNTANCY MODEL AND ITS PRELIMINARY DEMONSTRATION IN THE TOKAI REPROCESSING PLANT.

The study of a near-real-time (n.r.t.) material accountancy system as applied to small or medium-sized spent fuel reprocessing facilities has been carried out since 1978 under the TASTEX programme. In this study, a model of the n.r.t. accountancy system, called the ten-day-detection-time model, was developed and demonstrated in the actual operating plant. The programme was closed on May 1981, but the study has been extended. The effectiveness of the proposed n.r.t. accountancy model was evaluated by means of simulation techniques. The results showed that weekly material balances covering the entire process MBA could provide sufficient information to satisfy the IAEA guidelines for small or medium-sized facilities. The applicability of the model to the actual plant has been evaluated by a series of field tests which covered four campaigns. In addition to the material accountancy data, many valuable operational data with regard to additional locations for an in-process inventory, the time needed for an in-process inventory, etc., have been obtained. A CUMUF (cumulative MUF) chart of the resulting MUF data in the C-1 and C-2 campaigns clearly showed that there had been a measurement bias across the process MBA. This chart gave a dramatic picture of the power of the n.r.t. accountancy concept by showing the nature of this bias, which was not clearly shown in the conventional material accountancy data.

1. INTRODUCTION

From spring 1978 to May 1981 the governments of Japan, France and the United States of America mutually pursued a programme for improving safeguards techniques as applied to spent fuel reprocessing facilities, with particular emphasis on the applicability of these techniques to the PNC Tokai
facility. The International Atomic Energy Agency (IAEA) also participated in the programme, which was called by the acronym TASTEX for Tokai Advanced Safeguards Technology Exercise.

Among the thirteen individual tasks (designated A to M) of the TASTEX programme Task F investigated the feasibility of applying to the PNC Tokai Plant what has now come to be called near-real-time (n.r.t.) material accountancy. Task F was carried out as a joint effort involving several staff members of the IAEA, Los Alamos National Laboratory (LANL), PNC, and the Japan Atomic Energy Research Institute (JAERI). As the study progressed, the summary results were presented at meetings of ESARDA, the American Nuclear Society, and the Institute of Nuclear Materials Management (INMM) [1–3], and also published [4,5]. After the TASTEX programme closed, the investigation of the applicability of the n.r.t. accountancy system was continued, giving some valuable results particularly with regard to operational experience of the field test and analyses of the source of measurement bias.

Since the International Working Group on Reprocessing Plant Safeguards (ING-RPS) recognized that the n.r.t. accountancy concept would have a significant potential for improving IAEA safeguards, there has been considerable worldwide interest in it. A Consultants' Working Group Meeting, held by the IAEA in January 1982, on the current technical status of n.r.t. material accountancy recommended that there should be more field exercises and demonstrations, and in the light of this PNC and JAERI have continued development work in this field. The results obtained up to now are described in this paper. Since the n.r.t. accountancy model for the Tokai Plant and effectiveness evaluations of it have been described in some detail in the literature quoted, only an outline will be given here (Sections 2 and 3).

2. THE N.R.T. ACCOUNTANCY MODEL (TEN-DAY-DETECTION-TIME MODEL)

2.1. Basic concept

The basic concept of the PNC n.r.t. accountancy model is as follows:

(a) Weekly in-process inventories are taken of plutonium in the process MBA.
(b) Both flow and inventory quantities are measured either by conventional chemical methods or by instrumental (NDA) methods.
(c) All measurements are completed, and the resulting MUF is determined and evaluated, within a period of about two or three days after the in-process inventory.

(d) Evaluation of MUF data would be based on statistical techniques which utilize sequences of short-term material balances.

The n.r.t. material balance equation is solved weekly using a dynamic estimate of the in-process physical inventory. Some or all of the data used in the n.r.t. material balance are determined using procedures designed to give rapid but not necessarily highly precise results. The in-process physical inventory includes all significant inventory quantities, but several minor process holdup quantities are knowingly omitted. Thus during initial startup operations MUF includes this unmeasured in-process inventory. Under steady-state conditions, only the fluctuations in this in-process inventory appear in MUF.

Analysis of material accounting data for possible loss or diversion is one of the major functions of the n.r.t. accountancy system. Decision analysis, which combined techniques from estimation theory and system analysis, is well suited for the statistical treatment of the imperfect n.r.t. material balance data that become available sequentially in time.

2.2. Measurement for n.r.t. accountancy

In the PNC Tokai facility some 13 transfer measurement points have been identified for the purposes of conventional material accountancy. For n.r.t. accountancy, six additional inventory measurement points have been defined. These points all relate to buffer storage tanks feeding the first, second and third extraction cycles and the plutonium product evaporator.

2.2.1. Measurement of input quantities

The reference model facility will generate approximately 13 input dissolver batches during the course of one week's normal operation, each containing perhaps 3 kg plutonium. Since some of these batches will be generated several days before the analytical data are required, it can be assumed that the normal high-quality input measurement data will be available for these batches. It is, however, assumed that the last two or three batches will have to be analysed by some other method, e.g. one which can give a rapid provisional value. Since it should be possible to complete high-quality
measurements on all input dissolution samples within one additional week, the best procedure is to treat MUF for the latest material balance period as a provisional MUF, to be corrected as soon as all high-quality measurements are completed.

2.2.2 Measurement of output quantities

The model facility will generate approximately five batches of plutonium product solution during the course of one week's normal operation, each containing about 8 kg Pu. It is assumed that within the three days allowed for the completion of measurements and the statistical evaluation of n.r.t. material balance data, all these batches can be analysed by the normal high-precision methods. (Because of this period of one week plus 3 days, the model is called the ten-day-detection-time model). On occasion it may be necessary to use some provisional method for the last batch sampled, or possibly to delay the evaluation of the material balance data for the time required to complete the last analysis.

2.2.3 Measurement of in-process inventories

Timing and amount of in-process inventories

No means was identified for inventorying plutonium in the product evaporator, so the specification was adopted that in-process physical inventories would always be taken when the evaporator was empty and before restarting evaporation of the next batch. The model design flow sheet calls for discharging the evaporator once every 24 hours.

If the plutonium product evaporator is empty, the estimated total plutonium inventory in the process material balance area (MBA) is about 17 kg, with 13 kg of that amount being in the four identified buffer storage tanks. Essentially all of the remainder (4 kg) was estimated to be in the solvent extraction system.

Measurement methods

Since in-process inventories are to be taken at weekly intervals, and since the IAEA guideline for detection of possible abrupt diversions calls for detection within one to three weeks, the specifications for measurement methods neither require nor forbid the use of instrumental or NDA methods. Since each of the four buffer storage tanks is normally sampled daily for process control purposes anyway, it was convenient in this study to assume that conventional measurement methods would be used.
2.3. **Statistical evaluation procedures**

Statistical evaluation procedures used in the study are essentially those developed at the LANL [6]. The complete computer package includes four statistical tests, a straightforward cumulative MUF (CUMUF) test, a uniform diversion test based on Kalman filter statistics, a variance test, and a two-directional test based on two Kalman filter models operating in opposite directions.

3. **EFFECTIVENESS EVALUATION BY SIMULATION STUDY**

3.1. **Simulation model**

To evaluate the effectiveness of the n.r.t. accountancy system based on the ten-day-detection-time model, a computer simulation approach was used. A dynamic mathematical model of the Tokai Plant, DYSAS-R (Dynamic Safeguards Simulation Code for Reprocessing Facility), was developed. The model covers almost all major processes in the plant. The simulated data can be obtained for process material flows under normal operating conditions including plant start-up, flush-out and clean-out operations. Several additional computer codes were developed to perform the evaluation study.

3.2. **Capability of the proposed n.r.t. accountancy system**

Various simulation runs were performed to obtain quantitative information of the detection capability of the proposed n.r.t. accountancy system. Each of these cases is characterized by a combination of the following: (a) plant operation modes, (b) fuel types, (c) measurement methods and their accuracies, (d) recalibration frequencies, (e) n.r.t. material balance periods, (f) portions to which an in-process physical inventory is applied, and (g) loss or diversion modes. The major results of the simulation run are summarized as follows:

(a) Measurement errors associated with the in-process inventory have little influence on the estimation accuracy of MUF, since the quantity of in-process inventory is much smaller than the total throughput in a single n.r.t. material balance period, namely 7 days.

(b) The conventional material accountancy with a six-month-material-balance period will give LEMUf \((= 2\sigma_{MUF})\) values of 7 to 15 kg Pu. This suggests that the detection of a diversion of 8 kg Pu may sometimes be difficult.
The n.r.t. accountancy system based on the ten-day-detection-time model has the possibility of detecting a single or a series of random diversions of 8 kg Pu. It also has the possibility of detecting the protracted diversion of 8 kg before the total amount diverted exceeds 8 kg.

4. PRELIMINARY FIELD TEST OF THE PROPOSED N.R.T. ACCOUNTANCY SYSTEM

4.1 General

The field test has been carried out in the PNC Tokai Plant since April 1980, although there was an interruption in 1981. Up to the end of August 1982, there were five campaigns, identified as the C-1 and C-2 in 1980, the 81-1 and 81-2 in 1981, and the 82-1 in 1982. In the 81-1 campaign the data collection for the field test was intermitted for about six months because of operational restrictions. Up to the end of the 81-2 campaign, the field test data were collected by PNC and analysed by JAERI, but starting from the 82-1 campaign PNC have analysed their collected data by themselves using a computer software system which JAERI transferred to PNC. The present paper describes experience gained in collecting and evaluating the field test data during the first three campaigns, C-1, C-2 and 81-2. (Campaign 81-1 is ignored here for the reason given above). Because of the interruption between the C-2 and the 81-2 campaigns, the statistical analyses of MUF data were made for two data groups, one consisting of the C-1 and C-2 campaigns together and the other the 81-2 campaign.

4.2 Data collection

4.2.1 Volume measurement data

In the Tokai Plant, pneumatic bubble systems using dip tubes are generally used for the plant instrumentation. These systems are provided for all the vessels required for the in-process inventory. Therefore, volume measurements of these vessels can easily be made by reading the level indications of recorders corresponding to these vessels. These recorders are provided on a control panel at the main control room, and have 0.5% precision for a visual reading.

On the other hand, in the case of input and plutonium product accountability tanks, measurements are made using a water manometer system of higher precision.
4.2.2 Sampling for chemical analyses

At the time of an in-process inventory samples must be taken in a sufficiently short period of time, during which volume and concentration of solutions in the vessels should not vary significantly. Such a period of time must be found in the period of about one hour between emptying the evaporator and restarting evaporation. Nevertheless it took about two hours to complete sample takings at these vessels in the actual field test, because of the limited manpower available for the test and restrictions imposed on the operation of sampling equipment. This long period for sample taking may have had some, but not significant, influence on the accuracies of in-process inventory measurements. This problem could probably be resolved if sufficient manpower became available for performing the field test.

4.2.3 Chemical analysis

It is observed in the actual operation that highly active samples from the input accountability tank had so accumulated during a long campaign that the timeliness of analysis for plutonium concentration could no longer be attained. This is mainly because of the time-consuming processes used to remove fission products from those samples, and the specification imposed on the chemical analysis work, i.e. the work is allowed to be performed only in the daytime in order to maintain high-quality analysis.

To cope with this difficulty, spectrophotometric analysis was applied to all those samples when the resulting data are used for the n.r.t. accountancy purposes. The expected accuracy by using this method is 3% (1σ); the data produced were replaced by those obtained by the more accurate method usually used as soon as they became available.

4.2.4 Evaluation of the results of data collection

Time intervals between succeeding in-process inventories could not be kept constant, but were expanded to 11 days in steady operating conditions and to more than 20 days in unsteady operating conditions. More flexible timing of the in-process inventories should be considered, but too much flexibility could adversely affect the proper operation of an n.r.t. material accountancy system.

Product recycle and dilution tanks should be inventoried for n.r.t. accountancy purposes, because in the authors' tests they sometimes contained a significant quantity of plutonium. These tanks have been inventoried since the 81-2 campaign.
No additional unforeseen problems arose during the collection of the demonstration data. The time needed for an in-process inventory, exclusive of the time required to analyse samples, was less than half an hour. The time needed for analysing samples was short enough to satisfy the requirement of n.r.t. accountancy. The question of additional manpower needed for the plant operator to pursue in-process inventory was examined and the number of man-days required was found to be very small, e.g. increase in manpower needed for sample analysis was few percent in a campaign.

4.3 Evaluation of n.r.t. material balance data

4.3.1 Evaluation of MUF data of the C-1 and C-2 campaigns

The n.r.t. material balance data obtained during the C-1 and C-2 campaigns are adjusted to include an estimated 4000 g unmeasured inventory. Figure 1 shows a CUMUF chart, for which cumulative summations of adjusted MUFs are plotted sequentially with 1σ error bars. Alarm charts associated with the CUMUF and Shewhart charts indicated many alarms of significant levels. However, there has been no indication that a significant quantity of plutonium, e.g. greater than 1 kg, disappeared in the process MBA before the C-1 campaign, such that it could reappear in the process MBA during the C-1 and C-2 campaigns. Therefore, there are two logical alternatives. Either

(a) $\delta$MUF had been under-estimated, because measurement errors assigned for the Key Measurement Points (KMPs) were declared too small, or

(b) There had been a significant measurement bias across the process MBA, resulting in a net gain of plutonium.

The nature of this bias was not clearly shown in conventional material accountancy data. Figure 1 gives a dramatic picture of the effectiveness of the n.r.t. accountancy concept, and of the ten-day-detection-time model, in the PNC Tokai Plant, if adopted.

4.3.2 Bias estimation and correction

A simple way to estimate a bias is to use the last CUMUF value, which is approximately equal to the MUF value obtained by the clean-out physical inventory taken at the end of the C-2 campaign. The bias is given by the CUMUF value divided by the total number of n.r.t. material balance periods. The estimated value is equal to 510 g per material balance period, or 2% of the average throughput in an n.r.t. material balance period.
Figure 1. CUMUF chart for the C-1 and C-2 campaigns. Cumulative summation of MUF data, adjusted to include an estimated 4000 g Pu unmapped inventory, is plotted sequentially with 1σ error bars.

Figure 2. CUMUF chart based on MUF data corrected further for an estimated 510 g Pu bias.

Figure 2 is a new CUMUF chart, on which a cumulative summation of MUF data, which were corrected for the bias, is plotted. It shows that all CUMUF values except the 4th material balance number can be adequately interpreted in terms of measurement uncertainties with 5% of the significance level.
4.3.3 Possible source(s) of the bias

To concentrate the investigation effort to find out the source of the bias on a certain limited area, it is necessary to know the dominant component of the variance of the conventional MUF data ($\sigma_{\text{MUF}}$). This was calculated for the C-1 campaign. The result indicated that 99.8% of the total variance was due to random errors. The systematic errors were estimated to be 0.2% for the C-1 campaign and 0.4% for the C-2 campaign.

FIG. 3. Calculated $\sigma_{\text{MUF}}$ for the C-1 and C-2 campaigns. Systematic and random error components are varied parametrically.
FIG. 4. Shewhart (upper) and CUMUF charts for the 81-2 campaign. All MUF data are adjusted to include estimated plutonium unmeasured inventories taking rinsing effects into consideration.

Variance was derived from input and product measurements, and 55%, 18% and 27% of the total variance was derived from analyses, samplings, and volume measurements, respectively. The components of systematic and random errors were 92% and 8%, respectively. These estimates suggest that effort to reduce measurement errors should be concentrated on the systematic errors associated with analyses of input and/or product samples.

The next step is to see how much reduction should be required for input and product measurement errors, if an
explanation of the present large MUFs were required to be made by measurement errors alone. For this purpose, a parametric study of $^{13}$MUF (of the conventional material accountancy) was made, varying magnitudes of various error components to two times, three times, and six times the original ones. Combinations of these parameters and the result of calculations are illustrated in Fig. 3.

As shown in this figure, actual MUF data in the C-1 and C-2 campaigns could be explained by measurement errors alone in the case of No.9, No.10 or No.11. A common factor in these three cases is the assumed magnitude of the systematic error for the analysis of input samples. The magnitude is approximately 2.5%, which is six times larger than the original.

4.3.4 Evaluation of MUF data of the 81-2 campaign

The result of the field tests in the C-1 and C-2 campaigns triggered an operator's investigation on the bias problem. A special team was organized in the Tokai Reprocessing Plant to solve this problem. Two sources of bias were identified and corrected, but it is understood that on the basis of the available data these two biases should have accounted for only a part of the total observed bias. After these corrections the 81-2 campaign data were collected and found to be biased positively (Fig. 4). Change in the direction of the bias may be another problem, which suggests there may be another source of error that could not be accounted for by the two corrections identified in fact. The result of the investigation on the bias problem is reported elsewhere [7].

5. CONCLUSION

The study has shown that it might be feasible to apply n.r.t. material accountancy to the PNC Tokai Reprocessing Plant, that doing so could fulfil IAEA objectives in terms of detection timeliness and sensitivity, and that by such a system impacts on normal operations could be minimized. The next step in the development work of the n.r.t. accountancy system in the PNC Tokai Reprocessing Plant is to make a full demonstration of the system including Agency verification activities. It is planned to start this work as one of the projects under the Japan Support Programme for Agency Safeguards (JASPAS).
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INTERNATIONAL WORKSHOP ON THE NEAR-REAL-TIME ACCOUNTANCY MEASURE
Overview report

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Abstract

INTERNATIONAL WORKSHOP ON THE NEAR-REAL-TIME ACCOUNTANCY MEASURE: OVERVIEW REPORT.

An International Workshop on the near-real-time accountancy (NRTA) measure was established in December 1980 to investigate the capabilities and limitations of this measure for a large-scale reprocessing facility. The present overview report summarizes the activities and the results of this workshop as of July 1982. After establishing the process and accountancy data-base for a 1000 t HM/a reference reprocessing facility, the workshop developed simulation models for the sequential generation of data for throughput and inventory of plutonium in the process material balance area (MBA). A well defined set of boundary conditions and parameter values for measurement uncertainties and loss patterns was established, on the basis of which a number of sequential statistical test procedures were evaluated. One important condition for the application of the NRTA measure was the stipulation that routinely measured Pu inventories in process tanks only, would be used, since more than 95% of Pu inventories in the process MBA are in these tanks. About 12 kg of Pu, expected to be the normal inventory in six pulse columns, was assumed to be constant. In spite of the simplifications made and the fact that mainly simulated data were used, these investigations permit the conclusion that the NRTA measure provides a greater sensitivity in terms of the amounts which can be detected and the timeliness of detection, than the conventional material accountancy. Since measurements are restricted to process tanks only, routinely available measurement techniques can be used. The main thrust of R & D activities has to lie in the practical demonstrations of this measure under operating conditions some of which are already under way.

* The affiliations of the contributors are given in Annex 1. The names are listed as co-authors in the Author Index provided at the end of volume 2 of these Proceedings.

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An International Workshop on the near-real-time accountancy (NRTA) measure was established in December 1980 to investigate the capabilities and limitations of this measure for a large-scale reprocessing facility. The present overview report summarizes the activities and the results of this workshop as of July 1982.

NRTA has been considered by many as one of the promising measures which could extend the capabilities of present-day international safeguards. However, the fairly large volume of published information on this subject in the recent past may not always enable the reader to form an objective and unbiased opinion on the applicability of such a measure.

The International Workshop was specially established to develop a set of data base and guidelines with the help of which the capabilities and robustness of the NRTA measure can be investigated.

Organizations and internationally known experts from Japan, the USA and the Federal Republic of Germany, actively engaged in this area, were invited to provide advice and information as well as contribute actively to the workshop. The overview report is prepared on the basis of these contributions.

The contributing organizations and experts actively taking part in the workshop meetings and activities are listed in the Annex.

The International Workshop has covered up to now the following areas:

(1) Generation of throughput and inventory data for Pu for a reference layout of a reprocessing facility with a capacity of 1000 t HM/a.

(2) Measurement and other estimating systems for the generation of material balance data for the reference facility.

(3) Simulation of relevant process characteristics of the reference facility (in particular the behaviour of in-process Pu inventory under normal and unstable process conditions).

(4) Establishment of boundary conditions for comparison of different statistical procedures for the evaluation of
the material balance data with regard to different loss patterns including a possible diversion.

(5) Selection of relevant sequential statistical test procedures.

(6) Preliminary comparison of different statistical test procedures to evaluate their capability.

(7) Identification of areas in which specific R & D activities would be required for demonstrating the possibility of implementing NRTA as an international safeguards measure on a routine basis.

In the remainder of this report these subjects are treated one by one.

1. THE REFERENCE REPROCESSING FACILITY

An extremely simplified block diagram of the reference reprocessing facility is presented in Fig. 1, with some of the
TABLE I. CHARACTERISTICS OF THE 1000 t HM/YEAR REFERENCE REPROCESSING FACILITY

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<tr>
<td><strong>Throughput</strong></td>
<td><strong>1000 t HM/a = 5 t HM/d</strong></td>
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<tr>
<td></td>
<td><strong>10 t Pu/a = 50 kg Pu/d</strong></td>
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</tr>
<tr>
<td><strong>Pu inventory (kg)</strong></td>
<td><strong>525</strong></td>
<td>(feed adjustment - product accounting; about 25 process buffer tanks with significant quantities of Pu).</td>
</tr>
<tr>
<td><strong>Process tanks</strong></td>
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<tr>
<td><strong>Pulse columns</strong></td>
<td><strong>12</strong></td>
<td></td>
</tr>
<tr>
<td><strong>Mixer settler and oxidation equipment</strong></td>
<td><strong>6</strong></td>
<td></td>
</tr>
<tr>
<td><strong>Evaporator</strong></td>
<td><strong>12</strong> 30</td>
<td>555</td>
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information relevant to NRTA on throughput and inventory of plutonium summarized in Table I [1]. Table II presents data on plutonium inventories in various process tanks.

According to this layout about 95% of the process inventory of Pu is expected to be in about 25 process buffer tanks under normal operating conditions. This has been found to have important simplifying consequences for the NRTA measure. Since the main measurement efforts for assaying Pu process inventory are restricted to these tanks, most of the measurement data can be generated with well-tested measurement methods already existing for operating purposes for Pu assay in accountability tanks in reprocessing facilities. In case the Pu inventories in pulse columns become a relevant problem, simplified simulation methods for estimating these inventories could be used.

2. MEASUREMENT SYSTEMS FOR THE GENERATION OF MATERIAL BALANCE DATA [2,3]

In establishing the requirements and characteristics of measurement systems to be considered for the generation of the required material balance data for the NRTA measure, a number of boundary conditions were stipulated:
The plutonium amounts in inventories in and in flows to and from the process material balance area (MBA) will be determined, as far as practicable, on the basis of measurements carried out in connection with the normal operation of the facility.

As the first alternative, only the process tanks (and not process equipment or pipelines) will be assayed to establish the plutonium inventory in the process. Such inventories will be established during the operation of the process (physical inventory taking without plant shut-down).

The plutonium inventories in process equipments and pipelines (which correspond to approximately 6-7% of the total plutonium inventory in the process) will be considered to be approximately constant or fluctuating within a given range (e.g. ± 10%).

In most of the cases, the throughput and inventory of Pu will be assayed in tanks. The following steps may
be required for a measurement system for determining the Pu amount in such a tank:

- Homogenization of the solution
- Measurement of the respective tank volume
- Sampling from the tank
- Sample transport to the analytical measurement system
- Sample conditioning
- Measurement of the plutonium concentration
- Calculation of the plutonium inventory in the tank.

(NOTE: The main investigation was directed to the measurement systems involved in the determination of plutonium concentration, although the step involving sample conditioning could be the decisive one for establishing the suitability of a system. This fact was taken into account in the total time required for a given measurement system).

- The total measurement time including all the steps should be, whenever possible, less than the residence time of a Pu solution in a tank (e.g. in the range of 8-24 hours). In this manner the frequency of the NRTA will be determined by the inherent process parameter involving the residence time of the Pu-containing solution in a tank and not by the delays in the measurement system. In addition, the verification of the operator's measurement data could be carried out in principle during the time the solution is still in the tank.

The plutonium flows and characteristics of the process materials relevant for NRTA, are presented in Table III.

It is to be noted that:

- Both the Pu concentrations and the radioactivity in the different process streams undergo variations by several orders of magnitude.

- The sample material in the different process streams in the facility can be broken down into four categories:

  - The input solution
  - The end product
  - The liquid waste
  - The solid waste.
TABLE III. PLUTONIUM FLOWS, CONCENTRATIONS AND ACTIVITIES IN DIFFERENT PROCESS STREAMS OF THE REFERENCE FACILITY

<table>
<thead>
<tr>
<th>Process stream</th>
<th>Plutonium flow (kg Pu/a)</th>
<th>Material description (Approximate plutonium concentration; activity)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Input tank</td>
<td>10000</td>
<td>1-2 mg/ml ($\alpha,\beta,\gamma$)</td>
</tr>
<tr>
<td>End product</td>
<td>10000</td>
<td>250 mg/ml ($\alpha$)</td>
</tr>
<tr>
<td>Centrifuge waste</td>
<td>20</td>
<td>50 g/g ($\alpha,\beta,\gamma$)</td>
</tr>
<tr>
<td>Leached hulls</td>
<td>10</td>
<td>1 g/g ($\alpha,\beta,\gamma$)</td>
</tr>
<tr>
<td>High active waste (HAW)</td>
<td>40</td>
<td>1 g/g ($\alpha,\beta,\gamma$)</td>
</tr>
<tr>
<td>Medium level and low</td>
<td>3</td>
<td>1 g/g ($\alpha,\beta,\gamma$)</td>
</tr>
<tr>
<td>active wastes (MAW and</td>
<td></td>
<td></td>
</tr>
<tr>
<td>LAW)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

It has been assumed that for the first three categories of solution chemical analytical methods would be routinely used whereas for the fourth category (solid wastes) non-destructive measurement methods would have to be applied.

Different measurement methods were investigated in the context of their application possibilities, for the input product and waste streams.

After taking into consideration the different characteristics of the possible measurement systems, a set of values for measurement errors ($\sigma$) were established for carrying out the required sensitivity study for the NRTA measure. These measurement errors which are somewhat on the conservative side are presented in Table IV. For the purpose of this study the values for the random ($\sigma_R$) and the systematic ($\sigma_S$) errors have been assumed to be the same for the reference case.

The unmeasured inventory corresponding to an average of 30 kg of plutonium in the different process equipment and pipelines, is assumed either to remain constant or to fluctuate by $\pm10\%$ around the average for the different case studies.
TABLE IV. MEASUREMENT ERRORS FOR THE DIFFERENT FLOWS AND INVENTORIES IN THE REFERENCE FACILITY ($\sigma_R = \sigma_S$)

<table>
<thead>
<tr>
<th>Object of measurement</th>
<th>Measurement errors (1) (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Input accountability tanks</td>
<td>1.0</td>
</tr>
<tr>
<td>Product output</td>
<td>0.3</td>
</tr>
<tr>
<td>Centrifuge wastes</td>
<td>25</td>
</tr>
<tr>
<td>Leached hulls</td>
<td>50</td>
</tr>
<tr>
<td>HAW</td>
<td>25</td>
</tr>
<tr>
<td>MAW, LAW</td>
<td>25</td>
</tr>
<tr>
<td>Inventory in process tanks</td>
<td>1.0</td>
</tr>
</tbody>
</table>

3. SIMULATION OF RELEVANT PLUTONIUM FLOW AND INVENTORY CHARACTERISTICS [4 - 6]

Model simulation has been considered to play an essential role in evaluating the capabilities of the NRTA measure. This is of two types.

The first type deals with the mathematical modelling of the relevant plutonium extraction and purification steps in the reference facility and simulation of the distribution and flows of plutonium in these steps. The input data for such simulation studies are obtained from the data base for the process layout of the reference facility. This type of simulation is expected to indicate the behaviour of the flow and inventory of plutonium in the process MBA under start-up, normal and some abnormal operating conditions. This process model is then used for the simulation of accountancy data for establishing MUF values and their associated uncertainties, for different values of measurement uncertainties, fluctuations in the inventories and for different throughputs and inventories. The accountancy data thus generated in the simulation model are then used for applying different statistical test procedures for generating safeguards-relevant
conclusions with regard to the status of the plutonium in the process area. The capabilities of such test procedures can also be evaluated.

This type of simulation model is necessary for analysing the capability and limitations of the NRTA measure under different facility conditions. How much confidence one can place in the results of such models depends on how well they can simulate the actual conditions prevailing in the facility. It is, therefore, important that data based on actual operating conditions in a facility be used in such models. The initial results from such a simulation model are presented elsewhere in these Proceedings [4]. It is to be noted that these models are not required for routine implementation purposes of the NRTA measure.

The other type of simulation model, which can be a subset of the first type, deals with the estimation of plutonium inventory in process equipment such as pulse columns. Under the present series of investigations, the measured data base are generated mainly from measurements in process tanks. The unmeasured inventories in the process equipment are assumed to remain constant or vary within a limited range of ± 10%. If this inventory is found to vary much more under routine operating conditions than the assumed rates, the sensitivity of the NRTA measure would automatically go down. For such cases (which might be remote), some possibility of obtaining a value for the plutonium quantity in this equipment on the basis of some derived estimates would be useful. Such models are expected to be as simple as practicable and to be based on a few measurable and verifiable data. If required, they would then form part of the measurement systems for generating the accountancy data on a routine basis for the NRTA measure.

Up to now three different types of model for estimating the plutonium inventories in the pulse columns of the reference facility have been developed.

3.1 Exponential model

The exponential model for predicting holdup of special nuclear material in pulse columns provides a simplified method for approximating the in-column inventories. In this model it is assumed that the plutonium concentration profiles in the extraction section of the column vary exponentially. The model is intended for steady-state operation only and does not take into consideration the many complex variables that affect column performance and holdup.
3.2 Ideal stage model

In the ideal stage model, the holdup of plutonium in a pulse column is estimated by summing up the plutonium amounts per stage (i.e. average plutonium concentration x liquid volume per stage). The holdup of plutonium in this model is ultimately a function of the plutonium concentration and flow rates of the aqueous feed and the organic extractant at the input and output of the column, the separation coefficient for extraction, the number of ideal stages and the total volume of the column. In this model also, complex effects such as back mixing, temperature and molarity dependence of distribution coefficients, etc., are neglected.

3.3 Reduced-order linear model

The reduced-order linear model is a linear inventory estimator based on first-order perturbations about an expected steady-state value. The steady-state inventory value is calculated for the expected operational conditions using a detailed chemical model that has been validated experimentally for the particular contactor system. Alternatively, experiments can be performed directly to determine the expected inventory by bringing the contactor to steady state and then draining the contents to holding tanks for measurement.

The column inventory calculations are based on the following assumptions:

- The column is operating near a steady-state operating point.

- The column inventory near the operating point is linear in the concentrations.

- Concentration and flow-rate measurements are available in near-real time.

- The column inventory at the nominal operating point has been previously determined from chemical model calculations and calibration experiments.

A realistic simulation of solvent extraction columns at least near-equilibrium operation is essential if the respective models are to be of use for in-process holdup estimation or as elements of an overall process simulator.
In the first stage of the investigations, these three models were tested on the basis of the flow sheet data for pulse columns for the reference facility. The exponential model was also tested for the pulse columns of the Barnwell facility, for which experimental data for a uranium stream are available [7]. The responses (for plutonium inventory changes) of these models to 10% increases in feed and extractant flow rates and concentrations, show that the holdup variations are roughly proportional to the changes in input flows and concentrations. Further investigations with realistic operational data are required to demonstrate the usefulness of these models.

4. BOUNDARY CONDITIONS AND PARAMETERS FOR COMPARISON AND EVALUATION OF DIFFERENTIAL STATISTICAL TEST PROCEDURES

During the workshop activities a uniform set of boundary conditions and parameters was established for the comparison and evaluation of different statistical test procedures. One of the difficulties in the past in assessing the capability of the NRTH measure using different test procedures, had been the lack of a uniform set of such conditions for comparison.

From a large number of individual conditions [8 - 11] the more relevant ones are summarized as follows:

4.1 Boundary conditions

4.1.1 The evaluation will consider a series of determinations of the inventories and transfers corresponding to a single MBA. Inventories will be assumed to be taken at times \( t_i \) (without stopping the operation of the facility) starting at some initial time \( t_0 \) and continuing indefinitely. It will be possible to consider:

a) All the data
b) All the data subsequent to \( t_0 \)
c) All the data obtained during some fixed period \( T \).

4.1.2 The data will be characterized by

a) Assumed true values \( \mu_i \) of the inventories at time \( t_i \). In simulation studies these are derived from the process simulation.

b) Assumed errors of determination of net transfers and inventory. These are propagated from an
assumed set of measurement errors, accounting procedures and covariance structure.

c) Assumed losses $L_i$ in the periods $\Delta t = t_i - t_{i-1}$ between inventories. These may be present in known or unmodelled process errors, long-term or unmodelled measurement biases, or deliberate diversion.

4.1.3 In this exercise the primary concern is to estimate and test hypotheses concerning the $L_i$. The $\mu_i$ are modelled since (1) assumptions concerning these may affect the ability to estimate the $L_i$ and (2) in more general situations it may be desirable to estimate or test inventories as well as losses.

4.1.4. Characteristics of loss patterns are to be considered.

a) The evaluation is concerned with the total loss $M$ during $T$ which may be, for example, a calendar year or a campaign. The individual loss $L_i = \gamma_i M$ within this period are of concern only as the patterns $\gamma_i$ affect the ability to estimate and test $M$.

b) Within $T$, to take account of the nature or timeliness of losses, the time period of the initial loss, the time at which the amount $M$ was assumed to be available and the form of diversion (namely abrupt or protracted) will have to be considered.

c) Because of externally imposed conditions (legal, administrative) the evaluation procedures has to be concerned with the losses within $T$ from $\tau_1$ to $\tau_2$ following $\tau_0$. Several sub-cases need to be considered

- The existence of a base set of data known to be loss-free.
- Only the cases $L_i \geq 0$ will be considered.
- there exists a time $\tau_3$ by which detection of the losses between $\tau_1$ and $\tau_2$ should be achieved.
4.2 Parameters considered for the evaluation of statistical methods

4.2.1. Definitions

The following errors are treated in this section:

\[ \sigma_\eta = \text{Random error standard deviation in measuring inventory} \]
\[ \sigma_\varepsilon = \text{Random error standard deviation in measuring net transfers or flows} \]
\[ \sigma_\delta = \text{Systematic error standard deviation in measuring net transfers.} \]

Two loss patterns were defined. In loss pattern No. 1 the loss per time interval is uniform, beginning at interval \( I_0 \) (one parameter) and extending over \( m \) intervals (a second parameter). In loss pattern No. 2, \( I_0 \) and \( m \) are defined as in loss pattern 1 except that the loss pattern is not uniform over these \( m \) intervals; it alternates by \( \pm 50\% \) about the central value, \( M/m \).

The parameter values were defined relative to the value for \( \sigma_\eta \), which is fixed at one unit throughout. Also, \( \sigma_\delta \) is expressed relative to \( \sigma_\varepsilon \), i.e. \( \sigma_\delta /\sigma_\varepsilon \) is one of the factors varied.

4.2.2. Parameter values

In the first phase of the investigations, the following sets of parameter values were selected for uniform loss patterns (for both abrupt and protracted). The alternative loss patterns (loss pattern No. 2) have not been investigated in detail:

\[ \sigma_\varepsilon = 0.1, 0.55 \]
\[ \sigma_\delta /\sigma_\varepsilon = 2.5 \]
\[ M = 15, 25 \]
\[ I_0 = 1, 11, 21 \]
\[ m = 5, 10 \]

The cases investigated for the different statistical test procedures were 24 in number, consisting of all the possible combinations of the above-mentioned parameter sets. In addition, for \( I_0 = 11, m = 5, \) and \( \sigma_\varepsilon = 0.1 \) and \( 0.55 \), two cases were run at \( M = 0 \) to determine the values of the false alarm probability, \( \alpha \). The case number identification is given in Table V.
5. DESCRIPTION OF THE STATISTICAL TEST PROCEDURES INVESTIGATED

The realized values for accountancy data were generated with the process simulation model for the different parameter sets identified under section 4.2, with the assumed uniform loss patterns for different values of $M$. Under the present series of investigations these loss patterns were supposed to be detected with the help of different statistical hypothesis testing procedures. The probability of detection $P_D$ was taken to be the indicator for the sensitivity of a test procedure for a given set of accountancy data parameters, loss pattern and the value for $\alpha$.

Under this approach the observed material accounting data generated sequentially are applied to test the hypothesis $H_0$ of no material loss against the alternative hypothesis $H_1$ of material loss. Such tests are of two types: the fixed length test in which a predetermined number $N$ of balances are observed before deciding between $H_0$ and $H_1$, and the sequential test in which the possibility of a decision is allowed after each balance is observed.

For the purpose of checking the sensitivity, eight basic statistical test procedures were selected from those normally considered for NRTA measures. A short description of these tests follows [11].

<table>
<thead>
<tr>
<th>Case</th>
<th>$\sigma_\epsilon$</th>
<th>M</th>
<th>$I_0$</th>
<th>m</th>
<th>Case</th>
<th>$\sigma_\epsilon$</th>
<th>M</th>
<th>$I_0$</th>
<th>m</th>
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<tbody>
<tr>
<td>1</td>
<td>.1</td>
<td>15</td>
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<td>15</td>
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<td>6</td>
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<td>10</td>
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<td>25</td>
<td>1</td>
<td>10</td>
</tr>
<tr>
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<td>25</td>
<td>1</td>
<td>10</td>
<td>21</td>
<td>.55</td>
<td>25</td>
<td>11</td>
<td>5</td>
</tr>
<tr>
<td>9</td>
<td>.1</td>
<td>25</td>
<td>11</td>
<td>5</td>
<td>22</td>
<td>.55</td>
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<td>11</td>
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<tr>
<td>13</td>
<td>.55</td>
<td>15</td>
<td>1</td>
<td>5</td>
<td>26</td>
<td>.55</td>
<td>0</td>
<td>11</td>
<td>5</td>
</tr>
</tbody>
</table>
5.1 MUF

The MUF test is a test on the material balance for a given period. Letting \( D_i \) be the observed MUF for period \( i \), loss detection is said to occur if \( D_i \) exceeds some critical value determined by the value of \( a \) and the values of the measurement error standard deviations. The MUF test does not take into account any prior history. It is aimed at detecting an abrupt loss, one that occurs somewhere within the material balance period in question. As a test sequence, the MUF test is applied at each material balance period and loss detection over the \( P \) periods occurs if at least one MUF exceeds its critical value. The \( a \) value over all \( P \) tests is controlled by reducing the size of the significance level for each individual test.

5.2 CUMUF

The test statistic to be applied in period \( i \) is denoted by \( T_i \) and is the sum of the individual observed MUFs beginning at some point in time and extending through period \( i \):

\[
T_i = \sum_{j=1}^{i} D_i
\]

At a given point in time, \( T_i \) is independent of how the losses are distributed throughout the \( i \) periods. This is the cited advantage of the CUMUF test. As a test sequence, the CUMUF test is applied at each material balance period, as is the MUF test. Clearly, there is a close correlation between successive CUMUFs.

In this study, CUMUF is applied in sequence on the one hand, and only at the end of the 35 periods on the other. The single test in this latter instance is, of course, more powerful than is the 35th such test applied as the last test in the sequence. However, this increase in power is counterbalanced by the lack of timeliness, i.e., the inability of the test to detect losses that occur early in the sequence of time periods.

5.3 Uniform diversion (\( D_u \))

The test statistic is designed to detect uniform losses. Since uniform losses over a number of successive balance periods were the primary loss patterns studied in this phase, it would be expected that the uniform diversion test statistic would exhibit good detection capabilities in this study.
The linear statistic in question is the minimum variance unbiased estimate of uniform loss. Specifically, in this study, the statistic was defined for each group of four successive MUFs. It is a moving weighted average of four such MUFs, and it is clear that successive test statistics would be closely correlated.

The weighted average is derived as follows. Let

\[ T_i = a_1D_i + a_2D_{i+1} + a_3D_{i+2} + a_4D_{i+3} \]

where the \( a_j \)'s sum to 1 for \( j=1, 2, 3, 4 \). The \( a_j \)'s are chosen to minimize the variance of \( T_i \). The first test statistic is calculated at the end of the fourth balance period.

When \( j=1-4 \) as here, the calculation of the \( a_j \)'s is quite simple. For more complex cases, calculational algorithms are helpful. The oft-mentioned Kalman filter is a calculational algorithm used in this instance.

5.4 CUMUFR

CUMUFR is an acronym for cumulative sum of standardized MUF residuals. It is designed to detect changes in loss patterns. A uniform loss that occurs in all balance periods would not be detectable with the CUMUFR test.

The MUF residual for period \( i \), \( \text{MUF}_{Ri} \), is defined as

\[ \text{MUF}_{Ri} = D_i - E(D_1/D_1, D_2, ..., D_{i-1}) \]

where \( E(D_1/D_1, D_2, ..., D_{i-1}) \) is an appropriate linear function of \( D_1, D_2, ..., D_{i-1} \) chosen such that \( \text{MUF}_{Ri} \) has minimal variance. The standardized MUF residual is found by dividing \( \text{MUF}_{Ri} \) by its standard deviation \( \sigma_i \), and the CUMUFR test statistic for balance period \( k \) is found by summing \( \text{MUF}_{Ri} / \sigma_i \) from 1 to \( k \).

The time series of MUFs is a linear transformation of the time series of MUFs. They can be calculated exactly by applying this transformation or approximately through use of a Kalman filter.

The CUMUFR test may be applied as a two-sided test or as a one-sided test. In a two-sided test application, periods of losses followed by periods of no losses would also be detectable, whereas for a one-sided test, only periods of losses following periods of no losses would be detectable.
Note that in applying the CUMUFR test sequence, use is always made of all the MUF data extending back to period 1. This is a principal distinction in this study between CUMUFR and the $D_n$ test discussed next.

5.5 $D_n$

Like CUMUFR just discussed, $D_n$ is aimed at detecting changes in loss patterns. Unlike CUMUFR, in this study $n$ was fixed at 5, i.e. at the end of each balance period, and beginning with period 6, the current MUF is compared with some constant $\beta$ times the sum of the 5 previous MUFs where $\beta$ is chosen to minimize the variance. Specifically, the test statistic for period $i$ is

$$T_i = D_{i+5} - \beta \sum_{j=1}^{i+4} D_j$$

where $\beta$ is a simple function of the error variances in measuring net transfers and inventories.

In this study, the test was applied as a one-sided test. Thus, a period of losses followed by a period of non-losses would not be detectable.

5.6 Sequential probability ratio test

The sequential probability ratio test is related to the CUMUFR test in that the test statistic is the cumulative sum of the MUFs. However, the test is now a sequential test in the true sense of the word, as distinguished from a sequence of fixed length tests.

With a sequential test, when the value of the test statistic is calculated at the end of each period, the decision is made to either reject the hypothesis of no loss (i.e. declare that a loss has been detected), accept the hypothesis of no loss, or continue testing. When the hypothesis of no loss is accepted, then the test is restarted, and all prior data are ignored. This restarting of the test and deletion of prior data is what distinguishes the sequential probability ratio test from the CUMUFR test described earlier. With the CUMUFR test, the MUF data extending backward to period 1 are always retained.
5.7 Modified pages test

The modified pages test is also a sequential test in that the test may be restarted with all prior data eliminated when the accumulated evidence indicates that there has been no loss of material.

For the modified pages test, the test statistic is

\[ T_i = C_i - \min_j C_j \]

where \( C_j = \sum_{k=1}^{j} (D_k - \delta) \)

In effect, the test statistic is the current CUMUF minus the largest previous CUMUF, \( \delta \) being a constant.

The upper threshold (critical value) is a function of some parameter, \( A \), which controls the false alarm rate, and of the period number \( i \). The lower threshold is zero for the modified pages test.

5.8 Truncated sequential CUMUF

Like the sequential probability ratio test, the basic statistic is the cumulative sum of the MUFs. Also, the test is sequential in nature. This test procedure is called a truncated one because after a fixed number of material balance periods, a decision must be made as to whether or not a loss has occurred.

In evaluating this test procedure, a saddle-point solution is also found. The saddle point solution gives a guaranteed efficiency in the sense that it gives the detection probability corresponding to the least favourable loss pattern, i.e. it reacts to a diversion scenario in which the adversary chooses an optimum strategy.

6. TEST RESULTS

Eleven statistical test procedures were investigated including some variations and comparisons of the eight basic testing procedures described under section 5 for the 26 sets of parameter variations indicated in Table V. These eleven cases are identified in Table VI.
TABLE VI. IDENTIFICATION OF STATISTICAL TEST PROCEDURES INVESTIGATED

<table>
<thead>
<tr>
<th>Case identification</th>
<th>Case type</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>TS-1</td>
<td>Standard MUF test</td>
<td>Jaech [11]</td>
</tr>
<tr>
<td>TS-3</td>
<td>$D_u$ test</td>
<td>Jaech [11]</td>
</tr>
<tr>
<td>TS-4</td>
<td>$D_n$ test</td>
<td>Jaech [11]</td>
</tr>
<tr>
<td>TS-5</td>
<td>Truncated sequential CUMUF test</td>
<td>Beedgen [12]</td>
</tr>
<tr>
<td>TS-6</td>
<td>CUMUFR test; two-sided sequential test with power one</td>
<td>Sellinscheggl and Bicking [13]</td>
</tr>
<tr>
<td>TS-7</td>
<td>CUMUF; sequentially performed fixed length test</td>
<td>Sellinschegg [14]</td>
</tr>
<tr>
<td>TS-8</td>
<td>CUMUF (35); fixed length test at the end of 35 periods.</td>
<td>Sellinschegg [14]</td>
</tr>
<tr>
<td>TS-9</td>
<td>CUMUFR test; one-sided sequential test with power one</td>
<td>Sellinschegg [14]</td>
</tr>
<tr>
<td>TS-10</td>
<td>Sequential probability ratio test</td>
<td>Markin [15]</td>
</tr>
<tr>
<td>TS-11</td>
<td>Modified pages test</td>
<td>Markin [15]</td>
</tr>
</tbody>
</table>

6.1 Comparison of results

The results of investigations of the eleven statistical test procedures are summarized in Table VII. As mentioned earlier, the investigations in this phase were restricted to the uniform loss patterns.
TABLE VII. DETECTION PROBABILITIES ($P^*$) FOR PARAMETERS IDENTIFIED IN TABLE V AND STATISTICAL TEST PROCEDURES IDENTIFIED IN TABLE VI

<table>
<thead>
<tr>
<th>Case</th>
<th>TS-1</th>
<th>TS-2</th>
<th>TS-3</th>
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The results presented in Table VII illustrate some interesting points. Remembering that in this phase the simulation model has assumed that the inventories in the process columns would remain constant and that the systematic error components for the inventory measurements would cancel out (because of the fact that these data are generated as the difference of two measured values), the results are to be considered as indicating the highest sensitivity to be expected from the statistical test procedures, using the measurement data for the reference facility for the assumed uniform lost patterns.

a) The cases 25 and 26 give the actual \( a \)-values. When comparing detection probabilities, these differences should be kept in mind. Ideally they should be all about 0.05 for a fair comparison, but it is difficult in some cases to fix \( a \) precisely in advance. This would mean that for \( a = 0.05 \) the \( P_d \) values for the test procedures like TS-1, TS-3 and TS-7 could be higher than those obtained in the present cases.

b) It is to be noted that the test sequences TS-1 and TS-7 should give identical results with the TS-1 results calculated by the multivariate normal distribution and the TS-7 results by simulation. Taking into account the differences in the \( a \) values, the agreement is good.

c) The parameter cases 13-18 represent the worst cases considered since the uncertainties with throughput measurements are increased by a factor of about 5 for the same value of \( M = 15 \) compared to the other cases. There is in general a reduction in \( P_d \) values for almost all the test procedures excepting test TS-6 (cumulative sum of MUF residuals), in which the detection probabilities remain fairly high.

d) Test TS-6 in fact shows the highest probability values for all the 24 cases investigated in this phase.

6.2. CUMUF test

Since the particular test procedure TS-6 provided the highest set of probability of detection values investigated so far, this test was investigated in some more detail in the frame of the NRTA Workshop [13]. Using the values of Case 6 in Table VII as a basis, the results of these additional investigations are illustrated in Fig. 2, top diagram, taken from Ref. [13]. Converted to the data of the reference
FIG. 2. Detection probability for a total loss of 30 kg with different patterns [13].
facility, Case 6 would correspond approximately to the following absolute values:

\[
\begin{align*}
\sigma \eta &= 2.1 \text{ kg Pu} \\
\sigma \epsilon &= 0.21 \text{ kg Pu} \\
m &= 1 \text{ day} \\
I_0 &= 21 \text{ periods} \\
M &= 15 \text{ (equivalent to 30 kg of Pu)}
\end{align*}
\]

Loss patterns (Nos. refer to the Nos. in Fig. 2)

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Figure 2 illustrates a number of aspects in connection with the test procedures TS-6. They are mainly summarized from Ref. [13].

a) For abrupt diversion patterns (loss patterns 1, 2, 3) the \( P_D \) is above 95%.

b) For protracted diversion patterns (loss patterns 4, 5, 6, 7) the \( P_D \) increases with increasing balancing intervals, i.e. from \(~ 25\% \) to 78\% when the balancing intervals \( m \) are increased from 1 day to 10 days.

c) This sensitivity is obtained by assuming that the loss patterns start after 21 zero loss periods \( (I_0 = 21) \).

d) However, the main message which one gets from these illustrative examples as well as from those given under Table VII, is the fact that this type of NRTA measure brings about a significant improvement in the capability of material accountancy (in respect of detection probability for a given amount and detection time), over that for the conventional type of material accountancy. This remains valid in spite of the simplified assumptions made in this phase of the investigation. This basic fact is also illustrated through another in-depth work carried out by Ikawa [16] using the Burnwell reprocessing facility (BNFP) as the reference facility. One other important conclusion in that study is the suggestion that if the present chemical process were carried out in two parallel lines with half the processing capacity each,
the detection capability for protracted diversion might be significantly improved.

7. CONCLUSIONS

The NRTA Workshop has completed its activities involving simplified simulation models of the data for the reference reprocessing facility with 1000 t HM/a. In spite of the simplified assumptions and the fact that the capability of the NRTA measure could be investigated on the basis of mainly simulated data with little experimental validation, a number of generalized conclusions can be drawn. Some of these are similar to those drawn by a group of consultants at an IAEA consultants' meeting on this subject reported by Lovett et al. [17], and will not be repeated here.

The following conclusions, however, which include points from the conclusions of Ref. [17], are particularly relevant to the present study:

7.1. Measurement systems for the type of NRTA measure investigated in the workshop, i.e. sequential generation of the required material accountancy data sets, on the basis of Pu measurements in process tanks only (and not in process equipment), can be based on currently available technology.

7.2. Required statistical test methods for evaluating the material accountancy data in generating safeguards-relevant conclusions are available.

7.3. The sensitivity of the NRTA measure, in terms of amounts which can be detected with a given set of probability values and timeliness, is higher than that possible for the conventional type of material accountancy measure. For the type of NRTA measure investigated, this sensitivity will go down with high fluctuations of estimated Pu inventories in process equipment.

7.4. The main thrust of the R&D activities has to be in the direction of the validation of simulation models with actual operational data.

The in-depth activities on the bases of which the results presented in this overview report could be generated, were possible only through the support of the organizations involved and the excellent cooperation of the participants.
PARTICIPANTS AND AFFILIATIONS

International Workshop on the NRTA measure

Note: The names of those participants who contributed as co-authors to this report are listed under the title on the first page.

Abbreviations:

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REFERENCES


IN-PROCESS HOLD-UP AS A MEASURE OF SAFEGUARDS SIGNIFICANCE

A.G. HAMLIN
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United Kingdom Atomic Energy Authority,
Harwell, Oxfordshire,
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Presented by F.J. Walford

Abstract

IN-PROCESS HOLD-UP AS A MEASURE OF SAFEGUARDS SIGNIFICANCE.

This paper examines the use of the in-process hold-up itself, as a measure of safeguards significance. It is argued that for any process plant it is possible to define design limits for in-process hold-up, outside which the plant will not operate, or will operate in a detectably abnormal manner. It follows, therefore, that if the in-process hold-up can be derived at frequent intervals by input/output analysis from the start of the campaign, the only diversion that can be made from it during that campaign is limited to the quantity necessary to move the apparent in-process hold-up from its normal operating condition to the upper limiting condition. It also follows that detection of this diversion is as positive for protracted diversion as for abrupt diversion. If that part of the in-process inventory that is only measurable by input/output analysis has an upper operating limit that differs from its normal operating limit by less than a significant safeguards quantity of the material in question, the IAEA's criteria for both quantity and timeliness can be met by a combination of input/output analysis to determine in-process hold-up during the campaign, together with a material balance over the campaign. The paper examines the possibility of applying this measure to process plants in general, discusses means of minimizing the in-process inventory that must be determined by input/output analysis, and the performance required of the input and output analysis. It concludes that with current precision of measurement and with one input and one output batch per day, each measured, the method would be satisfactory for a campaign lasting nearly a year and involving 6 tonnes of plutonium. The paper examines the considerable advantages in verification that would arise from limiting safeguards analyses to the two points of input and output.

1. INTRODUCTION

It has been shown[1] that, for a process plant, reasonably frequent measurement of input and output batches would enable conventional nuclear materials accountancy to meet the criteria suggested by the IAEA for assessing safeguards performance up to fairly large throughputs.
This paper examines the use of the in-process hold-up itself, as a measure of safeguards significance. For simplicity the argument will be confined to process plants, but it should be noted that these plants contain no more than a few per cent of the material in the nuclear fuel cycle at any given time[2]. The arguments presented could apply equally to other facilities.

2. THE PROCESS PLANT PROBLEM

Nuclear material accountancy in a process plant at some time, t, after start-up is represented in Figure 1.

\[
\text{Input, } I_t \rightarrow \text{In-process hold-up } H_t \rightarrow \text{Output, } O_t \rightarrow \text{Measured Discards, } M_t
\]

**FIG.1. Accountancy of a simple process plant.**

If the plant was empty at the start, then at a time \( t \)

\[
H_{t_1} = \int_0^{t_1} I_t - \int_0^{t_1} O_t - \int_0^{t_1} M_t
\]

While the plant is operating, the safeguards uncertainty resides entirely in \( H_t \) since all errors in the factors on the right hand side of the equation are accumulated in \( H_t \).

Process plants are designed to operate with a specific in-process hold-up and it should be possible to specify an upper \( (H_U) \) and a lower \( (H_L) \) limit of hold-up outside which the plant becomes inoperable. It is thus possible to state that if

\[
H_U \geq H_t > H_L
\]

the plant is in control and there is no reason to suspect diversion.

At time \( t \) it might be feasible for the operator to move \( H_t \) in the direction of \( H_U \), thereby opening the possibility of diverting a quantity less than or equal to \( H_U - H_t \). (The effect of such diversion is to move the apparent value of \( H_t \) towards \( H_U \).)
If the integrals on the right hand side of the above equation are determined at frequent intervals, the effect of any such diversion will become apparent either as stepped movement of $H_t$ towards $H_U$ or as a gradual drift towards $H_U$ according to whether the removal of material is abrupt or protracted. Provided these movements can be detected by the measurement system, diversion is limited to a maximum of $H_U - H_t$ per campaign. There is no difference in detection ability for abrupt or protracted diversion over a campaign, nor, if it can be assumed that material diverted in a protracted manner is not useful until the end of its accumulation, is there a significant difference in timeliness of detection.

At the end of the campaign, $t$ reaches a constant value and the continuing subtraction of $t$ and $H_t$ reduces $H_t$ progressively to zero or to some residual minimum value, $H_{tr}$ operationally preferred to a complete clean out of the plant. Provided that $H_{tr}$ can be measured, the balance can be closed over the campaign. If diversion has occurred then the final value of $H_t$ obtained from equation (1) will not reduce to zero or to $H_{tr}$ but will be higher by the amount of the diversion.

The above argument suggests that the parameter of most safeguards interest in a process plant might be the in-process hold-up. If this shows significant excursions, then such excursions might be an effective signal for a drain down or clean out inventory to verify the accountancy of the process.

Possible objections to the above system are that:

(a) $H_U - H_t$ in large plants may exceed a significant quantity of nuclear material and therefore changes in the level of $H_t$ resulting from diversion, could be represented as normal operating fluctuations.

(b) The accountancy system may not be sensitive enough to monitor the movement of $H_t$ within $H_U$ and $H_L$ with a large enough degree of certainty.

These objections are examined in detail below.

2.1 The effective limits of in-process hold-up

In-process hold-up may include a selection from:

1. Material actually undergoing process.

2. Material held in stores or tanks between stages of the process for balancing operations.
3. Material deposited in process items which may be recovered by deliberate action or by random release.

4. Material recycled as a normal part of this process. (May be ~30% of throughput).

5. Rejected material waiting recovery for re-use in this or another process.

There is no predictable relationship between process time, throughput, and in-process hold-up because the quantities under (2) and (5) may be large compared with the material in process, and the material under (3) will be unpredictable and, although normally small, can be significant. The only items that are strictly related to the design flow sheet are (1) and (4) and even (4) may be subject to appreciable day to day variations.

This situation can be greatly simplified if significant inventories under (2) and (5) are accounted for separately from in-process hold-up, and if account is kept of the material returned for recycle (Fig. 2).

In this case, again starting from an empty plant at time zero

$$H_{t_1} = \int_{0}^{t_1} I_t - \int_{0}^{t_1} O_t - \int_{0}^{t_1} T_t - \int_{0}^{t_1} M_t - C_{t_1} - S_{t_1} - R_{t_1}$$  \hspace{1cm} (2)

Although the terms $C_{t_1}$, $S_{t_1}$, $R_{t_1}$, are the difference of the integrals over the time period of their inputs and outputs,
the only measurement required at the time that $H_t$ is determined is the inventory they represent. In many cases this can be checked quite quickly and accurately. Often there is no specific recycle store, the recycled material returning directly to the process so that $C_t$ is zero, the effect of recycle being to increase $H_t$ by a factor usually less than 1.3.

Although the process itself may be continuous, the materials involved are usually added or withdrawn in the form of batches. Thus equation (2) may be rewritten without integrals as

$$H_t = \sum_{i=0}^{n_i} I_j - \sum_{j=0}^{n_0} O_j - \sum_{j=0}^{n_t} T_j - \sum_{j=0}^{n_m} M_j - C_t - S_t - R_t$$

where $n_i$, $n_0$, $n_t$, $n_m$ are the number of batches $I_j$, $O_j$, $T_j$, $M_j$ of the various feeds and take-offs handled between start-up and time $t$. All the quantities in this equation are thus easily determinable, but because exact time correlation has been lost some uncertainty will be introduced into $H_t$. In practice this can be eliminated by defining the process area so that process stores containing the tail of an incompletely utilised input batch and portions of incompletely formed output, and discard batches are enclosed in the process area. However, if it is possible to measure these quantities when $H_t$ is determined they can with benefit be excluded from the in-process hold-up.

It may be possible to optimise the time of balance i.e. when no batches are in process of being added or withdrawn.

2.2 Application of Nuclear Materials Accountancy to the Determination of In-process Inventory

For simplicity, a process area with batched input and output only will be considered. The essential features of accountancy are not affected by the other real outputs included in equation (3). Their effect is to add some additional uncertainty. The relevant equation is:

$$H_t = \sum_{j=1}^{n_1} I_j - \sum_{j=1}^{n_0} O_j$$

from which the uncertainty in $H_t$ may be derived as:-
\[\sigma^2_{H_t} = \sum_{1}^{n_i} \sigma^2_{I_j} + \sum_{1}^{n_1} \sigma^2_{O_j} \quad (5)\]

It is normal operating practice for most input and output batches to be respectively of very similar size and to be measured by the same technique. Equation (5) may therefore be simplified to:

\[\sigma^2_{H_t} = n_I \sigma^2_{I} + n_O \sigma^2_{O} \quad (6)\]

Furthermore, under normal operation there is an approximately constant relation both between the size of an input batch and the size of an output batch such that over an appreciable period \(n_O \approx n_I\), and also between the uncertainty of measurement of an input batch and the uncertainty of measurement of an output batch such that \(\sigma_I \approx b \sigma_O\). Thus equation (6) may be approximately rewritten

\[\sigma^2_{H_t} \approx n_I \sigma^2_{I} (b^2 + a) \quad (7)\]

If, for the sake of an example, the daily input and the daily output are regarded as batches, over an appreciable period \(n_I \approx n_O\), and typically, \(a = 1\) and \(1 < b < 2\) for a single batch measurement. However, if as is usual, the output is continuous and a daily output batch is measured by the accumulation of material in a holding store or tank, the effective uncertainty will be approximately \(1.4 \sigma_O\). Thus \(b = 1\), under these conditions and,

\[\sigma^2_{H_t} \approx 2n_I \sigma^2_{O} \quad (8)\]

If a typical input batch is 5 kg of plutonium, the uncertainty attached to its determination may be 0.5% so that \(\sigma = 0.025\) kg. If the uncertainty in \(H\) is not to exceed a 1\(\sigma\) value of, say, 2.5 kg, then \(n_I\) must not exceed

\[
\frac{(2.5)^2}{2(0.025)^2} = 5000
\]

Table I shows the relationship between \(n_I\) and \(\sigma\), for a constant uncertainty in \(H_t\) of 2.5 kg according to equation (8).

This example has used conventional nuclear material accountancy. It has been suggested that greater sensitivity to
TABLE I. Permissible Operating Periods based on Uncertainty in In-process Hold-up

<table>
<thead>
<tr>
<th>Error in output batch measurement $\sigma_o$ kg</th>
<th>Number of input batches $n_I$</th>
<th>Operating days (2)</th>
<th>Nominal throughput t Pu/a (3)</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.2</td>
<td>78</td>
<td>78</td>
<td>12</td>
</tr>
<tr>
<td>0.1</td>
<td>310</td>
<td>310</td>
<td>6</td>
</tr>
<tr>
<td>0.05</td>
<td>1250</td>
<td>1250</td>
<td>3</td>
</tr>
<tr>
<td>0.02</td>
<td>7182</td>
<td>7182</td>
<td>1.2</td>
</tr>
</tbody>
</table>

(1) to reach uncertainty in $H_t (\sigma_{H_t}) = 2.5$ kg Pu
(2) at one input batch per day
(3) with output batch s.d. = 0.5%, 300 days operation per year.

The detection of diversion can be obtained by dividing the process area into smaller accountancy areas around each of which a balance can be drawn. The important features, from the accountancy point of view, are:

(a) that some or all of the flows other than input and output may be zero.

(b) that the input and output rate of each accounting area is essentially the same and essentially equal to the throughput of the plant if, as is normal, Measured Discards and material removed as scrap do not exceed 1-2% of throughput.

(c) that input and output may be continuous or batched. In the former case, these flows may be treated as batches, by taking the flow of material over a convenient time interval (say a day) as a batch.

These facts allow a simple treatment of the linked accountancy areas such as was used for the single facility of Figure 2. If there are $k$ units then

$$H_t = \sum_{i=1}^{k} H_{t_i}$$

(9)

where $H_{t_i}$ is the in-process hold up of the $i$ th unit at time $t$. 


From equation (7)

\[ \sigma_{H_t}^2 = \sum_{i=1}^{k} \sigma_{H_{t_i}}^2 \]  \hspace{1cm} (10)

If there is an approximately constant ratio between the size of the input batch to the plant and the size of the input batch to the i\textsuperscript{th} unit \( n_{I,i} = c_i n_x \) and if the uncertainty of the measurement of the output batch from the i\textsuperscript{th} unit bears an approximately constant relationship to the uncertainty of the measurement of the output batch from the facility, \( \sigma_{O_i} = d_i \sigma_o \). Then equation (10) may be rewritten

\[ \sigma_{H_{t_i}}^2 = n_i \sigma_{O_i}^2 (b_i^2 + a_i) \]  \hspace{1cm} (11)

This may be compared with equation (7) from which it may be seen that the ratio of the variances of \( H_t \) obtained by the single accountancy area (\( V_s \)) to the variance obtained with the multiple accountancy areas (\( V_m \)) is

\[ \frac{V_s}{V_m} = \frac{b^2 + a}{\Sigma \{ c_i d_i^2 (b_i^2 + a_i) \}} \]  \hspace{1cm} (13)

In equation (13) it is probable that if the daily throughputs of all accountancy areas are taken as batches for control purposes \( a_i = 1, c_i = 1 \). Since the measurement on the output batches of product is probably the most precise in the system \( d_i > 1 \) and \( b_i \) may have any value over a small range of units according to the measurement facilities available. On the basis of the assumptions made to derive equation (8) \( b^2 + a = 1 \).

The most likely solution of equation (13) would therefore seem to indicate that the most probable value of \( V_m \) is of the order of \( k \) times the value of \( V_s \), i.e. that there is no advantage in using multiple accountancy areas within the facility to determine the in-process hold-up and that there is probably a disadvantage.
(1) The measurements of input and output batches are probably the most precise that can be made in the system and they make minimum intrusion on operations.

(2) In some cases the measurements can be made by the same techniques and instruments. At most two techniques are required, and quality control and verification effort can be concentrated on these.

(3) The uncertainty in $H_1$ can be reduced by increasing batches, or measurements on the same batch, without increase in quality control or verification effort.

(4) If drift in $H_1$ is observed, its most likely cause is undetected systematic error. At most two measurement systems have to be checked to verify this.

By contrast, a multiple area system utilising process instrumentation or even instrumentation installed for safeguards purposes would have to utilise a larger number of measurement systems, all with individual random and systematic errors, which would be difficult to keep in control and which could result in considerable safeguards intrusion into operations.

3. PRACTICAL APPLICATION

The fine detail of practical systems are likely to be plant specific, but it is possible to generalise on a plant handling liquid feed and product. The findings may then be transferable to a plant handling solid feed and product without too much difficulty.

Material held up within the facility (Figure 3) in recycle loops or in internal scrap recovery loops may be regarded as part of the in-process hold-up, although it may be desirable, in the case of scrap that accumulates in large
quantities with erratic recovery cycles, to account for it as if it were to be recovered elsewhere. This avoids unnecessary inflation of the in-process hold-up.

The additional outputs of discards and scrap for recovery elsewhere may be included in equations (4) and (7) quite simply, giving

$$H_t = \sum_{i=1}^{n_i} I_i - \sum_{j=1}^{n_j} O_{j0} - \sum_{d=1}^{n_d} D_{jd} - \sum_{s=1}^{n_s} S_{js}$$

(14)

where $D_{jd}$, $S_{js}$ are the size of the $j$th discard and scrap batches and $n_d$, $n_s$ the numbers of discard and scrap batches produced up to time $t$. The equation can be made mathematically rigorous for all values of $t$ by defining accountancy periods $t^*$ such that $t = nt^*$, during which all measured flows represent a batch. Equation (14) then becomes:

$$H_{nt^*} = \sum_{i=1}^{n} I_i - \sum_{j=1}^{n} O_{j} - \sum_{d=1}^{n} D_{d} - \sum_{s=1}^{n} S_{s}$$

(15)

When there is no flow in the $j$th period a dummy batch of zero size and variance is inserted in the summed terms.

The variance of $H_{nt^*}$ is

$$\sigma^2_{H_{nt^*}} = \sum_{i=1}^{n} \sigma^2_{I_i} + \sum_{j=1}^{n} \sigma^2_{O_{j}} + \sum_{d=1}^{n} \sigma^2_{D_{d}} + \sum_{s=1}^{n} \sigma^2_{S_{s}}$$

(16)

Since equation (15) can be applied to a complete campaign rather than an equilibrium condition, equation (16) cannot be simplified exactly as equation (5). However, taking note that the batches in each summed term in (15) would probably be of very similar size, and the variance in their measurement would be approximately constant and would bear an approximately constant relationship to one another, it is possible to write

$$\sigma^2_{H_{nt^*}} = \sigma^2_{O}(pb^2 + q + rc^2 + sd^2)$$

(17)

where $\sigma_I = \sigma_{O}$, $\sigma_D = c\sigma_{O}$, $\sigma_S = d\sigma_{O}$, and $p$, $q$, $r$, $s$ are the numbers of non-zero batches accumulated in $n$ periods.
For reasons discussed previously, $b^2$ will be approximately unity. If therefore the variance of $H_{nta}$ is not to be unduly influenced by the last two terms in the bracket, $rc^2 + sd^2$, will have to approximate to $p + q$. This means that the total uncertainty contributed by the accountancy of Measured Discards and Scrap must not be greater than that contributed by the main input and output measurements. This requirement - though seldom stated - is also an important requirement for safeguards approaches based upon the MUF criterion.

If the above requirement can be achieved then:

$$\sigma^2_{H_{nta}} = 2\sigma^2_O (p + q) \approx 4n_1 \sigma^2_O$$  \hspace{1cm} (18)

Comparing equation (18) with equation (8) shows that inclusion of Measured Discards and Scrap flows would approximately halve the number of operating days shown in the third column of Table I under similar conditions.

The assumptions made in achieving this result are probably optimistic. Nevertheless it does indicate that plants of quite large throughput of plutonium can be operated for prolonged periods before the contribution of random error to the in-process hold-up becomes a major uncertainty.

In the liquid-handling plant considered, the measurement systems required would be as follows.

The input batch may be made by total dissolution of a measured quantity of solid, received as a liquid batch, or prepared by extraction of a batch of solid material, leaving a residue, as in dissolution of fuel elements. In the first two cases, the quantity of element of interest may be sufficiently accurately known for the solution to be transferred completely to the feed tank of the process as the input batch. Alternatively, it may be more convenient to treat them in the same way as the third case, where the clarified extract is transferred to a holding (accountancy) tank for measurement before transfer to the feed tank.

The discharge of this accountancy tank to the feed tank may represent the input batch to the process section, but if this event is too infrequent for adequate measurement of the in-process inventory as described above, input batching would need to be carried out in the feed tank itself. For this, the feed tank would need measurement and homogenisation facilities equivalent to the accountancy tank. The input batch to the process section is then constituted by the change in content of
the feed tank over the chosen interval of time. Discharges from
the accountancy tank would conveniently be received at the
boundaries of such time intervals, i.e. accuracy would probably
be improved if the balance was closed with a full accountancy
tank and a nearly empty plant feed tank so that the former
should be discharged after rather than before the balance.

The above function for the feed tank would virtually
limit its configuration to a vertical cylinder or group of
cylinders of comparatively narrow bore. Theoretical calcula­
tions [3], which have subsequently been confirmed in practice
on tanks of less suitable geometry, have indicated that the
elemental content of input batches should be determinable to a
precision (\(\sigma\)) of \(\pm\) 0.5%.

Under this arrangement, the accountancy tank measures
input to the facility, and the feed tank input to the process
section. (The functions are essentially similar to main
head-end store and head-end working store in a solid-handling
plant). Any complete batch in the accountancy tank and the
contents of the feed tank are accurately known at the end of an
accountancy period and are not included in the in-process
inventory.

At the product end of the process section, a number of
functions are performed.

(a) The output, usually continuous, accumulates in a holding
tank.

(b) The accumulated material is examined for specification
acceptance. This may be done either by discharging a
product batch from the first holding tank or by diverting
product flow to another holding tank when a product batch
has accumulated in the first.

(c) Any material out of specification is returned for recycle.

It can be seen that the measurement requirement at the
product end is for a number of tanks of similar specification
to the feed tank discussed above.

An essential requirement is that operational means exist
for ensuring that, so far as is operationally possible, the
accumulation of a batch is not interrupted. When accumulated
in a single tank a batch can be defined by difference in initial
and final content of the tank in terms of the element of
interest. If material is diverted to other tanks during the
accumulation period, then the difference has to be determined
for all tanks involved. This would lead to considerably increased analytical work and a decrease in accuracy and precision of the final measurement.

When flow into a particular product tank is terminated the tank is isolated and after verification of its contents could be sealed and the contents removed from the in-process inventory.

These product tanks would correspond effectively with different product batches in the product store of a solid-processing facility.

The main measurement system for a liquid processing plant controlled by in-process inventory thus consists only of a series of tanks of similar design requiring one procedure for calibration and one system of level measurements and at most two methods of analysis.

The minor measurement system is likely to be highly plant specific, but may be expected to consist of one method of measuring the element of interest in effluents of low element content destined for discard, and one method for measuring the element of interest in effluents of higher element content destined for recovery, should the process generate the latter. The measurements should be as accurate as possible within technical and economic constraints. The essential minimum requirement is that in sum they do not contribute more uncertainty to the determination of in-process hold-up than is contributed by the main measurement system.

The same considerations apply to solid-processing facilities, except that these would tend to produce a smaller proportion of discard material, and a higher proportion of material for recovery. The latter might require particular attention to be paid to homogenisation prior to analysis if the preceding requirement is to be made.

4. CONCLUSIONS

1. In process plants it is conceptually possible to use in-process hold up as a useful measure of safeguards significance.

2. The use of this concept allows conventional materials accountancy to be extended to plants of considerable throughput by adjustment of existing input and output batching procedures and by more frequent analysis using existing analytical procedures.
3. Some attention needs to be paid to optimising plant design, but no fundamental changes are called for.

4. Quality control systems and verification operations are reduced to a minimum in number.

5. The concept can be applied to both liquid- and to solid-processing plants.

REFERENCES


Current investigations of near-real-time material accountancy methods for safeguarding large-scale commercial reprocessing facilities suffer from a lack of available plant data. Few commercial plants are in operation, and those that are do not provide routinely and in near real time the required material balance data. Detailed process simulations on digital computers can provide an alternative until real in-process inventory and throughput measurement results become available. In addition, they provide a relatively inexpensive means of experimenting with different operating schemes and material accountancy strategies with a view to optimizing safeguards effectiveness; in particular, the robustness of statistical test procedures may be investigated.

A computer program has been developed for simulating the operation of the process MBA of a 1000-t reference reprocessing facility running on a Purex flow-sheet. The program describes dynamically all Pu flows and hold-ups throughout the process, beginning at the input accountability tank and ending at Pu-nitrate load out. It writes "true" values for volumes, concentrations and flow-rates to an output file for subsequent simulation of throughput and in-process inventory measurements taken for near-real-time accounting.

The programming language used was SIMULA, which provides a high degree of flexibility in accommodating wide ranges of operating strategies as well as changes in the flow-sheet design.

To produce data that are as realistic as possible not only the individual process units were modelled in detail, but consideration was also given to operating modes (rework, independent operators for each Pu-cycle, etc). The program serves

† The authors deeply regret to announce the unexpected death of their colleague, Andreas Berliner.
primarily as an input for testing the effectiveness (detection probability, timeliness, false-alarm rate) of evaluation procedures for detecting abrupt and protracted material losses under "realistic" conditions such as:

- Variations in burnup of dissolved fuel;
- Variation of "hidden inventory" as a result of deterioration;
- In-pulse column operation; and
- Deviation from steady-state conditions.

As an example of preliminary results, Fig. 1 shows the simulated hold-up in the 2A extractor of the reference plant during a 30-h interval. The inventory variation is induced mainly by fluctuations in the input mass flow-rate.
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