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RESEARCH AND DEVELOPMENT ASSOCIATION,
THE INSTITUTE OF NUCLEAR MATERIALS MANAGEMENT
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FOREWORD

This publication presents the results of the seventh in a series of international symposia on international safeguards. The symposium was held from 14 to 18 March 1994 at the IAEA Headquarters in Vienna in co-operation with the American Nuclear Society (ANS), the European Safeguards Research and Development Association (ESARDA), the Institute of Nuclear Materials Management (INMM) and the Nuclear Society International (Moscow). The symposia are sponsored by the IAEA to foster a broad exchange of information on concepts and technologies related to international safeguards.

Since the last IAEA symposium on this subject, held eight years ago in 1986, the world of safeguards has experienced a number of momentous changes which have opened a new period of intensive development in safeguards. The important events were: the discoveries in Iraq during activities under United Nations Security Council resolutions, South Africa’s decision to become a party to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), the IAEA–Argentina–Brazil–ABACC Quadripartite Safeguards Agreement, the break-up of the former USSR into newly independent States, and the problems encountered in the implementation of NPT safeguards in the Democratic People’s Republic of Korea. The consequences for international safeguards of these events were presented in papers at this symposium, with special emphasis on verification of a State’s declaration as well as on detection of undeclared activities.

Other fundamental changes stem from converging relationships between nuclear arms reductions and the civil use of plutonium, and the international debate on the associated issues. Furthermore, the review and extension of the NPT is due in 1995. Events have opened the possibility for ambitious new concepts for verification regimes. These matters were addressed at the symposium in the opening session and in the closing panel discussion.

Traditionally, a symposium of this nature is an opportunity for discussing experience gained by the safeguards community on conventional safeguards measures and emerging technological developments. While recent developments in these areas were well covered, the emphasis of this symposium was rather different, addressing the recent issues which have evolved from new realities. More than 400 participants from 45 countries and nine international organizations offered their expertise through 200 papers which were presented in 16 oral and four poster sessions. It is hoped that the proceedings will provide guidance in setting the stage for safeguards verification technologies for the 21st century.
EDITORIAL NOTE

The Proceedings have been edited by the editorial staff of the IAEA to the extent considered necessary for the reader's assistance. The views expressed 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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INVITED OVERVIEW PAPERS

(Session 1)

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Invited Paper

IAEA SAFEGUARDS
_STATUS, CHALLENGES AND OPPORTUNITIES_

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Abstract

IAEA SAFEGUARDS: STATUS, CHALLENGES AND OPPORTUNITIES.

The discovery of the clandestine nuclear weapon programme in Iraq raised questions about the ultimate effectiveness of IAEA safeguards as developed in the 1970s and consolidated in the 1980s. In addition, the unresolved ambiguities in the nuclear programme of the Democratic People's Republic of Korea and the unclear nuclear status of some of the newly independent States emerging from the former Soviet Union raise some concern. On the other hand, the decision of South Africa to give up its well advanced nuclear weapon programme and to accept IAEA safeguards on all its nuclear material and the bringing into force of a full scope Safeguards Agreement with Argentina and Brazil increase the credibility of the international non-proliferation regime. One consequence of these developments was the initiation of an in-depth reassessment of the assumptions and methodologies used in the IAEA safeguards system. It was concluded that safeguards should no longer be based exclusively on nuclear materials accountancy, rather it should be extended to look for and follow up inconsistencies that might be early indications of a possible clandestine nuclear weapon development programme. The implementation of new concepts and technologies such as environmental sampling, increased use of sophisticated computer technology, unattended verification systems and remote operation monitoring are expected to bring the right answers to the new challenges and opportunities for IAEA safeguards which emerged from the changed political situation in the world.

In mid-1971 — three years after the entry into force of the Non-Proliferation Treaty — the Safeguards Committee of the IAEA Board of Governors finished its work on the model Safeguards Agreement, and formulated its efforts in what became a fundamental document of the safeguards regime, namely the Information Circular INFCIRC/153. After a phase of intensive development in the 1970s and a phase of consolidation in the 1980s, we are now in a phase of transition.

The INFCIRC/153 safeguards system depends strongly on nuclear materials accountancy and its international verification. It is based on the concept that, as long as all nuclear-weapon-usable material is verified to be in peaceful activities, one can
be confident that it is not used to produce nuclear explosive devices. Since weapon-useable nuclear material is essential for any such device, a tight material control was considered to be sufficient for international non-proliferation verification purposes.

While in the 1970s the concepts and verification techniques were indeed developed and implemented, we saw in the 1980s the full implementation of the system and its continuous improvement. The system was never considered to give total assurance of non-proliferation because of the possibility that weapon-useable material could be produced clandestinely in an unsafeguarded, unreported parallel programme. There was also the theoretical possibility that a country could prepare for a large size nuclear weapon development programme without using any significant quantity of nuclear material. It would stockpile the necessary weapon-useable material in peaceful installations under IAEA safeguards and would only divert this material from safeguards at the last moment, when the Government would be certain that its experts could produce functioning nuclear weapons within a very short period of time.

At any rate, in the INFCIRC/153 concept, the timeliness of detection of diversion was considered to be critical. Of course, this concept turned out to be expensive in terms of inspection effort. There was, certainly, some expectation that any strategy to produce nuclear weapons from unreported weapon-useable material could most probably be detected at an early stage by national intelligence organizations, for example through the use of satellite surveillance. The case of Iraq has taught us otherwise. Even though the Government of Iraq had spent enormous resources in terms of money and manpower on a large complex of dedicated facilities for the nuclear weapon development programme and made remarkable progress in some parts of the programme, this effort only became known after the Gulf war, and only then did the locations involved become accessible to IAEA inspections.

As a consequence, the safeguards community began seriously to rethink some fundamental tenets of safeguards. Already in September 1991, the Director General of the IAEA told the Board of Governors that the IAEA's safeguards system would have to undergo a threefold strengthening to cope effectively with suspect cases, namely through the access to additional information, through the unrestricted access to any relevant location and through the strong support by the world community, explicitly the United Nations Security Council.

Among the strengthening options considered by the Board in 1992, the most important involves the clarification of the IAEA's rights to conduct, when appropriate, special inspections at locations that might be relevant for safeguards. Other options refer to the need for the early provision and verification of design information commencing during construction of facilities, and extending over their life, through commissioning and normal operation. This will provide an improvement in the foundations for implementing nuclear materials accountancy and containment and surveillance measures, in particular such as may relate to undeclared activities within declared facilities. Next, more extensive information will be analysed to look
for patterns that might suggest undeclared nuclear activities within a State. Additional reporting on exports and imports of nuclear material, specified equipment and non-nuclear material will constitute one means to gain access to such information.

After 1992, it became indeed mandatory to contemplate a safeguards strategy that would no longer be based exclusively on nuclear materials accountancy, but that would also look for and follow up inconsistencies in information that might be an early indication of a possible nuclear weapon programme. These points will be discussed later in this paper, whilst other papers in this symposium will discuss many aspects of this new strategy, including some that are related to the streamlining of safeguards, i.e. to the proper management of resources. We shall discuss measures suggested or already implemented to strengthen safeguards and new techniques under development, such as environmental monitoring. Here a word of caution is appropriate. As it took years to achieve political agreement on the INFCIRC/153 system, it might take quite some effort and time to achieve a political consensus on its expansion.

After these more general thoughts, I shall now elaborate on some recent events in the safeguards field which have influenced or are still influencing the development of the expanded safeguards system.

First of all, as already noted, there was the case of Iraq, which exposed some weaknesses of the INFCIRC/153 system. Here was a country — which had agreed to a comprehensive Safeguards Agreement — launching and proceeding far into a nuclear weapon development programme, and all this without reaching the level of alarm within this safeguards system. This event not only opened the way for some rethinking on the INFCIRC/153 system but also promoted the willingness of many countries to permit IAEA safeguards in a less restrictive and more open manner. Several countries have since invited the IAEA to visit any locations it wishes to, even if the location was not reported in the safeguards system. In general terms one can say that through the events in Iraq — and certainly also through the end of the Cold War — the co-operation and openness in many countries has further improved. But the case of Iraq has also given the IAEA a valuable experience that went well beyond normal safeguards practice: for the first time the IAEA learned to recognize the signs of a clandestine nuclear weapon programme, its components, its industrial infrastructure, its research and development requirements, its overt and covert procurement paths.

Secondly, there was the case of South Africa. When South Africa concluded its Safeguards Agreement with the IAEA in 1991, the IAEA was confronted with the problem that major unsafeguarded facilities, including one plant for the production of highly enriched uranium, had been previously operated outside any kind of international control for many years. Therefore, the IAEA General Conference requested the Director General to verify to the extent possible the completeness of the inventory of nuclear material and installations included in South Africa’s initial
report to the IAEA. As a result of the request by the General Conference, an IAEA team made a number of visits to South Africa to consult with officials and to examine historical accounting and operating records of both operating and closed-down facilities. The team’s general conclusion was that it had found no evidence to suggest that the declared inventory of nuclear installations and material was incomplete. Then came unexpectedly, in March 1993, South Africa’s announcement about the abandoned nuclear weapon programme. The Government of South Africa extended at that time an invitation to the IAEA to examine with full transparency the scope, the nature and the facilities of the nuclear weapon programme. The IAEA accepted the invitation. After numerous additional visits and the examination of records, facilities and remaining non-nuclear components of the dismantled nuclear weapons, the IAEA concluded that the cumulative amount of highly enriched uranium produced by the South African pilot enrichment plant was consistent with that programme, and that no indications suggest that there remain any sensitive components of the nuclear weapon programme not having been either rendered useless or converted to commercial non-nuclear applications or peaceful nuclear use. From the findings, one can conclude first that the nuclear weapon programme of South Africa was terminated, secondly that all nuclear devices were dismantled prior to South Africa’s adherence to the NPT, and thirdly that all nuclear material involved in the weapon programme was returned to peaceful uses prior to the conclusion of the Safeguards Agreement. No violation of the NPT or the Safeguards Agreement by South Africa has therefore been detected. I wish to mention here that the Government of South Africa and the national institutions and organizations concerned were very co-operative and arranged access to any location for which access was requested by the IAEA, including access to military installations and ammunitions factories. The South African case has certainly further expanded the experience of the IAEA, sharpened its inspection skills and heightened its capability to look into non-nuclear material related activities of a clandestine nuclear weapon development programme.

Now, let me devote a few words to the situation in the Democratic People’s Republic of Korea (DPRK). The DPRK had acceded to the NPT on 12 December 1985 and brought the Safeguards Agreement with the IAEA into force seven years later, on 10 April 1992. Inspections conducted in 1992 revealed many inconsistencies between the information provided by the DPRK and the results obtained by the analysis of samples of material taken by IAEA inspectors. These inconsistencies suggested that the DPRK had not declared all its nuclear material. The IAEA tried repeatedly until early 1993 — without success — to obtain additional and conclusive information about the DPRK’s declaration. Then, the IAEA Secretariat and Board were shown photographic information on two undeclared sites thought to contain relevant nuclear waste. Because the DPRK would not allow the IAEA to carry out further activities to resolve the inconsistencies and refused the IAEA access to the two sites, the matter was reported to the IAEA Board of Governors, which decided to refer the issue to the Security Council. The Security Council urged the DPRK to
co-operate with the IAEA, but the DPRK initiated the process of withdrawal from the NPT, as a result of which the comprehensive Safeguards Agreement with the IAEA would have also expired. Shortly before the period of announcement of the withdrawal from the NPT ended, the DPRK suspended the withdrawal. The Safeguards Agreement is therefore — in our view and in line with international law — still in force. So far the issue has not been resolved and, at this moment, an IAEA team is in the DPRK for an inspection at the seven declared sites. We were not granted access to the two locations suspected to contain nuclear waste.

Although the circumstances differ, the cases mentioned above have brought home to everyone concerned the fact that initial inventory taking is not easy in States with extensive nuclear programmes prior to an NPT Safeguards Agreement. The matter was successfully resolved in South Africa through the good and open co-operation of the Government. To date, we have not been able to solve this problem in the DPRK.

Following the ratification of the related agreements, the IAEA is now facing the need to verify the completeness of the initial inventory in two large countries — in real and nuclear terms — of South America. After an early ratification by Argentina, the Brazilian Parliament and Senate have now approved the IAEA–Argentina–Brazil–ABACC\(^1\) Quadripartite Safeguards Agreement. Both Argentina and Brazil have operated nuclear facilities, including small enrichment plants, over extended periods of time outside the IAEA safeguards system. We are nevertheless confident that the question of completeness of the initial inventory will, as in the case of South Africa, be rapidly resolved with the full co-operation from Argentina, Brazil and the ABACC.

A similar problem, which may turn out to be more complex, faces us when some of the newly independent States of the former USSR join the NPT as non-nuclear-weapon States. Belarus and Kazakhstan have done so; Ukraine will also, sooner or later. In these cases it may indeed be extremely difficult to reconstruct historical data on nuclear material, even with the utmost support and openness of the Governments involved. Yet, the IAEA will have to satisfy itself that all nuclear material is declared.

Having briefly reviewed the most significant external events that have marked — and are still marking — the recent history of the IAEA and the challenges that result therefrom, let me point to some of the opportunities that are opening up. Specifically, I wish to refer to a few particular aspects of potential importance to the safeguards system, to activities and ideas that will evolve and emerge in the latter part of this decade, namely, first the reinforcement of conventional safeguards, secondly the strengthening and streamlining of safeguards, and thirdly the broadening of the NPT contract.

\(^1\) ABACC = Brazilian–Argentine Agency for Accounting and Control of Nuclear Materials.
I believe that, before the IAEA gets involved in new technologies to strengthen its ability to detect undeclared activities/facilities and in new kinds of verification activities, improvements in conventional safeguards should remain high in the priority list of the IAEA Department of Safeguards.

The great majority of IAEA safeguards work involves the day-to-day verification of nuclear operations under existing Safeguards Agreements. This is by no means a static activity. In such conventional activities, the IAEA will have to cope with an expanded workload in the newly independent States of the former Soviet Union, in Argentina and Brazil, and in South Africa, and cover increasingly challenging facilities, especially in Japan. For nearly a decade, the IAEA and the Department of Safeguards have been required to meet these challenges under zero growth budget constraints, which has added an additional complication.

Most of the papers presented at this Symposium relate to these more conventional safeguards measures. Some address the experience gained by Governments, facility operators and inspectors. Many more address emerging technology developments, which are essential for the IAEA to maintain its capabilities and to establish new technical methods where existing ones are insufficiently effective.

First, on this matter of new technologies, I would like to make a general obvious statement, namely that the use of computers by inspectors in the field is having the most profound impact on safeguards implementation and yet we are at a very early stage of this revolution. Secondly, in the area of safeguards instrumentation development, I wish to mention specifically the emergence of unattended verification systems and of digital image surveillance.

Unattended verification systems have already been successfully used to reduce inspection effort, decrease the burden on facility operators and provide expanded verification coverage. Unattended verification systems combine computer operated non-destructive assay systems with containment and surveillance, such that the measurements are made under controlled and authenticated arrangements. Such systems are sometimes the only way to implement safeguards at complex nuclear facilities — especially in automated plants. Several unattended monitoring systems are now under consideration, under development and even in use. Examples are the plutonium assay systems for use in Japanese mixed oxide conversion and fuel fabrication facilities, a core discharge monitor developed in Canada for on-load power reactors, the Consulha system developed in France for monitoring the unloading of spent fuel and the integrated verification system under development in Germany.

The development of the second generation bundle counter is particularly important since it is the prototype for the next generation of unattended monitoring systems. The goal is to develop modular hardware and modular software incorporated in an open architecture system. With this concept, the flexibility for accommodating a variety of applications will be designed into the basic architecture, without the need to establish a customized system for each facility. Moreover, since an international standard will be employed, developers in various laboratories
around the world can contribute sensors that can be accommodated within such a
system, confident that appropriate interfaces will be available.

In the last two years, there has been a tremendous growth in digital image
transmission, together with the adoption of agreed standards for high speed real time
data compression, digital imaging, digital processing, digital storage, as well as digi­
tal encryption of image data. Digital image technology will have a fundamental
impact on the surveillance measures used by the IAEA. The overall effectiveness of
our optical surveillance will be significantly improved and the technology will allow
innovative applications, such as the use of 'mail-in' arrangements and remote
monitoring. The mail-in concept foresees the mailing of encrypted surveillance
information by the facility operator to the IAEA offices. This concept would save
IAEA inspection resources by reducing the need for IAEA inspectors to visit certain
facilities, such as light water reactors, as frequently as currently required.

Furthermore, the IAEA continues to investigate innovative methods to apply
randomization principles in safeguards. Recently, a field test was performed on the
application of short-notice random inspections for inventory change verification at
a fuel fabrication plant. According to this approach, the plant operator declares the
contents of nuclear material items before knowing if an inspection will take place to
verify them.

Indeed, the IAEA Safeguards Development Programme includes many
requirements and tasks related to the current routine implementation of safeguards.
Much of the development work is carried out in the frame of Member States Support
Programmes under which not only financial help, but first of all, the relevant exper­
tise of these Member States comes into play.

Beyond the development of hardware and software, our catalogue of work
covers a host of other activities to ensure that IAEA safeguards continue to provide
the assurance sought by our Member States. This work includes updating the
safeguards criteria currently in effect for 1991–1995, to strengthen them as soon as
techniques and inspection modes are judged appropriate and feasible. Examples of
such elements are the application of safeguards to small quantities of nuclear
material, the streamlining of the departmental procedures for granting requests for
an exemption of nuclear material from safeguards, and for the termination of
safeguards for measured discards.

Now, let me devote a few words to the strengthening of the safeguards system.
In reviewing the Iraqi experience, it is clear that IAEA safeguards did not provide
adequate assurance that States subject to comprehensive Safeguards Agreements
would submit all nuclear materials to safeguards or that undeclared operations were
not carried out in facilities that were submitted for safeguards. As a result, a substan­
tial amount of work has been initiated in the Department of Safeguards on new
approaches aimed at strengthening the safeguards system, and while most of the
evaluation and planning activities necessary to realize these improvements will not
be completed for some time, the outcome of this work will have a fundamental
impact on technical aspects of IAEA safeguards in the future.
In 1993, the General Conference and the Board of Governors asked the Secretariat to explore alternative means to strengthen the safeguards system and to improve its cost efficiency. In April 1993, the Director General’s Standing Advisory Group on Safeguards Implementation (SAGSI) had formulated a set of specific recommendations. After having been discussed by the Board in its June 1993 meeting, these recommendations were translated into the Secretariat’s development programme for a strengthened and more cost effective safeguards system, under the internal codename ‘Programme 93+2’. This effort will provide for the evaluation of the technical, legal and financial implications of various recommendations, first of all those of SAGSI. This is discussed in paper IAEA-SM-333/223 by R. Hooper, Programme Manager and Director of Concepts and Planning in the Department of Safeguards.

The development programme for strengthening IAEA safeguards requires extensive participation by Member States. All strengthening measures that go beyond the scope of Safeguards Agreements can only be implemented with the agreement of the States concerned. The Secretariat should be in a position to make a proposal, including the legal implications, for a strengthened and more cost effective safeguards system by early 1995.

One area that appears particularly interesting is the application of environmental sampling for safeguards purposes. These methods allow for chemical and isotopic analysis of minute samples (as small as $10^{-15}$ g) which may be collected within declared facilities or away from nuclear facilities, e.g. water, soil or biota samples, that might provide indications of clandestine activity. This method has been and will continue to be used in Iraq.

Several Member States have offered their assistance in the conduct of environmental monitoring field trials and related technical areas. A plan for environmental sample collection and analysis has been established for 1994 with a number of participating Member States. The usefulness of field trials is not limited to environmental monitoring. Ways and means to increase the co-operation with national accounting systems are also candidates for field trials in two Member States and consultations have begun with a third one. Informal consultations indicate that other States may be coming forward with offers to host field trials.

At this point, I wish to return to a broader and more political issue, to state that the INFCIRC/153 system has not yet achieved the desired broad degree of universality. As any worldwide arms limitation arrangement, the non-proliferation regime will only achieve its full intended purpose if all relevant countries participate. Substantial progress has been made over recent years in this field: South Africa joined the NPT; Argentina, Brazil and Chile ratified the Treaty of Tlatelolco; China and France joined the NPT as nuclear weapon States; and full scope safeguards will soon be in force in Brazil and Argentina. Algeria has announced its intention to join the NPT. On the other hand, India, Israel, Pakistan and several newly independent States of the former Soviet Union, in particular Ukraine, remain outside the system.
However, things are moving. New confidence-building initiatives have been put forward by the United States of America. In particular, if and when the process of nuclear arms reduction in nuclear weapon States reaches the phase of releasing substantial quantities of direct weapon-usable material from the weapon programmes into civil use or possibly only to storage, IAEA safeguards on such material could provide assurances that the material would not be used in nuclear weapon programmes again. Until now only the highly enriched uranium released when South Africa terminated its nuclear weapon capabilities falls into this category of direct-use material previously used in a weapon programme. This material is now placed under IAEA safeguards and is dedicated to peaceful uses. In this connection the US initiative to submit excess fissile material from the US defence programme to IAEA safeguards is an important step which will give the IAEA safeguards system additional responsibilities, possibly beginning already in 1994. This is discussed in paper IAEA-SM-333/233.

The IAEA may also be given a role in the verification of the Comprehensive Test Ban Treaty, discussed at the Conference on Disarmament in Geneva, and possibly also in the verification of a fissile material production cut-off convention which has been suggested recently. These are new challenges and opportunities for the IAEA. While they are to a large extent a separate matter, they will certainly impact on the IAEA’s safeguards system in the future.

To conclude my broad presentation, I cannot talk about challenges and opportunities of the IAEA’s safeguards system without also referring to certain developments that may threaten its credibility.

First, there is the present ambiguity in the DPRK. If the IAEA remains unable to verify that there is no nuclear weapon programme in the DPRK, the application of safeguards there will at some point be of questionable value. We can only hope that, eventually, a solution will be found which will confirm the peaceful character of the nuclear programme of the DPRK.

Secondly, there are the long existing restrictions on our resources: more than ten years of zero growth budget during which there have been continuing increases in workload have unfortunately led to a reduction in our inspection goal attainment, if not yet to an unacceptable degree. Although I am fully aware of the economic situation in many of our Member States, it must be emphasized that with a continuing zero growth budget the IAEA will not be able to cope with the extended programmes and demands placed on it.

For the successful execution of its functions, the IAEA needs the continuing full support of its Member States, individually and collectively, if the reputation of the safeguards system is to be maintained.

Certainly, the IAEA has reacted to the challenges of recent years and has tackled the opportunities by launching important initiatives. It is, however, up to the Member States and their political judgement to determine the objectives and scope of our work. The discussions on our programme and budget in the IAEA Board of
Governors and the General Conference, and certainly also the results of the NPT Review and Extension Conference in April 1995, will have a strong influence on the direction in which IAEA safeguards will develop. I am convinced that through its safeguards activities the IAEA has also contributed substantially to the promotion of the peaceful use of nuclear energy throughout the world, by providing assurance that nuclear trade and co-operation would not lead to the proliferation of nuclear weapons. Without the verification activities of the IAEA, nuclear commerce would have hardly found the present degree of public acceptance.

The new challenges and opportunities may indeed permit the IAEA to contribute even more directly to world peace and prosperity.
Invited Paper

1995 NON-PROLIFERATION TREATY REVIEW AND EXTENSION CONFERENCE
Policy and institutional issues

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Abstract

1995 NON-PROLIFERATION TREATY REVIEW AND EXTENSION CONFERENCE: POLICY AND INSTITUTIONAL ISSUES.

A preparatory process is under way for the Non-Proliferation Treaty review and extension conference, which will take place from 17 April to 12 May 1995. The conference will conduct the regular five year review of the operation of the Treaty, but the major question facing the conference will be the duration of the extension of the Treaty. The paper outlines the views of advocates both of extension for a fixed period and of indefinite extension. The major issues are: the extent to which Article VI of the Treaty — nuclear disarmament — has been implemented; the adequacy of security guarantees by the nuclear weapon States to the non-nuclear-weapon States; and the concern of Arab States about an indefinite commitment to non-nuclear status in the absence of a comparable commitment by Israel. The principal procedural issues are also outlined.

The Treaty on the Non-Proliferation of Nuclear Weapons (NPT) is the single most important component of the non-proliferation regime. Under it the 158 non-nuclear-weapon States party have committed themselves not to manufacture or otherwise acquire any nuclear explosive device and to accept IAEA safeguards on all source or special fissionable material to verify that commitment. Under it also the five nuclear weapon States party have committed themselves to embark on effective measures relating to nuclear disarmament. In accordance with Article X of the Treaty, which entered into force in 1970, the States party to it need to decide, in 1995, on its extension. Article X provides that "twenty-five years after the entry into force of the Treaty, a conference shall be convened to decide whether the Treaty shall continue in force indefinitely, or shall be extended for an additional fixed period or periods. This decision shall be taken by a majority of the parties to the Treaty."

A preparatory process for the extension conference is already under way and a decision has been taken that the 1995 conference will take place from 17 April to
12 May 1995 in New York. The conference will conduct the regular five years review of the operation of the Treaty, but the major issue before the conference will be the extension decision. Although the Treaty provides that the extension decision will be taken by a majority of the parties to the NPT, there is a widespread feeling that, for a Treaty of such vital importance, the extension decision needs to be reached, if possible, unanimously or by consensus. A majority of parties have already declared themselves in favour of indefinite extension of the Treaty. There are others, however, who advocate extension for a fixed period, accompanied by a mechanism to enable further extensions. Much debate on these matters is going on amongst lawyers and politicians, including on the different modalities and formulations through which extension could be effected. It is important to understand the reasons underlying the views of advocates both of extension for a fixed period and of indefinite extension.

A major — and perhaps the most important — first substantive issue centres on the extent to which Article VI of the Treaty has been implemented. Article VI provides that each of the parties to the Treaty "undertakes to pursue negotiations in good faith on effective measures relating to cessation of the nuclear arms race at an early date, and to nuclear disarmament, and on a Treaty on general and complete disarmament under strict and effective international control". For some of the non-nuclear-weapon States party to the NPT the implementation of Article VI was part of the deal they entered into in 1970: in return for a commitment by the non-nuclear-weapon States not to acquire nuclear weapons, the nuclear weapon States committed themselves to eliminate their nuclear arsenals. The argument runs that, since the nuclear weapon States have not totally honoured their part of the deal, indefinite extension of the NPT would not be appropriate. Particularly relevant to proponents of this view is the commitment made in the preamble to the Treaty to achieve a Comprehensive Test Ban Treaty at "an early date". Such proponents also maintain that if the Treaty is extended indefinitely as it stands, it will perpetuate discrimination between nuclear weapon States on the one hand and non-nuclear-weapon States on the other.

Many other parties, of course, believe that substantial progress has been made towards nuclear disarmament and accordingly towards implementation of Article VI of the NPT. When the START I and II Treaties are implemented, the two major nuclear weapon States will have reduced their arsenals from 30,000 warheads to about 3000 each. Moreover, the recent initiative by President Clinton for a Multilateral Agreement Prohibiting the Production of Fissile Material for Weapon Purposes, and the decision now taken to start negotiations on a Comprehensive Nuclear Test Ban Treaty at the Conference on Disarmament are further signs of the kind of progress which could justify indefinite extension of the NPT. The argument runs that extension of the NPT, the cornerstone of the non-proliferation regime, should not be held hostage to further disarmament measures. In other words, the best should not be the enemy of the good.
The second substantive issue raised by some of the non-nuclear-weapon States party to the Treaty relates to security guarantees. Commitments, using different formulations, have been made by the five nuclear weapon States not to attack or threaten to attack with nuclear weapons any non-nuclear-weapon State that has made a legally binding non-proliferation commitment. These commitments are referred to as 'negative security assurances'. There is also a Security Council resolution, SCR 255 (1968), which embodies commitments by some of the nuclear weapon States to provide or to support immediate assistance, in accordance with the United Nations Charter, to any non-nuclear-weapon State party to the NPT ‘that is a victim of an act or an object of a threat of aggression in which nuclear weapons are used’. These undertakings are referred to as ‘positive security assurances’. Critics argue that such commitments are too vague, highly qualified, and not sufficiently far-reaching. They argue that if non-nuclear-weapon States commit themselves indefinitely to maintain that status, they are entitled to more tangible and credible security assurances, whether negative or positive.

Another substantive issue at the forefront of debate is the perceived concern amongst many of the Arab States party about committing themselves indefinitely to the NPT as non-nuclear-weapon States, in the absence of a comparable commitment by Israel. Such States argue that it would be difficult for them, from a national security perspective, to commit themselves to indefinite extension of the NPT without real progress by Israel towards a nuclear non-proliferation commitment. In the view of this group of States, an important step would be Israel’s accession to the NPT, to be followed by the establishment of a nuclear-weapon-free zone in the Middle East region. However, Israel takes the view that it could make a nuclear non-proliferation commitment only in the context of a Middle East nuclear-weapon-free zone established as part of a comprehensive peace agreement that takes into account the region’s history and characteristics.

These are some of the key substantive issues which will have a bearing on the major question facing the conference, namely the duration of the extension of the Treaty. However, a number of procedural issues — not entirely unrelated to substantive ones — are yet to be resolved. Among them are the Rules of Procedure for the conference. Some advocate a right to adopt the conference Rules of Procedure by a vote. Others insist that the Rules of Procedure can only be adopted by consensus. Clearly underlying the debate is the realization of the impact which the Rules of Procedure might have on the proceedings of the conference itself and on its outcome.

Another procedural issue still outstanding relates to the question of observers to the extension conference, i.e. non-NPT parties and non-governmental organizations. Included in that issue is the question of how much the outcome of the conference is of interest to non-NPT States and to the public at large. My expectation is that an agreement will eventually be reached on inviting non-NPT States and possibly non-governmental organizations to the conference as observers. Another procedural question is whether the extension component of the conference will be preceded or
followed by the review component. While some States believe that the conference should address first the major question before it, namely the extension of the Treaty, others believe that since a review of the Treaty might influence the decision on extension, it should precede the discussion on extension. Another question yet to be decided is whether the conference will have the regular committees usually established for the purpose of review throughout its proceedings, or whether it will work in a more informal setting with respect to the question of the duration of extension, employing such modalities as "friends of the President", etc. A further procedural question to be settled is how to finance the extension conference and to fix the rate of assessment for each party. Included in that discussion is the question of whether the depositary Governments, Russia, the United Kingdom and the United States of America, should bear a larger share of the financing, and also whether the other nuclear weapon States, i.e. France and China, would also have to pay more in their capacity as nuclear weapon States.

The IAEA has a major interest in the outcome of the 1995 conference because of the impact on the application of IAEA safeguards. The majority of the Safeguards Agreements under which the IAEA implements safeguards are those pursuant to the NPT. Such agreements provide, inter alia, that they shall remain in force "as long as the State is party to the Treaty".

Because of its central role in relation to the implementation of Article III of the NPT on verification and of Article IV on the transfer of technology, the IAEA has been asked to prepare relevant background papers for the 1995 conference. These papers are now being prepared and will be reviewed at the third meeting of the preparatory committee in Geneva in September 1994. The fourth and final preparatory committee is scheduled to take place in January 1995 in New York, to be followed in April by the conference itself. It is to be hoped that whatever the outcome of the conference, it will be one that fosters the cause of non-proliferation and efforts to make it universal.
Invited Paper

US FISSILE MATERIAL INITIATIVES
Implications for the IAEA

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Abstract

US FISSILE MATERIAL INITIATIVES: IMPLICATIONS FOR THE IAEA.

In a statement of US non-proliferation policy on 27 September 1993, President Clinton proposed new initiatives to reinforce the international nuclear non-proliferation regime. Several have important implications for the IAEA, and especially for the IAEA’s crucial role in applying international safeguards. The United States of America intends eventually to submit all fissile material no longer needed for the US deterrent or other defence purposes to inspection under the US–IAEA voluntary Safeguards Agreement. It is expected that the IAEA will begin applying safeguards on some of this highly enriched uranium (HEU) and plutonium in non-sensitive forms by the end of 1994. Looking ahead, submitting nuclear weapons components to IAEA safeguards will pose particularly challenging issues, since the USA and the IAEA must devise an inspection approach that will provide an opportunity for credible verification while protecting sensitive nuclear weapons design information. This will be the first instance in which the IAEA will play a role in verifying certain aspects of the disarmament process, and new resources will need to be found. President Clinton has also called for an international treaty prohibiting the production of HEU and separation of plutonium for nuclear explosives or outside international safeguards. In October 1993 the United Nations General Assembly adopted by consensus a resolution along similar lines. Verification of the cut-off treaty will place significant demands on the IAEA’s ingenuity and resources. Adequate verification will require the IAEA to have the right to carry out its safeguards responsibilities to ensure against undeclared activities prohibited by the treaty. The IAEA will also be called upon to safeguard old reprocessing facilities that were never designed to facilitate the application of safeguards. It will also take on such new tasks as verifying that certain enrichment and reprocessing plants are shut down and safeguarding enrichment facilities that are producing HEU. These challenges will require the development of new safeguards approaches. A cut-off treaty could result in a doubling or tripling of the resources required for the IAEA’s safeguards function. At the same time, it will significantly increase the IAEA’s contribution to an effective international non-proliferation regime.
1. INTRODUCTION

In a comprehensive statement of US non-proliferation policy on 27 September 1993, President Clinton proposed a number of major new initiatives to help strengthen US policy and practice in this area of vital importance to US and global security and, more generally, to help reinforce the international nuclear non-proliferation regime. Some of these initiatives can and will be carried out by the United States of America acting on its own. In the case of others, however, it is the hope of the USA that it will be joined by other countries in a common effort to build a more secure future for all humankind. Many of the proposed initiatives have important implications for the IAEA, and especially for the IAEA’s crucial role in applying international safeguards.

Key among the initiatives proposed by President Clinton are several designed to mitigate the continuing threat posed by fissile material usable for weapons. This paper focuses on two initiatives in particular: the US intention to submit excess fissile materials from the US defence programme to IAEA safeguards, and the proposed fissile material production cut-off treaty.

2. SUBMITTING EXCESS FISSIONE MATERIAL FROM US WEAPONS TO SAFEGUARDS

The USA has begun a process that will lead to the eventual submission of all US fissile material no longer needed for the US deterrent or other defence purposes to inspection by the IAEA. As a nuclear weapon State party to the Treaty on the Non-Proliferation of Nuclear Weapons, the USA is not obliged to place its nuclear activities under IAEA safeguards. However, in 1980 the USA concluded a Safeguards Agreement with the IAEA which makes eligible for safeguards all source and special fissionable materials on all US nuclear facilities except those associated with activities of direct significance for national security. Historically, the IAEA has typically selected for safeguarding one to three of the some 230 nuclear facilities that the USA has made eligible for inspections. It is the US intent to place excess highly enriched uranium (HEU) and plutonium from the US defence programme under this USA–IAEA voluntary Safeguards Agreement.

The Nuclear Weapons Council, an interagency body charged with the responsibility of determining how much nuclear material will no longer be needed to meet US defence requirements, has made some initial decisions on what nuclear materials are excess and therefore eligible for safeguards. This will be a continuing process, and it is the intent of the USA eventually to submit all fissile material no longer needed for the US deterrent or other defence programmes to inspection by the IAEA. This will take some time, and it is impossible to predict at this stage how long this process will take.
Nuclear materials excess to defence requirements are located in a variety of facilities, some of which maintain a national security mission. Excess materials will need to be segregated from nuclear materials remaining in the strategic reserve in order to permit IAEA inspection. They will also be in a variety of different forms including residues, spent fuel, HEU in metal form and plutonium in oxide and metallic forms. Much of it will be in the form of nuclear weapons components since the USA currently has no facilities for converting such components into less sensitive forms.

The USA is proceeding in a step-by-step fashion. Current plans contemplate placing several tonnes of former defence materials in non-sensitive forms of HEU sometime in 1994. Our intention is to set aside this HEU in a vault in Oak Ridge, Tennessee. Sometime in spring 1994 we hope to submit the required notification to Congress, where it must lie for 60 calendar days before it may take effect. If everything goes according to plan, the IAEA should have conducted its initial verification by September 1994.

We are also planning to submit to safeguards several tonnes of plutonium in non-sensitive oxide and metallic form located in Hanford, Washington, and/or Rocky Flats, Colorado. Options are being developed for this phase, and we hope that IAEA inspections on such excess plutonium could begin by the end of 1994.

Submitting nuclear weapons components to IAEA safeguards will pose particularly challenging issues, since the USA and the IAEA must devise an inspection approach which will provide the IAEA with the opportunity for credible verification of the nuclear material concerned while at the same time protecting sensitive nuclear weapons design information.

The USA is conducting two major reviews to address the issue of component inspection. In the first study, we are examining potential inspection and measurement alternatives to those involved in standard IAEA practices. Such approaches include verification of non-sensitive characteristics of weapons components, or confirmation of sensitive information, without such information being revealed to inspectors. At the same time, a study is under way to examine whether declassification of certain information about nuclear weapons components, such as mass, would involve serious proliferation risk.

We plan to complete both studies within several months. Their results will be closely co-ordinated to identify inspection options that result in a high level of verification while minimizing proliferation risk. The USA intends to work closely with the IAEA in assessing the inspection options and in designing procedures which will provide a high degree of assurance to the international community that material removed from nuclear weapons will not be returned to such use.

In addition to this unilateral step, President Clinton and President Yeltsin issued a joint summit statement on non-proliferation on 14 January 1994 in which

"They agreed among other things to establish a joint working group to consider steps to ensure the transparency and irreversibility of the process of
reduction of nuclear weapons, including the possibility of putting a portion of fissionable material under IAEA safeguards. Particular attention would be given to materials released in the process of nuclear disarmament and steps to ensure that these materials would not be used again for nuclear weapons.”

They also agreed to consider including in their voluntary safeguards offers all source and special fissionable materials, excluding only those associated with activities having direct national security significance.

We hope that the Russian Federation will be able to join us in placing materials no longer needed for its defence under safeguards. US and Russian steps in this direction can have only a salutary impact on arms control, non-proliferation and international and regional peace and security. They will also have a major impact on the IAEA, as this will be the first instance in which it will play a role in verifying certain aspects of the disarmament process. Over time they will also have an important effect on the costs of IAEA safeguards. Some argue that the benefits of safeguards in nuclear weapon States are not commensurate with the costs. I think they are, and many share this view. Such safeguards are in the security interests of all States. We must therefore find the resources for the application of safeguards to nuclear materials excess to defence needs.

3. TREATY ON CUT-OFF OF FISSILE MATERIAL PRODUCTION

In his non-proliferation statement of 27 September 1993, President Clinton also called for an international treaty prohibiting the production of HEU and the separation of plutonium for nuclear explosives or outside international safeguards.

In October 1993, the United Nations General Assembly adopted by consensus a resolution on the prohibition of the production of fissile material for nuclear weapons or other nuclear explosive devices. This resolution, inter alia,

— expresses the conviction of the international community that a non-discriminatory, multilateral and internationally and effectively verifiable treaty banning the production of fissile material for nuclear weapons or other nuclear explosive devices would be a significant contribution to nuclear non-proliferation in all its respects;
— recommends the negotiation of such a treaty in the most appropriate international forum;
— requests the IAEA to provide assistance for examination of verification arrangements for such a treaty as required; and
— calls upon all States to demonstrate their commitment to the objectives of such a treaty.

The USA attaches great importance to the proposed treaty and envisages a key role for the IAEA in verifying the commitments made pursuant to it.
The purpose of such a treaty is to strengthen international nuclear non-proliferation norms generally, and to give constraints on nuclear material usable for weapons the additional weight of a binding international commitment.

The USA believes the main undertakings of such a treaty should be commitments to:

— refrain from producing fissile materials for nuclear explosive devices;
— refrain from assisting other States to produce fissile materials for proscribed purposes; and
— accept IAEA safeguards to verify the undertaking not to produce fissile materials for purposes proscribed by the treaty.

The USA believes that the treaty should be open to universal membership, and should be non-discriminatory in its provisions.

The USA does not envisage the treaty as prohibiting the production of HEU or the separation of plutonium for civil nuclear activities under safeguards, nor does it see the treaty as requiring full scope safeguards.

It would, however, have the important effect of imposing a ‘cap’ on the fissile material available to the treaty’s signatories — both nuclear weapon States and non-nuclear-weapon States — for nuclear explosives.

It is particularly important that the ban on HEU production and plutonium separation for nuclear explosives be credibly verified. The USA sees the IAEA as the appropriate agency to carry out this task. The verification measures themselves should be non-discriminatory and applied in a similar manner in all States party to the treaty.

The verification of the basic obligations of the treaty, now commonly referred to as the cut-off convention, raises a number of significant safeguards questions, the answers to which will have profound implications for the IAEA’s safeguards system. By far the most important question is what facilities and materials would be subject to safeguards under the treaty. There are various possibilities.

The first is what I would call a limited option. It would apply safeguards to all reprocessing and enrichment facilities in States party to the treaty as well as the plutonium and HEU products of these plants. One question is how far through the fuel cycle safeguards should follow the HEU and plutonium. In the view of the USA, in order to provide minimum credible verification of the basic undertaking of the treaty, safeguards would have to apply to these materials at least up to the point of their insertion in a reactor. Whether safeguards would follow the HEU and plutonium through the spent fuel storage stage would be an open question under this option, but safeguards would of course apply to any reprocessing of the spent fuel.

A second option would be a more comprehensive one in which safeguards would apply to all nuclear materials in a State party to the cut-off convention except the unsafeguarded special fissionable materials produced prior to its entry into force. This would not be full scope safeguards but would provide a greater level of...
assurance of the undertakings of the cut-off convention than would the limited option. It would, however, significantly raise the cost of verification.

A third approach would be an evolutionary one which would start with the limited option described above and move over time to the comprehensive option. The broadening of safeguards coverage could take place according to a predetermined schedule, or the parties to the treaty could meet periodically to take a decision on whether and to what extent safeguards coverage should be expanded under the treaty.

It is also possible to consider certain transparency measures to supplement classical safeguards. For example, States parties could declare the location of all nuclear activities in their territories, whether civil or military. Depending on whether the limited or comprehensive option were selected and on the sensitivity of the activity, these declarations could range from a simple declaration of the location and purpose of facilities to detailed reporting on the nature of the activities and the quantities of nuclear material. Such transparency measures would, of course, be a complement to, not a substitute for, IAEA safeguards.

Clearly, States will have to weigh options such as these (and perhaps others) very carefully. Each has profound implications for the credibility and effectiveness of the IAEA’s safeguards system as well as the resources required not only for the IAEA but also for the States and operators being inspected. Each will also have to be assessed against its political acceptability and its effect on the IAEA’s existing safeguards regime.

The USA also believes that adequate verification of a cut-off convention will require the IAEA to have the right to carry out its safeguards responsibilities to ensure against undeclared activities prohibited by the convention. Special inspections under a cut-off convention raise certain questions since States will have sensitive facilities on their territories. Perhaps some form of managed accessibility along the lines of that found in the Chemical Weapons Convention (CWC) should be examined for its applicability to the cut-off convention.

Several important technical safeguards questions will also arise under such a treaty. The treaty, as we envisage it, will prohibit the production of HEU, plutonium and $^{233}$U for nuclear explosives. It would not, however, prevent the production of tritium or the use of HEU for non-explosive military uses such as naval reactors. In the case of tritium production, safeguards may have to be applied to HEU fuel subject to the treaty without exposing inspectors to information which States regard as classified. Some States with nuclear propulsion programmes will most likely wish to cease the application of safeguards to HEU subject to the cut-off convention when this material is to be used as fuel in nuclear submarines. Article 14 of INCIRC/153 sets forth a general procedure for non-application of safeguards to nuclear material for non-proscribed military uses under the Non-Proliferation Treaty (NPT). While we have no precedent for implementing this provision in NPT Safeguards Agreements, we are very likely to face requests with a cut-off convention. We will need
to think very carefully about the appropriate procedures under which HEU may be removed from safeguards coverage. On the one hand, such procedures need to recognize the legitimate interests of the State to protect classified information. On the other hand, if not sufficiently rigorous, the procedures could be a huge loophole which could undermine the credibility of the convention. Detailed questions will need to be resolved such as (a) at what point would safeguards cease to apply, (b) what reporting should be required of the withdrawing State, and (c) at what point should the withdrawn material be reintroduced into the safeguards system.

The IAEA will also be called upon to safeguard old reprocessing facilities which were built to separate weapons grade plutonium for a nuclear weapons programme and were never designed to facilitate the application of safeguards. This will place significant demands on the IAEA’s ingenuity and resources. The IAEA will also take on some new tasks such as verifying that certain enrichment and reprocessing plants are shut down and safeguarding enrichment facilities which are producing HEU. These challenges will require the development of new safeguards approaches.

Another important issue arising from the proposed cut-off convention is what sort of legal instrument should be used to define the IAEA’s safeguards rights and obligations in verifying the undertakings of the convention. In considering this question we must keep two facts in mind. First, the parties will be nuclear weapon States, non-nuclear-weapon States which have full scope Safeguards Agreements and non-nuclear-weapon States which have certain unsafeguarded nuclear activities. Secondly, whatever the legal form or forms of the safeguards arrangements chosen, the verification of the convention’s undertakings must be non-discriminatory in its effect. The safeguards obligations of nuclear weapon States, NPT parties and States without a full scope Safeguards Agreement must be the same.

Several options are possible. One is to draft a single, standard Safeguards Agreement for all States party to the convention. It would have the advantage of being non-discriminatory in form and substance. It would also provide an opportunity to tailor the Safeguards Agreement to the provisions of the cut-off convention. For example, safeguards obligations could be created in perpetuity. It could also provide for managed access. The Safeguards Agreement could be in a protocol to the cut-off convention along the lines of the verification provisions of the CWC or in separate Safeguards Agreements between the IAEA and the States parties, using the NPT model. The disadvantages of this approach are that it might take a long time to negotiate, would be redundant with INFCIRC/153 agreements, and any conflicting obligations between INFCIRC/153 and the new safeguards arrangement would have to be clarified.

A second approach would be to utilize a different agreement for each type of State party. INFCIRC/153 could satisfy the safeguards requirements of the cut-off convention for NPT parties. The voluntary offers of the nuclear weapon States could be amended to provide for an undertaking consistent with the basic obligations
of the treaty, to limit the right to withdraw material for non-explosive military uses and to obligate the IAEA to apply safeguards to materials subject to the treaty. Alternatively, the nuclear weapon States could enter into a new Safeguards Agreement with the IAEA which would make all their relevant facilities and materials subject to IAEA safeguards. Non-nuclear-weapon States which do not have full scope safeguards could conclude a similar type of Safeguards Agreement with the IAEA. While each of these agreements would differ in form, they would create identical obligations for all States for the purposes of this treaty.

There might be a number of variations to these options. In any event, consideration will need to be given to the available options in the light of their political acceptability, their relation to existing Safeguards Agreements and their practicality of negotiation.

We do not expect that a cut-off convention and its associated safeguards arrangements will be concluded over night. Many issues need to be thoroughly vetted and resolved not only with respect to the safeguards aspects but also with respect to such matters as requirements for entry into force, duration, withdrawal and review of provisions. Nevertheless, the USA strongly favours moving forward in negotiating this convention as expeditiously as possible.

There is little doubt that a cut-off convention, once in effect, will have a profound impact on the IAEA’s safeguards responsibilities. It will greatly increase its inspection activities and could result in a doubling or tripling of the resources required for the safeguards function of the IAEA. Most importantly, it will significantly increase the IAEA’s contribution to an effective international non-proliferation regime.
Invited Paper

CONFIDENCE BUILDING MEASURES FOR PLUTONIUM AND HIGHLY ENRICHED URANIUM

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Abstract

CONFIDENCE BUILDING MEASURES FOR PLUTONIUM AND HIGHLY ENRICHED URANIUM.

The IAEA Secretariat initiated discussions with key States on the need for, and the means to achieve, improved transparency in plutonium and highly enriched uranium (HEU) programmes. The first part of the paper describes the motivations for such an enterprise, which arise from the converging relationship between nuclear arms reductions and the civil use of plutonium, and the international debate on the associated issues. Next, the role suggested for a system of international plutonium and HEU confidence building measures is considered in relation to non-proliferation, physical protection and radiological safety. Examples of specific confidence building measures suggested for consideration are then identified. The States involved are now in the process of determining which measures can be implemented and how best to arrange for their implementation.

1. THE NEED FOR CONFIDENCE BUILDING MEASURES FOR MILITARY AND CIVIL USE OF Pu AND HEU

The nuclear world is undergoing a fundamental change: the peaceful use of Pu is expanding while military use is declining. This expansion comes amid concerns voiced by some States and non-governmental organizations that expanding the civil use of Pu will increase the risks related to proliferation, terrorism and safety. These concerns were voiced most recently in connection with the shipment of Pu from France to Japan, and the licensing of the thermal oxide reprocessing plant (THORP) in the United Kingdom.

At the same time, steps are under way to eliminate most of the nuclear weapons in the US and Russian arsenals, rendering surplus vast amounts of Pu and highly enriched uranium (HEU) from military inventories. While, in the past, efforts have been made to draw a sharp distinction between civil and military nuclear programmes, one disposition option under consideration for surplus military Pu is to consume it in nuclear power reactors. This option is seen by some as the only proven
technical scheme to eliminate the surplus Pu, leaving spent fuel which is not substantially different from that normally produced in connection with nuclear power generation.

The civil use of Pu is made complex by the following factors: Pu is a natural by-product of nuclear energy production using U fuels; it is a concentrated energy source that is already displacing the need for U fuels in some applications; and when it is removed from spent fuel through reprocessing, the resulting waste is reduced in toxicity and volume. But Pu is highly toxic and there is no means to render it impossible to use in nuclear explosives.

States have differing perceptions of the balance of opportunities associated with the civil use of Pu and related proliferation, terrorism and safety risks. The IAEA is an intergovernmental organization created to serve the interests of its Member States. As IAEA Member States hold diverse views on the civil use of Pu, the IAEA Secretariat cannot take sides in the debate. Its policy-making organs, the Board of Governors and the General Conference, have not developed, so far, a policy with respect to the various alternatives.

While a range of views remains, all would agree that Pu and HEU must be subject to the most stringent controls to eliminate any plausible risks of proliferation and terrorism or safety hazards, and that any civil use of these materials must be carried out in a prudent and responsible manner. In the face of these conflicting views and the different perceptions of risk associated with civil use of Pu, the time seems opportune to consider establishing a system of confidence building measures that would complement the existing arrangements as well as those that will hopefully emerge in the future, following initiatives that are under way. The confidence building measures would seek to minimize the causes for concern that might arise in conjunction with prudent and responsible civil plans involving the production, storage, transport, processing and use or disposal of Pu and HEU. They would address all Pu and HEU, except that committed to defence programmes.

While issues of civil and military use of HEU are also converging, the situation is somewhat different from that for Pu. HEU can be blended down to low enrichment levels and at that point it becomes indistinguishable from low enriched uranium (LEU) used in civil nuclear power applications.

The debate over military and civil uses of Pu encompasses the following considerations.

1.1. Military use of Pu and HEU

In September 1993, the USA announced two initiatives that will have a fundamental impact on the production of Pu and HEU for nuclear weapons use, and on the disposition of military Pu and HEU rendered surplus as a result of arms reductions [1]. Under its first initiative, the USA announced that it will unilaterally submit surplus Pu and HEU from military stockpiles to IAEA safeguards. The first materials
will be in bulk form, but later, expecting that suitable verification provisions can be implemented that do not risk the disclosure of sensitive information, the USA will submit to IAEA safeguards warhead components in sealed containers.

The unilateral submission by the USA of Pu and HEU from military inventories will begin during 1994. According to an agreement announced by Presidents Yeltsin and Clinton, Russia will consider steps to ensure the transparency and irreversibility of the process of the reduction of nuclear weapons, including the possibility of putting a portion of its fissionable material under IAEA safeguards. These initiatives will hopefully be followed by similar steps by other nuclear weapon States, and by States having Pu or HEU which is not currently subject to IAEA safeguards.

In its second initiative, the USA announced its intention to establish a treaty banning the production of Pu or HEU for use in nuclear weapons or in other nuclear explosive devices, such that any subsequent production would be for peaceful purposes and would be carried out under IAEA safeguards. It is assumed that all nuclear weapon States and all other States producing and/or using Pu and HEU would accede to such a treaty, submitting considerable portions of their Pu and HEU stocks to IAEA safeguards. Negotiations on a production cut-off convention will require a considerable time to complete, because acceptance by the five nuclear weapon States party to the NPT, and by other States currently producing Pu or HEU which is not subject to IAEA safeguards, will be a precondition. The verification measures adopted in the context of that convention should complement the measures foreseen for civil nuclear programmes and for surplus military inventories.

1.2. Use of civil Pu in nuclear weapons

The suitability of Pu created in civil nuclear programmes for use by a proliferating State or a terrorist group in nuclear explosives represents the most critical challenge to reprocessing and to the use of Pu as a fuel in nuclear power reactors. Weapons designers from the USA, the UK and Russia agree that, in theory, a crude nuclear explosive can be produced with the implosion design developed during World War II, which would provide an explosive force equal to a few kilotonnes of TNT. Furthermore, by using more sophisticated designs, high explosive yields could be attained [1,2].

Two complications make the use of Pu from power reactors in weapons increasingly difficult as burnup increases. The first complication is the need to avoid a premature initiation of fission that could be caused by neutrons emitted in the spontaneous fission of the even-numbered isotopes of Pu. The second complication may be more severe than the first. Pu decays primarily through alpha particle emission, producing heat, and special measures would be required to avoid adverse effects [3].

These problems may be within the competence of nuclear weapon States which have extensive modelling capabilities and have tested different designs of warheads,
and which have developed the capabilities for all aspects of nuclear weapons manufacture. Designers are less confident that reliable weapons could be made without testing, and for threshold States and terrorist groups these limitations could spell the difference between success and failure. Some weapons designers and most of those who would advocate the peaceful use of Pu believe that all the complications facing a State establishing a nuclear weapons capability make it very unlikely that high burnup Pu could be used in any nuclear explosives, without testing. The possibility to assemble and maintain a skilled team for the required tasks is considered to be substantially less likely for a sub-national terrorist organization than for a State, if such tasks could be performed at all [2].

Some States maintain that the proliferation, terrorism and safety risks do not justify the civil use of Pu or HEU, and that there is no economic or energy security argument that would compel any State to expand the civil use of Pu in the short term.

There may be no fully effective means to stop a State intent on acquiring nuclear weapons from finding some means to obtain separated Pu. The different means of restricting access to reprocessing technology (e.g. the guidelines of the Nuclear Suppliers Group and the trigger list of the Zangger Committee) or the national restrictions on exports of sensitive nuclear technologies should force any such State to develop indigenous capabilities, rather than find ready short cuts. The risk to a State of detection of such activities through intelligence or safeguards, for example, and the anticipated severity of political and economic reactions should serve to deter such ambitions.

1.3. Civil use of Pu

States favouring the civil use of Pu maintain that these risks are exaggerated and that, through national and international measures currently in place, civil use of Pu is prudent and safe. States (and non-governmental organizations) favouring the civil use of Pu indicate that the costs currently experienced are greater than they will be as the scale of the industry expands and the regulatory environment stabilizes. Some utilities currently using Pu in light water reactors claim that the economics already warrant further use. States favouring the use of Pu are interested in extracting the full energy potential of U and in reducing the toxicity and volume of the nuclear waste resulting from nuclear electricity generation. Regarding the disposition of surplus military Pu, those in favour of the peaceful use of Pu argue that the use of Pu in nuclear power reactors is the only proven method available now to begin to eliminate these inventories and to provide a basis for balancing future civil Pu production and consumption. While some new concepts offer interesting prospects for simplifications in reactor safety, reduced costs, and the elimination of most Pu isotopes and actinides responsible for the long term hazards of nuclear waste, these concepts will not be available for at least 15 years.
The expansion of civil use of Pu is fact. The IAEA Secretariat believes that further steps to improve the transparency of prudent and legitimate peaceful nuclear programmes would enhance understanding and thereby reduce exaggerated perceptions of the risks associated with such programmes.

2. A SYSTEM OF CONFIDENCE BUILDING MEASURES

The following confidence building measures have been suggested for consideration. They include principles to guide the production, transport, storage, processing and use or disposal of Pu and HEU, and transparency provisions to ensure that the actions taken are peaceful, secure and safe.

Most of the measures identified relate to non-proliferation, but there is no intention to cast doubt on the effectiveness of the IAEA safeguards system, and this initiative is not intended to strengthen safeguards per se. It is rather the international arrangements within which IAEA safeguards are applied that should be strengthened, as well as physical protection and nuclear safety to shape future programmes for the storage, peaceful use or disposal of Pu or HEU.

2.1. Confidence building measures related to arms reductions

(a) Nuclear weapon States could determine the minimum inventories of Pu and HEU required to support essential defence requirements and could submit all excess Pu and HEU to IAEA safeguards.

(b) As soon as is practicable, all Pu and HEU rendered surplus could be converted to forms bearing no sensitive information in composition or configuration. Forms bearing such information could be submitted to IAEA safeguards under arrangements that would preclude IAEA access to this information.

2.2. Confidence building measures related to non-proliferation

(a) States could submit all Pu and HEU in civil activities to IAEA safeguards. (In non-nuclear-weapon States subject to comprehensive safeguards agreements with the IAEA, all nuclear materials (U, Pu and Th) are subject to IAEA safeguards, and thus there would be no additional obligation for these States. All nuclear weapon States party to the NPT have concluded voluntary safeguards agreements with the IAEA, but there are significant inventories of Pu and HEU in civil activities which have not been made subject to these agreements. Also, these agreements currently allow these States to withdraw nuclear materials or facilities from safeguards at their wish; clearly, this is not consistent with the present intention. Finally, several non-NPT States maintain
inventories of Pu or HEU which are not subject to IAEA safeguards, some of which is used in civil nuclear programmes. Pu or HEU used in civil programmes in these States might also be submitted to IAEA safeguards under the confidence building measures suggested.)

(b) Provisions could be adopted to ensure that the schedules for reprocessing Pu in non-nuclear-weapon States conform to firm commitments for the use of the separated Pu, so as to minimize surplus inventories. Similarly, for reprocessing carried out in nuclear weapon States for non-nuclear-weapon State customers, provisions could be adopted to return the Pu in the form of finished fuel items, or to schedule the returns of separated Pu according to the manufacturing schedules for committed peaceful projects.

c) A framework could be provided for States wishing to begin reprocessing operations and use of Pu. States could agree to deny assistance to other States in the production or acquisition of Pu except under the agreed provisions.

d) Imports and exports of Pu or HEU could be subject to specified conditions.

e) Steps could be taken to ban any production of HEU, and existing stocks of HEU could be diluted to less than 20%.

(f) Use of Pu could be limited explicitly to nuclear power reactors for the production of electricity, district heat or desalinization, or to research reactors for the development of Pu fuelled power reactors. Specific projects could be vetted through a peer review process.

(g) States wishing to dispose of their Pu could mix it with fission products and store it in vitrified form in geological repositories, under IAEA inspection.

(h) Pu and HEU could be stored in depositories under specified supervision arrangements, so that they could only be removed from these depositories when they are actually needed for an authorized project. Such depositories could be established within each State under the legal system of the State, with appropriate international custody or supervision. They could be sited in close proximity to major nuclear establishments in order to minimize the amounts of Pu or HEU needed outside of the depositories. A ‘two-key’ type of physical control system could be employed to inhibit unauthorized movement of materials deposited; the stored Pu and HEU could only be removed from the depository when a specified representative is present. The provisions could differ according to the specific nature of the material and the locations of the depositories in which they are stored. It would also be possible for States to deposit their Pu or HEU in depositories in other States.

(i) A register for Pu and HEU could be established, which could include facilities in which separated Pu or HEU would be stored, processed or used. The register could be published periodically, providing information on relevant aspects of nuclear programmes under way in each State, and on the amounts of Pu and HEU in storage, processing or use. Domestic and/or international transport vehicles could also be registered during periods of use.
(j) As distinct from information required under IAEA Safeguards Agreements, advance information and progress reports could be provided, giving, for example, descriptions of programmes being planned or under way, Pu and HEU inventories, schedules for processing and for use, and rates of production and consumption. States could agree to provide public reports according to agreed guidelines, and to provide input for comprehensive reports of Pu and HEU. The IAEA might be asked to compare the information provided with that obtained through its various programmes, and to publish comprehensive reports on Pu and HEU inventories and use, for example.

(k) Co-operation among States could be encouraged to improve transparency in the specific features of nuclear programmes involving Pu and HEU. Arrangements could include: training and staff exchange; exchange of technical information regarding safety and security measures; development of equipment and procedural standards; research and development; and joint fuel cycle and nuclear power projects.

2.3. Physical protection confidence building measures

(a) States could agree to adhere to the Physical Protection Convention and to implement the provisions of INFCIRC/225/Rev. 3.

(b) Specific physical protection guidelines could be promulgated for Pu and HEU for fixed sites (including depositories) and for domestic and international transport.

(c) States could agree to submit physical security plans for peer group reviews, and security specialists could confirm that the provisions applied are adequate.

2.4. Nuclear safety confidence building measures

(a) An extensive body of national and international nuclear safety guidelines and codes of practice already exists. Moreover, the IAEA carries out examinations of operating plants to review their safety. States could take steps to consolidate recent practice and to update guides in order to ensure that all States having activities involving Pu or HEU have the information required to execute those activities safely.

(b) States could agree to adhere to the impending Nuclear Safety Convention and to additional relevant radiological protection requirements for Pu and HEU at fixed sites and in transit.

(c) States could agree to submit safety plans for peer group review, and safety specialists could confirm that the provisions implemented are adequate.
3. IMPLEMENTATION ARRANGEMENTS

A variety of alternative arrangements have been considered through which a system of confidence building measures could be adopted. The only essential requirement is that, whatever form of collective union is adopted, it must have sufficient prestige to ensure that dubious projects are discouraged, marginal projects are upgraded, and projects conforming to the requisite standards can proceed with the endorsement of the collective forum.

The adoption of any collective mechanism will reflect a balance between the sovereign rights of States to engage in actions, as they deem appropriate, and the common good of the world community. States are naturally reluctant to surrender their sovereignty until the case is well made, and then only to the minimum extent necessary to meet the agreed conditions.

To the extent possible, the confidence building measures should be implemented within a single framework. That would provide for universal applicability and give a single focus to ensure that all agreed measures are applied with equal effectiveness and efficiency. But it may be more practical, for example, to establish separate arrangements for Pu and HEU, given the differences noted. Moreover, in terms of commitments made under the NPT and the restrictions on commerce adopted through the Nuclear Suppliers Group, for example, it may be simpler to begin with the NPT.

4. CONCLUSION

In the light of recent steps by the industry to expand the peaceful use of Pu and in view of recent progress in the elimination of nuclear weapons, certain deficiencies in the existing non-proliferation, physical protection and nuclear safety arrangements are evident. By adopting additional confidence building measures, such as those suggested in this paper, the actual risks arising from the production, storage, transportation, processing and peaceful use of Pu or HEU may be minimized, and the true risks may be perceived more accurately. As a result, public and governmental acceptance will be enhanced, creating a more predictable planning environment for future energy programmes. Also, the confidence building measures will complement steps taken towards the storage, use or disposal of Pu and HEU from military programmes under international auspices.

The creation and value for civil use of HEU and Pu are fundamentally different. There is a need to establish confidence building measures related to HEU for the inventories arising from defunct or decreasing military programmes, but there is no viable need for HEU to be used for nuclear power generation, and HEU can be diluted to LEU. Thus, the confidence building measures for HEU need to focus on storage and blending to LEU, and on the provisions required to support other remaining uses until those uses can be phased out.
The situation for Pu is much more complex. States hold different opinions regarding the balance between the perceived benefits of reprocessing and the perceived proliferation, security and safety risks. The IAEA takes no position with regard to this balance, but it must react to the reality of the nuclear programmes of its Member States. Clearly, the linkage between civil reprocessing and proliferation is debatable, and IAEA safeguards and national technical means provide the ability to detect possible abuses and enable the international community to respond in a very pronounced manner, should the need arise.

The confidence building measures suggested by the IAEA would address Pu and HEU originating from either civil or military programmes. These measures would encompass future production, storage, transportation, processing and peaceful use (or disposal, if chosen). They would attempt to ensure that any such activity would be prudent and responsible, and would be carried out in such a manner as to ensure that all such activities are peaceful, secure and safe. They would address non-proliferation, physical protection and nuclear safety.

In December 1992 and again in November 1993, the Director General of the IAEA invited representatives of the five NPT nuclear weapon States as well as Germany and Japan to consider the emerging issues and suggested a range of confidence building measures that might be adopted. At the November 1993 meeting, the States agreed to consider measures that could be supported through a multilateral framework.

The first meeting of this group of States, including also Belgium and Switzerland, took place in Vienna on 26 February 1994, at the invitation of the United Kingdom. It was agreed that this group would convene regularly during the periods of meetings of the IAEA Board of Governors, and would work towards attaining an agreed framework by the time of the NPT extension conference.

REFERENCES


Invited Paper

IAEA SAFEGUARDS EXPERIENCE

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Abstract

IAEA SAFEGUARDS EXPERIENCE.

The paper presents an overview of inspection work performed by the IAEA Divisions of Operations in the Department of Safeguards. The general increase in the workload from 1986 to 1992 in terms of countries, facilities and nuclear material is shown in a general 'zero growth' budget situation. Staffing levels have remained stable over these years, and certain problems are beginning to show. Changes redefining the role of IAEA safeguards are being discussed, some of which may provide a potential to reallocate some resources, but many others will create new demands on safeguards operations with regard to capacity, new skills and training, reflecting adaptation to new requirements from a re-interpretation of international safeguards. With the investigation of new requirements it will be inevitable also to re-examine the resources necessary to enable IAEA safeguards to fulfil the changing or rather increasing expectations.

1. INTRODUCTION

This paper provides an overview of IAEA safeguards implementation from the viewpoint of the IAEA inspectorate for the years 1986-1992. It attempts to provide continuity to previous reports [1, 2] delivered to past IAEA safeguards symposia. While it reports some details on work accomplished during those years, it also addresses the problems that had to be faced at the time. It will be remembered that, at a time of 'zero growth' safeguards budgets, the Department of Safeguards was confronted with an entirely new scenario of events.

Expected changes with regard to new Member States, new safeguards arrangements, additional facilities and nuclear material will be discussed, as well as possible changes in the role of the Department, particularly safeguards operations, as recently discussed during meetings of the IAEA's policy-making organs, and the conditions under which these may be implemented from the point of view of safeguards operations.

The paper does not address specific future requirements in respect of equipment that will foreseeably result from a changing of activities. Current progress in
equipment management and instrument development are reported in other sessions of this symposium.

2. IAEA SAFEGUARDS

The Divisions of Operations have been confronted with a number of new challenges over the past few years. The number of countries and facilities and the nuclear material under international safeguards increased significantly, and new types of facilities required the devising of new effective and efficient safeguards approaches. This had to be accomplished under severe constraints in the IAEA’s budget.

In 1991, the United Nations Security Council passed Resolution 687 (and a number of subsequent resolutions) on Iraq, which placed an additional and new demand on the IAEA Divisions of Operations’ resources and subsumed all other safeguards activities in that country. For the first time a Member State with declared nuclear facilities (which had been regularly inspected by the IAEA under a Safeguards Agreement) had clandestinely established additional nuclear facilities and had begun to produce nuclear material in violation of the Agreement. Other papers and presentations provide more detail of this situation. It must be noted that the basic staffing of all inspection missions to Iraq has come from the Department of Safeguards, mainly from the Divisions of Operations, with specialized experts from Member States1 [3].

Another new situation occurred in 1991 with a Member State accepting a fully comprehensive Safeguards Agreement. Member States required special treatment of this situation to ensure the completeness of the list of facilities and material declared and verified by the IAEA’s inspectorate2 [4–6]. During the IAEA’s verification activities on the correctness and completeness of the initial report, the State President declared on 24 March 1993 that in the past his country had followed a path that eventually led to the possession of nuclear weapons. Although they had been dismantled and destroyed prior to accepting the new safeguards arrangements, additional work ensued, to ascertain the rendering useless of all components from the former weapons programme.

The request to clarify differences between the State’s initial report pursuant to the entry into force of a fully comprehensive Safeguards Agreement and results obtained during an inspection (correctness and completeness assessment), and to ask

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1 Twenty-one inspections in Iraq had taken place by the end of 1993; results of all inspections have been reported to the IAEA Board of Governors and are available as GOV documents.

2 The Secretariat has provided a number of status reports to the IAEA Board of Governors on South Africa, the most recent one for the September 1993 Board of Governors Meeting (GOV/2684) [7].
for a special inspection at two additional sites, led the Government of the Democratic People’s Republic of Korea (DPRK) to announce withdrawal from the NPT and later to suspend ‘the effectuation’ of the withdrawal. Details have been discussed at meetings of the IAEA Board of Governors many times, and, more recently, also at the IAEA General Conference\(^3\) [8, 9]. Such details have also been given much attention in the media. Though the prevailing problem is widely considered to require some progress at the political level, it also requires the IAEA inspectorate to provide all technical support, so as to be able to resume safeguards implementation at any moment.

2.1. Countries, facilities, and material under IAEA safeguards

The number of Member States with Safeguards Agreements has increased from 96 in 1986 to 111 in 1992. The number of States with significant nuclear activities, i.e. more than one significant quantity (SQ) of nuclear material, rose from 58 (1986) to 68 (1992). In 1992 the IAEA had 188 Safeguards Agreements in force (1986: 164, see Fig. 1), an increase by about 15%.

The number of nuclear facilities grew from 494 (1986) to 521 in 1990, then fell again to 493 (1992). Likewise, the number of locations to be inspected increased from 910 in 1986 to 938 in 1990, then decreased to 814 in 1992 (see Fig. 2).

Another important parameter for the workload of safeguards operations is the amount of nuclear material under IAEA safeguards. The total number of SQs almost doubled from 37 890 (1986) to 65 878 in 1992 (1981: 16 723 SQs). Most of these

\(^3\) ‘Separated plutonium’ includes recycled plutonium in fuel elements until their discharge from the reactor.
Sqs consist of plutonium, the amount of which also almost doubled from 1986: 202.9 t to 403.8 t in 1992 (1981: 80 t). This refers to the total amount of plutonium (contained in irradiated fuel and as separated plutonium). The amount of separated plutonium is only a fraction of the total amount; in 1992, 35.3 t of unirradiated plutonium outside reactor cores and 2.3 t of recycled plutonium in fuel elements in reactor cores [8, 9] were under IAEA safeguards (1986: 8.4 t, or an increase of 4.5 times).

The amount of highly enriched uranium (HEU, 20% of $^{235}$U) decreased from 13.2 t in 1996 to 10.9 t in 1992 (1981: 10 t). This trend may well reverse in the future, should HEU from former weapons programmes be placed under IAEA safeguards (see also Section 4.2). Low enriched uranium (LEU, <20% of $^{235}$U) under safeguards has grown from 27 911 t in 1986 to 35 835 t in 1992 (1981: 16 000 t), and other source material [10] from 47 402 t (1986) to 77 958 t (1992). Figures 3 and 4 show the quantities of nuclear material under IAEA safeguards from 1986 to 1992.

The considerable increase in nuclear material under safeguards results from the doubling of total plutonium (over +99% increase, constituting 77% of all Sqs) and source material (+64%, 8% of all Sqs), less from LEU (+28%, 15% of all Sqs), and slightly set off by the decrease in the amount of HEU (−17%, but only less than 0.5% of all Sqs).

2.2. The Safeguards Statement

As in the past, the main result of the IAEA safeguards implementation activities is expressed in the Annual Report and the Safeguards Implementation Report (SIR) as the Safeguards Statement: “In carrying out safeguards obligations... On the
basis of all information available to the Agency, it is considered reasonable to con­
clude that nuclear material and other items which had been placed under Agency
safeguards remained in peaceful nuclear activities or were otherwise adequately
accounted for" [11]. This statement must be seen in the light of observations con­
cerning information on safeguards activities carried out, the sensitivity of inspection
and evaluation activities based on defined criteria, the level of assurance associated
with the Secretariat’s findings and the conclusions of the SIR. This resulting state­
ment arises from both the efforts of the Department of Safeguards in application of
its implementation and evaluation criteria, and the strong support and close co­
operation with Member States.

The most recent SIR (for 1992) refers to the verification of the correctness and
the assessment of the completeness of State declarations. It also refers to evidence
that “owing to limitations in the information available to the Agency ... , non­
compliance with safeguards agreements could occur without detection by the Agency
in the case of undeclared activities.”

Several measures were discussed and some approved by the IAEA Board of
Governors for implementation with a view to improving this situation and strength­
ening the safeguards system, inter alia the early provision of design information,
special inspections, the universal reporting of exports, imports and the production
of nuclear material for peaceful purposes, and the universal reporting of exports
and imports of certain equipment and non-nuclear material for peaceful
purposes [12, 13].

2.3. Inspection effort

Staff resources for safeguards inspections rose from 192 person-years in 1986
to 200 in 1992 (+4%). The Department as a whole contributed by providing support
concerning development, purchase and maintenance of equipment, development of
better standards and procedures, processing and analysis of computerized informa­
tion, training, evaluation and administration, and in a number of cases also by
providing routine and special support in the field.

The available resources for inspections are matched with the requirements of
person-days of inspection (PDI) for all inspections. Necessary training prior to the
first inspection assignments for inspectors and inspection assistants must be taken
into account. Also designation procedures and resulting delays are included.

The total person-days of inspection increased from 8292 (1986) to 10 381 in
1990 and fell back to 8385 PDI in 1992. The number of inspections performed was
2054 in 1986, rose to 2196 in 1989, and dropped back to 2047 in 1992 (Fig. 5).

At the same time, a number of efficiency steps were taken: the use of surveil­
lance increased substantially, the number of films reviewed by inspectors increased
by 68% (from 1946 in 1986 to 3260 in 1992). In addition, beginning in 1989, the
use of video surveillance was established. In 1992, 2060 video surveillance films
were reviewed. Nuclear material sampling is a well established safeguards technique. In 1986, 1080 such samples were taken, in 1992: 1439 (+40%, Fig. 6). 3240 samples were analysed and reported in 1992 (2837 in 1986), a 14% increase.

2.4. Inspection goal attainment

Goal attainment\(^4\) in major facilities increased from 63.4% in 1986 to 80.6% in 1990 and then dropped back to 68.8% in 1992 (Fig. 5). Of 138 light water reactors inspected and evaluated, only 14 did not attain the inspection goal, for on-load reactors four out of 15, for research reactors and critical assemblies three out of 40, and for facilities other than reactors three out of 95 failed to attain the inspection goal in 1992.

3. NEW FACILITIES

For the forthcoming years a further growth in both number of facilities and amount of nuclear material must be expected. About 40 additional power reactors will begin operating under safeguards before the end of 1996, and other nuclear installations, including reprocessing and enrichment plants, will come under IAEA safeguards during that time. It is reprocessing and enrichment technology that will place a significant additional demand on the IAEA's inspectorate.

Nuclear material will also increase. Estimates until 1999 indicate an approximate +60% for plutonium, +40% for LEU, and +35% for source material. No increase is expected for HEU, unless material from former weapons programmes is

\(^4\) As defined in Ref. [11].
placed under IAEA safeguards. This, however, would also significantly change the estimates for additional plutonium, not only in quantity, but possibly also as regards the composition and material specification for the plutonium.

4. COSTS AND RESOURCES

Almost all safeguards operations workload parameters indicate growth (Figs 1–4), sometimes at a significant level, but expenditure, in absolute terms, for safeguards implementation has shown a modest or no increase (Fig. 7). Compared with resources available in 1985, however, an overall change of +3% (in 1992), coupled with negative relative growth in 1985, 1986 and 1991, can be seen as having depleted all internal reserves of the organization. The continued application of such constraints by IAEA Member States (since 1989) must now result in reduced performance, both in efficiency (PDI) and effectiveness (goal attainment). An illustration of this can be found in Fig. 8.

5. FUTURE CHALLENGES AND OPPORTUNITIES

The development of a more systematic and consequent methodology of safeguards implementation, as defined in the 1991–1995 Safeguards Implementation and Evaluation Criteria, and their rigorous implementation, have contributed to a relaxation of the otherwise more visible constraints. This had been done without
compromise regarding the safeguards goals. Any changes to these criteria as required by strengthening measures must take into account future resource constraints.

Other ways to reduce the effects of a budget constraining implementation have been examined. New models of co-operation and co-ordination with State systems of accounting and control (SSACs) of nuclear material in Member States or regional organizations are being explored. One example is the new partnership relationship with the Euratom organization.

Ways and means must be found to set off foreseeable additional costs generated from new methods, techniques and procedures to provide better assurance against undeclared facilities or undeclared production of special nuclear material at safeguarded facilities under comprehensive Safeguards Agreements. Proposals were made by the Special Advisory Group on Safeguards Implementation (SAGSI) [14].

5.1. Co-operation arrangements

The co-operation between the IAEA and Euratom has made it possible to implement the New Partnership Approach, aiming at the optimization of all practical arrangements, the use of commonly agreed safeguards approaches, and inspection planning, procedures, activities, instruments, methods and techniques. To facilitate implementation of this approach a number of technical measures are being developed. In addition to these arrangements, of which details are presented elsewhere in these Proceedings, possibilities to enter into similar or different arrangements with other SSACs will be considered. It is understood that the IAEA must fulfil the prime obligation for independent verification of facilities' design characteristics, processes within the facilities and all nuclear material.

An example of a new organization, set up by Argentina and Brazil, for their bilateral efforts with respect to nuclear material accountancy and verification, a common SSAC, is the Agencia Brasileño-Argentina de Contabilidad y Control de Materiales Nucleares (ABACC) with headquarters in Rio de Janeiro, Brazil. Based on the Quadripartite Agreement [15] between Argentina, Brazil, ABACC and the IAEA, certain activities will be co-ordinated between the IAEA and ABACC, as defined in the existing draft Subsidiary Arrangements. These arrangements will enter into force with the Agreement.

Dependent on common interests from participating Member States, the nature of Safeguards Agreements with the IAEA, and the IAEA's own capabilities, other or similar arrangements, where countries establish a joint SSAC, could develop.

5.2. Nuclear material from former weapons programmes

Recent developments in nuclear disarmament in the United States of America and Russia raise the possibility of large amounts of special nuclear material formerly
used in weapons components returning into the civilian nuclear cycle of these States. President Clinton has announced that the USA intends to place special fissile material no longer required for US national defence under IAEA safeguards\(^5\). Other nuclear weapon States may follow.

The IAEA has experience with the verification of large amounts of unirradiated direct-use material, mainly separated plutonium and, to a lesser degree, highly enriched uranium. Relevant routine verification activities permit the statement that the IAEA safeguards inspectorate has the skills and possesses the necessary measurement equipment to perform verification activities with respect to the transfer of nuclear material from the military sector to peaceful use. For the safeguarding of material still in the form of weapons components, new techniques may have to be defined to protect information on the design and composition of the components.

### 5.3. Multilateral cut-off of special nuclear material production

Verified termination of production of plutonium and highly enriched uranium for weapons or other explosive purposes has been under discussion for some time. The present international climate could lead to concrete proposals for an agreement. The premise would be that there was already more than enough weapons-usable material and no need for further production of such material. Methods and techniques for verification exist, but might need to be further developed.

The statement by President Clinton in September 1993 includes such a proposal for a production cut-off. It would be too speculative to discuss a time frame for implementation and volume of work for safeguards operations involved.

### 5.4. Newly independent States

All States of the former Soviet Union, with the exception of the Russian Federation, have declared their intention either to become or remain non-nuclear-weapon States. At the end of 1992, eight of the newly independent States (NIS) were parties to the NPT (Armenia, Azerbaijan, Belarus, Estonia, Latvia, Lithuania, the Russian Federation and Uzbekistan). Additionally, according to the terms of the Lisbon Protocol of 23 May 1992, and recent developments (meetings with US President Clinton in January 1994), Kazakhstan and Ukraine are expected to follow soon.

The scope of the nuclear programmes in the NIS is substantial. Future safeguards are expected to include the world's largest fuel fabrication plant, a number of large on-load (RBMK type) facilities, a fast breeder, and various types of unique research reactors. This will amount to about 50 new installations. The IAEA

\(^5\) Statement by US President Clinton at the United Nations General Assembly in September 1993. See also statement of the US Secretary of Energy, H. O'Leary, at the 37th General Conference of the IAEA.
has initiated a series of activities to prepare the groundwork for the application of safeguards in these States. Upon their adherence to the NPT, major inspection work is expected, requiring a significant amount of IAEA effort and resources.

6. CONCLUSIONS

Observations over the past years indicate that safeguards goal attainment, as one measure for safeguards performance, is decreasing. This is happening in step with other parameters influencing the overall performance, e.g. number of inspectors and inspections, person-days in the field, and similar such parameters. These, again, move in step with the funds expendable for safeguards work.

The amount of nuclear material to be placed under IAEA safeguards must be expected to continue to grow. Member States join comprehensive Safeguards Agreements under the NPT or the Tlatelolco Treaty, countries from the former Soviet Union with significant nuclear programmes have joined the IAEA and entered into arrangements that will soon lead to such agreements, and material from nuclear disarmament efforts will soon be placed under IAEA safeguards.

New responsibilities will add significantly to the future workload. Safeguards will continue to rationalize inspection work; advances in equipment technology, modified methods and procedures for verification and evaluation will be examined and implemented, provided they are effective, feasible and cost saving. Prudence demands careful assessment of all changes.

However, there can be little doubt that a substantial increase of assurance against nuclear proliferation will also require an increase in verification resources. This ultimately means that Member States must be willing and ready to provide adequate funds for future and better safeguards implementation.

REFERENCES


NATIONAL AND REGIONAL SYSTEMS FOR ACCOUNTING AND CONTROL OF NUCLEAR MATERIAL

(Session 2)

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THE ROLE OF EURATOM IN INTERNATIONAL SAFEGUARDS

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Abstract

THE ROLE OF EURATOM IN INTERNATIONAL SAFEGUARDS.

International developments in the 1990s required a reassessment of the elements forming international safeguards. A consensus is emerging that in future these should go beyond the classical measures of the past. There should be a re-evaluation of bilateral and multilateral obligations. The mandate of Euratom safeguards is established in the Euratom Treaty. Derived from it are co-operation with the support programmes for IAEA safeguards, co-operation with third countries, and with States and organizations of the former Soviet Union. In the 1960s Euratom concentrated more on technical issues than on political ones. The relationship with the IAEA developed over the years, and in 1992 the New Partnership Approach (NPA) was initiated. The NPA aims at a modus vivendi which will allow the IAEA to reduce its inspection effort in the non-nuclear-weapon States of the European Community to a level lower than in places where such a regional system as Euratom's does not exist. The paper addresses further technical and political issues currently under discussion, such as the strengthening of IAEA safeguards, decentralization of safeguards measures and the challenges posed by safeguarding the use of plutonium in peaceful uses of atomic energy, including plutonium and highly active uranium transferred in the course of nuclear disarmament. Finally, the paper discusses the relationship between the envisaged strengthening of the NPT regime, including the extension of the NPT in 1995, and the problem of strengthening bilateral controls and agreements.

1. INTRODUCTION

International safeguards conventionally denote the set of measures performed to verify that nuclear material and equipment is not diverted from peaceful nuclear activities to non-peaceful ones. The non-proliferation regime, created in essence in the early 1950s, has undergone several changes in its emphasis, scope and coverage but not in its basic objectives. It has always been emphasized that verification performed by an international inspectorate of the non-proliferation obligations assumed by the parties to the various Safeguards Agreements was an essential element of the entire regime.
Following the conclusion of the NPT the discussion on the mode of operation of IAEA safeguards continued intensely but was restricted to the safeguards environment. The discussions concentrated on implementation issues such as access possibilities of IAEA inspections, resource allocation or, more generally, on safeguards effectiveness issues. Triggered by events in Iraq, South Africa and the Democratic People’s Republic of Korea (DPRK), by the dissolution of the bipolar strategic equilibrium with all its consequences, by the forthcoming review and extension conference of the NPT and by other developments, nuclear non-proliferation has attracted full attention, and an essentially political discussion has started in a climate that is more favourable than before regarding non-proliferation and nuclear disarmament.

In this process a consensus appears to be emerging that international safeguards should comprise responsibilities or operations in fields beyond the classical measures envisaged under INFCIRC/153 and INFCIRC/66. These may also comprise the re-evaluation of bilateral/multilateral obligations, export controls of other than nuclear materials and/or equipment, and nuclear materials transferred from the non-civil cycle during nuclear disarmament. The collaboration with and the drawing of maximum use from regional safeguards systems are important themes in this political discussion.

The Euratom nuclear safeguards system was established by the Euratom Treaty signed in Rome on 25 March 1957. Through this Euratom Treaty (Chapter VII: Safeguards) the Member States of the European Community — now the European Union — have allocated to the European Commission extensive executive and some regulatory powers which it exercises, in its own right, in those areas for which it is competent and which may take effect directly within the legal systems of its Member States. Through this delegation of safeguards responsibilities to the multinational Euratom system, each Member State could largely relinquish the need to establish its own State System of Accounting and Control (SSAC) of nuclear materials as would otherwise be required, for example, under the NPT.

The above mandate of Euratom does, however, not comprise the responsibility for the non-proliferation policy, which remains essentially the responsibility of the Member States of the European Union (it may be noted in this context that all Member States of the European Union are parties to the NPT but not to the Community as such), co-ordinated and harmonized, however, through the European Council’s relevant procedures.

The role of Euratom in non-proliferation is to implement and to facilitate non-proliferation (IAEA) safeguards pursuant to the agreements INFCIRC/193 (NNWS), INFCIRC/263 (UK) and INFCIRC/290 (France), thereby enabling the Commission, pursuant to Article 77b of the Euratom Treaty, to “… satisfy itself that, in the territories of Member States, … any particular obligations assumed by the Community under the agreement concluded with … an international organisation are complied with”.
2. NPT SAFEGUARDS: THE FIRST TWENTY YEARS

Non-Proliferation Treaty safeguards following the model agreement INFCIRC/153 and, more specifically, the Verification Agreement INFCIRC/193 led to several significant changes to the Euratom safeguards operation under the Treaty owing to the specific obligations the Community and its Member States had assumed. This contained:

— Concentration on the verification of the accountancy of the nuclear material under safeguards;
— Positioning of containment and surveillance measures as complementary;

or, in other words, concentration on the so-called quantifiable safeguards measures.

It was, however, realized in the mid-1980s that a reorientation of the Euratom safeguards concepts, approaches and operation was necessary because of:

— The then visible arrival of the commercial use of reprocessed plutonium, i.e., large scale reprocessing plants, large scale mixed oxide (MOX) fabrication plants and MOX LWRs;
— The need to implement the Euratom Treaty without discrimination against anyone setting up or operating an installation for the production, separation or other use of source material or special fissile materials, i.e., for all civil nuclear material;
— The consequential necessity to safeguard the civil nuclear material in the so-called mixed installations in a way that there would be no net loss in quantity and quality of that civil nuclear material;
— The awareness of the dilemma between the rapid increase in the stocks of civil nuclear material, the number and complexity of the installations and of additional tasks — e.g., adherence of new members to the European Community — and the restrictions of the increase of human resources;
— The growing attention in the public domain as well as in Parliament to all nuclear questions, including nuclear safety, environmental protection and safeguards, including non-proliferation.

This reorientation included:

— Changes of the instrument development and procurement programmes; key words are modularity, off-the-shelf material, etc.;
— Decentralization of the analytical capabilities, e.g., mobile instrumentation for destructive analysis;
— Development, testing and implementation of motion detection features for video equipment to increase the detection capabilities and to render the evaluation process more efficient;
— Increasing emphasis on the verification and re-verification of basic technical characteristics to improve verification of the use of nuclear materials;
Development of safeguards approaches (including Commission approval) for large scale reprocessing facilities;

Increase of productivity by streamlining in-plant inspections, clustering of facilities and improvement of logistics;

Reassessment of the role of non-quantifiable parameters;


The impact on non-proliferation of these developments in Euratom safeguards was visible but, albeit presented at and discussed in international fora, less important than could be expected because of the particular political constraints of the IAEA.

According to Euratom's role in implementing the obligations under the NPT assumed by the Member States of the Community (Article 77b of the Euratom Treaty) and to facilitate IAEA safeguards pursuant to the Verification Agreement INFCIRC/193, to INFCIRC/263 and to INFCIRC/290, the relationship with the IAEA developed over the years steadily, but not without problems.

As the mandate of IAEA safeguards was to build confidence that States would not proliferate, the IAEA always had difficulties to appreciate fully the possibilities offered by a multi-State safeguards system, and Euratom — with its primary mandate to ensure that operators would follow the law — had difficulties to understand the way the IAEA, following pressure from the superpowers, implemented the agreements. Areas of concern were:

- A considerable duplication of inspection effort and less than optimum inspection productivity resulted even after certain 'compromises' were agreed upon;
- The IAEA spent 45–50% of its inspection effort and resources in the European Community, an effort which, when taken together with the effort spent in Canada, Japan, Scandinavia and certain States of the former East Block, left only a small portion of the resources available for countries or regions of non-proliferation concern;
- On the other hand this interaction between Euratom and the IAEA improved safeguards procedures on both sides as well, as it established the safeguards and non-proliferation credentials of the European Community as second to none.

The discussion on basic non-proliferation issues, starting with the above mentioned developments in the early 1990s, and Euratom difficulties with certain IAEA criteria, resulted in 1992 in the initiation of and agreement on a New Partnership Approach (NPA) between the IAEA and Euratom. As discussed below, this will constitute a further step toward a rational modus vivendi between the international and, so far, the only regional safeguards system.
3. NPT SAFEGUARDS: RECENT ISSUES

Following the discoveries in Iraq and South Africa, and, slightly later, in view of the problems in the DPRK, the issue of non-proliferation gained widespread public attention and a necessary political discussion started in the IAEA Board of Governors and, of course, in the safeguards community. The importance of the issues and problems was amplified in view of the forthcoming conference on the review and the extension of the NPT in 1995. This discussion focused on the role of IAEA safeguards in a changing world and how the IAEA safeguards system could be strengthened.

When considering the established and envisaged tasks of IAEA safeguards, these appear formidable:

— The INFCIRC/153 safeguards system needs strengthening to avoid further Iraq-like cases, i.e. to detect clandestine non-peaceful activities;
— The newly independent States (NIS) of the former Soviet Union need to join the safeguards regime to maintain its universality;
— If possible, those IAEA Member States not agreeing to complete and comprehensive IAEA safeguards should be motivated to do so;
— The nuclear material from the military cycle needs to be safeguarded either in the framework of the cut-off convention or in the more general framework of an enhanced implementation of IAEA safeguards in the nuclear weapon States;
— Several technical challenges such as the implementation of IAEA safeguards at large scale plutonium processing plants and the modernization of safeguards instruments, methods and techniques need to be met.

While answers to the above challenges to the IAEA need to be found, if possible contemporary with the NPT extension, these cannot be addressed in this paper, which is limited to certain issues where Euratom could probably make further contributions.

While not disputing the need to augment the IAEA safeguards resources, it is doubted that any realistic augmentation of these will suffice without a fresh look at some well-known issues.

How should IAEA safeguards resources be allocated? While there is a rational solution to this allocation problem, i.e. pro rata to the probability of diversion (or the ranking of it), this solution remains entirely theoretical and not feasible. Since no consensus on numerical values of diversion probabilities or a ranking scale of non-proliferation credentials is likely to emerge. The political doctrine of non-discrimination is (still) given priority vis-à-vis the necessities of non-proliferation. Without criteria and yardsticks for the allocation of resources:

— If the resources continue to be allocated pro rata to the amounts and/or strategic values of the nuclear material, then the portion of the resources available for
areas of concern and for new tasks must shrink — thus reducing the non-proliferation effectiveness — as long as the augmentation of resources is not related to the growth of nuclear material, an augmentation unlikely to be accepted by the IAEA Member States;

— If any other yardsticks for the allocation of resources are deployed, the IAEA may be criticized for violating the non-discrimination doctrine.

How can the effectiveness of safeguards be maintained? Under the constraints of the non-discrimination doctrine, zero growth budgets, and in view of the new tasks and responsibilities as well as the requirement to maintain at the same time the principles of INFCIRC/153 and INFCIRC/66, the IAEA — given the incompatibility of these constraints — had and has no choice but to adjust the verification effectiveness accordingly. This is being performed in the expectancy that the safeguards and non-proliferation effectiveness will be maintained through the provision of more complete, more timely and more accurate information. In other words, replacing partially the inspections through the collection and evaluation of information — activities far less costly than inspections and related logistics — achievement of the same effectiveness to detect clandestine nuclear activities is postulated. While this approach seems very worth while pursuing, it will necessarily require further transparency and assurances involving new types of safeguards techniques and/or new modes of IAEA inspection modalities.

Can decentralization of safeguards tasks and responsibilities contribute to the solution of the non-proliferation problems? Two different approaches to decentralization may be distinguished:

— Decentralization through the strengthening of bilateral controls by supplier countries;
— Decentralization through the co-operation with SSACs and/or regional safeguards systems.

Strengthening the bilateral controls of supplier States might be effective when:

— The number of such supplier States is limited;
— These supplier States co-operate in a way that non-proliferation is assigned a higher priority than market interests;
— The spread of technological knowledge can be contained.

The opinion of the European Commission services is that none of the above conditions can be met any longer — bilateral controls being essentially a pre-NPT approach to non-proliferation — and that strengthening or tightening of bilateral controls may provide disadvantages:

— States' internal legislation may lead to the subordination of international agreements under such national legislation;
— IAEA safeguards may become the tool of national policies rather than maintaining independence and impartiality;
— States outside of or critical to the NPT regime would not be encouraged to join.
In general, the strengthening of bilateral controls could render the international safeguards regime redundant or meaningless at a time when the internationalism and universality of non-proliferation safeguards need to be strengthened.

Decentralization through the co-operation with SSACs or with a regional safeguards system appears promising, but certain points may be borne in mind:

— As States (or their nuclear activities) are — pursuant to the present non-proliferation doctrine — subject to international verification measures, these SSACs cannot carry out verifications in lieu of the IAEA: he to be controlled would control himself. SSACs may facilitate IAEA verification through transparency of their nuclear programmes, through minimizing restrictions of IAEA access to and in the State/installations, through logistical support to IAEA inspections and through support programmes. The advantages for the State concerned can be expected to be less in a formal reduction of routine IAEA inspection effort — as this would contradict the non-discrimination doctrine — but in cost saving for the IAEA as such and in the interpretation of IAEA safeguards criteria toward their ‘lower boundaries’ combined with a worldwide recognition of the State’s low proliferation risk. Further trade-offs for a State, when accepting new techniques or new inspection modalities, need to be discussed.

— Regional safeguards systems can contribute significantly to the solution of the present non-proliferation problems, provided they meet the criteria of:

• Independency and multinationality;
• Constitutional framework and common commitment to non-proliferation;
• Legal framework including sanction possibilities;
• Organizational framework including independent budgets;
• Experience.

These contributions to non-proliferation include:

• All the contributions an SSAC may provide;
• The performance of particular verification operations by regional safeguards systems in lieu of the IAEA, thereby allowing the IAEA to allocate inspection effort at a level required for quality control purposes only.

The advantages, in general, for the IAEA of a close co-operation — or partnership — with a regional system of recognized credibility are of an economic but also of a non-proliferation nature, because the regional system, through its independence from single States, may share verification activities and information on the nuclear activities in its region not filtered by national interest but based on its very own and fundamental incentive (due to the law, etc.) concerning the peaceful character of the nuclear activities.
The risk, on the other hand, for IAEA safeguards and for non-proliferation in the light of decentralization of verification tasks to regional systems should, however, be evaluated also:

- The IAEA’s right to draw independent conclusions may be questioned or eroded;
- Any group of States may request such terms even if no or insufficient evidence for their credibility, i.e. their proven meeting of all the criteria, is available and the acceptance of such requests would become an issue of political opportunity.

Therefore, any recognition of the credibility and value for non-proliferation of a regional system can only take place when the above criteria for its credibility are established, discussed in an open and transparent way and decided on a case by case basis, i.e. no consent by definition or by default.

4. CONTRIBUTIONS OF EURATOM TO INTERNATIONAL SAFEGUARDS

As a spin-off from its legal obligations under the Euratom Treaty, the Euratom safeguards system meets — or rather has to meet — all the criteria for a regional safeguards system and it is, moreover, under the supervision of the democratically elected European Parliament, which adds further elements to accountability and control.

Euratom contributions to international non-proliferation safeguards may be subdivided into three parts:

- Technical contributions;
- Contributions in operations and logistics;
- Political contributions.

Technical contributions such as in R&D include activities carried out in the area of development of instruments, methods, techniques and informatics. Several papers presented by Euratom staff at this symposium are intended to show this.

It may be noted in this context that the interest of the IAEA Department of Safeguards and the Euratom Safeguards Directorate should be considered as identical, i.e. to have available reliable, accurate and easy-to-use safeguards equipment, tools and informatics at minimum cost for investment and maintenance, irrespective of where these come from.

Contributions also include the Euratom development, implementation and routine use of unattended measuring and verification stations which have both economic and safeguards advantages, as these stations offer the way to safeguard large plutonium processing plants without the excessive deployment of expensive man-
power. As a related development the work performed by Euratom for the implementation of on-site laboratories may be mentioned.

Further technical contributions are the development of and participation in the discussion of safeguards concepts and approaches. Apart from contributions to the safeguards approaches for the established fuel cycle facilities — including large reprocessing plants — contributions should be mentioned to the safeguarding of 'mixed' installations that might be relevant in the implementation of the cut-off convention or other activities aimed at transferring military nuclear material to civil use.

As regards contributions in operations and logistics, it may be recalled, for example, that since about one decade the main load for inspection logistics under INFCIRC/193 has rested with Euratom. This and the load sharing under the one-man-one-job system should be regarded as typical for the co-operation between the international inspectorate of the IAEA and the regional inspectorate of Euratom. Other examples relate to common evaluation, to the provision of Euratom inspection findings to the IAEA, to the provision of information pursuant to INFCIRC/415 and others.

Political contributions of Euratom are provided in the context of the European political co-operation, through the liaison committee with the IAEA, bilateral discussions on safeguards questions of mutual interest and, finally, through the co-operation with third States to foster the establishment of effective and high quality SSACs. Euratom also contributes to the political discussions on the issue whether, based on the Euratom model, the establishment of regional systems in other parts of the world may improve the non-proliferation regime or may reduce tensions.

Except for some elements mentioned in the area of political contributions, all the other technical and operational contributions of Euratom are fully reflected in the NPA. Apart from the expectation that the NPA will provide a balanced compromise for the operation of the Verification Agreement INFCIRC/93 combined with significant economic advantages for the IAEA, the NPA provides for the optimization of necessary practical arrangements, for the use of commonly shared analysis capabilities, for co-operation in the training of inspectors, for co-operation in R&D, including common procurement of equipment, and for co-operation in the use of new technology. The impact of this co-operation between the two safeguards inspectorates, once set up and further developed, on the further improvements of safeguards in Europe, as well as on the strengthening of IAEA safeguards worldwide, is expected to be considerable.
PRESENT STATUS OF
SAFEGUARDS IMPLEMENTATION IN JAPAN

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Abstract

PRESENT STATUS OF SAFEGUARDS IMPLEMENTATION IN JAPAN.

The fact that the research, development and utilization of nuclear energy in Japan is
strictly limited to peaceful purposes is stressed by making reference to Japan's Atomic Energy
Basic Law and the legal framework. One of the basic policies of the Atomic Energy Commiss­
on (AEC) and of the Japanese Government from the beginning has been to recycle and fully
utilize plutonium, since this material is a very precious energy source for Japan. Another
policy of the AEC and the Government is that international safeguards and the State System
of Accounting and Control (SSAC) of nuclear materials should be fully and strictly enforced.
The nuclear share of total electricity generation and the total amount of nuclear materials, as
well as the number of facilities under safeguards, are discussed. Japan's national safeguards
system is explained with reference to the role of the Nuclear Material Control Center, which
is responsible for conducting destructive and non-destructive assays, among other tasks. A
description is given of typical safeguards techniques and research and development activities,
including those which are being conducted currently on reprocessing plants. The paper also
discusses the co-operative relationship with the IAEA in the field of safeguards and inspection
activities, and research and development work designed to contribute to more efficient and
more effective safeguards. In this connection, reference is made to various extra-budgetary
contributions made by the Japanese Government. The Government is prepared to contribute
to the effort of the IAEA in streamlining the safeguards system to cope with the growing
safeguards requirements and to overcome financial constraints.

1. INTRODUCTION

More than forty years have passed since the beginning of the peaceful use of
nuclear energy. It is indeed a great joy to witness the promotion of nuclear use and
development in various countries throughout the world. It was thanks to the efforts
on the part of the IAEA, IAEA Member States and various international institutions
after the entry into effect of the nuclear Non-Proliferation Treaty that the peaceful
use of nuclear materials was assured and international confidence in the peaceful use of nuclear power was fostered.

As a nation promoting the use of nuclear energy and a Member State of the IAEA, Japan has been a promoter of strictly peaceful uses and development of nuclear energy. Japan actively accepts international safeguards as part of its nuclear non-proliferation measures, based on an agreement with the IAEA.

This paper introduces safeguards being actively carried out in the field of nuclear energy today in Japan along with the status of nuclear use and development.

2. JAPAN'S BASIC POLICIES CONCERNING SAFEGUARDS

As far as research, development and utilization of nuclear energy is concerned, the Japanese people have firmly decided that it should be strictly limited to peaceful purposes. When nuclear energy related activities were initiated in 1955, the Atomic Energy Basic Law was enacted and a specific provision was formulated, stating that "the research, development and utilization of atomic energy shall be limited to peaceful purposes".

This particular part of the Law has been kept intact for almost forty years without any amendment. Another important policy that should be stressed on this occasion concerns the fuel cycle policy of the Atomic Energy Commission (AEC) and of the Government. The AEC was created in January 1956. One of the basic policies of the AEC from the beginning has been to recycle and fully utilize plutonium that has been produced as a result of irradiation in a reactor core. Plutonium is considered to be a very precious energy source for Japan, where resources are extremely scarce. The annual report of the AEC for 1993, which was made public recently, clearly describes this and reads: "Japan has kept consistently from the outset a basic policy that the spent fuel should be reprocessed and plutonium and uranium thus recovered should be recycled as nuclear fuel".

A third policy of the AEC as well as of the Government is that the international safeguards and the State System of Accounting and Control (SSAC) of nuclear materials according to the full scope safeguards system under the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) should be fully and strictly enforced and implemented by the Government agencies and by facility operators alike, whether they be Government laboratories, public corporations, universities, utilities or industry.

Furthermore, to facilitate nuclear use and development, the Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (Nuclear Regulation Law) was enacted in 1957, in the spirit of the Atomic Energy Basic Law.

In the international arena, Japan has, since the launch of its nuclear use and development programme, concluded bilateral agreements with other nations to ensure the peaceful use of nuclear energy and has promoted nuclear use and develop-
ment while maintaining co-operative relations with these countries. The countries with which Japan has concluded agreements and maintains friendly co-operative relations at present include, in chronological order, the United States of America, Canada, the United Kingdom, France, Australia and China. In particular, the research co-operation agreement with the USA, which has been in effect since 1955, was revised in 1958, and major amendments were introduced in 1987 to emphasize further the spirit of nuclear non-proliferation and to clarify bilateral responsibilities.

Since the establishment of the IAEA in 1957, our country, as one of its Member States, has contributed to the management of the IAEA. In addition, after the ratification of the NPT in 1976, our country swiftly concluded an INFCIRC/153 type Safeguards Agreement with the IAEA in 1977 and accepted comprehensive (full scope) international safeguards, which assumes the maintenance and management of a SSAC with an independent verification system.

As our country relies for most of its supply of nuclear materials on other countries, it is vital that we gain the understanding of other countries with respect to the use of plutonium. To do this, it will become increasingly necessary to maintain transparency in the nuclear fuel cycle programmes, and gain international understanding and support on non-proliferation and other issues.

3. NUCLEAR MATERIALS AND FACILITIES RELATED TO SAFEGUARDS

Of the total electricity generation of $797.8 \times 10^9 \text{ kW} \cdot \text{h}$ at the end of fiscal 1992, 28.2%, or approximately $225 \times 10^9 \text{ kW} \cdot \text{h}$ generated comes from nuclear power. According to energy demand forecasts, $505 \times 10^9 \text{ kW}$ of power generation will be targeted for the year 2000.

Japan’s nuclear source materials and nuclear fuel materials, which support this nuclear power generation, are all subject to safeguards under the Nuclear Regulation Law and the NPT. At the end of 1992, Japan held roughly 3500 t of natural and depleted uranium, about 9400 t of low enriched uranium (LEU) and 33.5 t of plutonium. Figure 1 indicates the locations of our country’s major facilities, including nuclear power generation and a part of the nuclear fuel cycle.

In August 1993, the number of facilities which deal with nuclear materials under safeguards reached 248, including the major facilities indicated in Fig. 1. The Nuclear Regulation Law mandates all of these facilities to receive permits for installation, business, use and change. In addition, the Law mandates that these facilities receive permits for procedures which stipulate the implementation of strict accountability within these facilities. A breakdown of the types of facilities is indicated in Table I.
Characteristic of our country’s safeguards related facilities is that uranium enrichment, conversion and fuel fabrication facilities, which support nuclear power generation through the use of low-enriched uranium fuel, as well as light water reactors, are operated on a commercial scale; these facilities also cover a wide range of facilities to establish nuclear fuel cycles, including fast breeder reactors, advanced thermal reactors, a spent fuel reprocessing facility and a mixed oxide (MOX) conversion and fuel fabrication facility. These aim at establishing nuclear power generation through the use of plutonium fuel. With respect specifically to the plutonium cycle,
which originates in the Tokai Reprocessing Plant (TRP), an annual reprocessing capacity of approximately 90 t U is maintained. Meanwhile, the MOX conversion facility, which takes nuclear proliferation resistance characteristics into consideration, has adopted a microwave denitration method, unique to Japan, to produce MOX fuel as a joint effort with TRP. Furthermore, fuel for the MONJU fast breeder reactor (FBR) is being produced at the Plutonium Fuel Production Facility (PFPF). Located within the Tokai Works of the Power Reactor and Nuclear Fuel Development Corporation (PNC), these facilities are controlled by strict safeguards measures. The MONJU FBR, built in Tsuruga by the Sea of Japan, is currently in the final commissioning stage before criticality. Concerning safeguards, the Government of Japan and the IAEA were able swiftly to arrange facility attachment before fuel delivery.

Future projects include the Rokkasho Reprocessing Plant (RRP), which is to become the base for our country’s commercial use of plutonium. The facility is currently under construction to meet the planned operational launch date of 2000. Regarding safeguards for this facility, following the conclusion reached at the recent

<table>
<thead>
<tr>
<th>Facility type</th>
<th>Number</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nuclear power plants</td>
<td>46</td>
</tr>
<tr>
<td>Research reactors, critical assemblies</td>
<td>23</td>
</tr>
<tr>
<td>Mining and conversion facility</td>
<td>1</td>
</tr>
<tr>
<td>LEU fabrication plants</td>
<td>5</td>
</tr>
<tr>
<td>Enrichment plants</td>
<td>2</td>
</tr>
<tr>
<td>Reprocessing plant</td>
<td>1</td>
</tr>
<tr>
<td>Pu conversion development facility</td>
<td>1</td>
</tr>
<tr>
<td>Pu fabrication development facilities</td>
<td>2</td>
</tr>
<tr>
<td>Research facilities</td>
<td>19</td>
</tr>
<tr>
<td>Storage facility</td>
<td>1</td>
</tr>
<tr>
<td><strong>Subtotal</strong></td>
<td><strong>101</strong></td>
</tr>
<tr>
<td>Outside facilities</td>
<td>147</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>248</strong></td>
</tr>
</tbody>
</table>
large scale reprocessing (LASCAR) meeting, a study of specific approaches to be applied, and authentication and production of safeguards equipment which are expected to be adopted are being implemented with the co-operation of the IAEA.

4. JAPAN'S NATIONAL SAFEGUARDS SYSTEM

As was stated earlier, ever since the launch of its nuclear use and development programme, Japan has limited nuclear uses to peaceful purposes and has managed the programme under a strict domestic control system. In response to the ratification of the NPT in 1976, the Government amended the Nuclear Regulation Law and set up a national system which enabled the country to cope with the IAEA's full scope safeguards. An outline of the system is shown in Fig. 2.

Japan has set up during the past 22 years a unique SSAC, in which the Government's administrative organizations have very close business relations as regards technical aspects of the implementation of safeguards. As far as the processing of safeguards accountancy information is concerned, the Government has authorized

![Diagram of Japan's National Safeguards System]

FIG. 2. The safeguards system in Japan.
TABLE II. RESULTS OF NATIONAL INSPECTIONS IN 1992

<table>
<thead>
<tr>
<th>Facility type</th>
<th>Person-days</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nuclear power plants</td>
<td>310</td>
</tr>
<tr>
<td>Research reactors, critical assemblies</td>
<td>211</td>
</tr>
<tr>
<td>Mining and conversion facility</td>
<td>4</td>
</tr>
<tr>
<td>LEU fabrication plants</td>
<td>115</td>
</tr>
<tr>
<td>Enrichment plants</td>
<td>64</td>
</tr>
<tr>
<td>Reprocessing plant</td>
<td>687</td>
</tr>
<tr>
<td>Pu conversion development facility</td>
<td>150</td>
</tr>
<tr>
<td>Pu fabrication development facilities</td>
<td>280</td>
</tr>
<tr>
<td>Research facilities</td>
<td>34</td>
</tr>
<tr>
<td>Storage facilities</td>
<td>2</td>
</tr>
<tr>
<td><strong>Subtotal</strong></td>
<td><strong>1857</strong></td>
</tr>
<tr>
<td>Outside facilities</td>
<td>4</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>1861</strong></td>
</tr>
</tbody>
</table>

the Nuclear Material Control Center (NMCC) to conduct business on behalf of the Government in accordance with the Nuclear Regulation Law. The Safeguards Analytical Laboratory (SAL) located at Tokai-mura has been constructed by the Government and is being operated by the NMCC under contract with the Government. Technical support, including maintenance and operation of the non-destructive assay equipment at the inspection sites, is also entrusted to the NMCC.

Japan’s safeguards system begins with the approval of the control and accountancy procedures, applications for which had been submitted by facility operators before the start of operation of each nuclear facility. This entails scrutiny and approval of the procedures on the control and accountancy system, including records, the reporting system, the system for controlling measurement errors and the system for calculating material unaccounted for (MUF). Facilities for which approval has been granted are required to report to the Science and Technology Agency (STA), at each designated period, on the handling of nuclear source materials, nuclear fuel materials and nuclear equipment under international control. These collected reports are transmitted to the NMCC and checked for the self-consistency of each operator’s report as well as the mutual consistency among different facilities. After this has been confirmed, the reports are modified to IAEA
formats and sent to the IAEA via diplomatic channels. At the same time, submission of information on the country specific inventory based on bilateral nuclear co-operation agreements is mandatory. This information is also periodically sent to supplier countries. The major aim of our country’s control and accountancy system is to maintain the quality assurance of accountancy information that is vital to enhance the reliability of safeguards.

In addition, Japan maintains an independent verification system. The purpose of this system is to verify whether each facility complies with the control and accountancy procedures that have been approved to implement facility level safeguards, and whether high quality control and accountancy are carried out. National inspectors enter the facility and check the facility’s books, confirm the actual materials through non-destructive assays, and, in the case of facilities handling materials in bulk form, take samples of nuclear materials to verify the precision and accuracy of the facility’s accountancy. These samples are sent to SAL, where chemical analyses are conducted to check for any biases in the control and accountancy reports submitted by each facility. Meanwhile, IAEA inspectors check and confirm these inspections and, if necessary, conduct additional verifications to maintain the independence of international inspections.

National inspectors always carry out verification activities in close cooperation with IAEA inspectors. Table II shows the results of domestic inspections in 1992. Over 100 surveillance cameras and over 3300 seals are applied in domestic inspections.

The STA is promoting several research and development projects other than regulatory activities. For example, currently it is carrying out R&D work for a large scale reprocessing plant under contract with the NMCC. The establishment of accurate measurement techniques for the input solution vessel, inventory estimation technique for the product evaporator, and the related physical protection guidelines to be instituted by the operator are major tasks. The results of these tasks are expected to be applied to the Rokkasho Reprocessing Plant. The STA is encouraging the NMCC to act as a kind of a think-tank organization as far as R&D activities are concerned.

5. CO-OPERATION WITH THE IAEA

International safeguards based on the full scope Safeguards Agreement are implemented through good co-operative relations between the IAEA and our Government. Japan maintains a powerful co-operative relationship with the IAEA not only in regular safeguards activities but also regarding the development of safeguards technology.

Japan possesses a comprehensive nuclear fuel cycle, which makes it necessary for the country to accept a variety of safeguards measures. This is one of the reasons
why the Japanese Government has co-operated with the IAEA by dispatching professionals to numerous international scientific and technical meetings to study safeguards approaches, organized mainly by the IAEA and other agencies, such as the Tokai Advanced Safeguards Technology Exercise (TASTEX) and Hexapartite safeguards projects.

TASTEX was organized with the objective of developing safeguards technologies that should be adopted by small scale reprocessing plants. More than 13 developmental tasks were established and studied with the participation of the various countries concerned. Results of these tasks, including the use of an electro­manometer to measure the acceptance volume of spent fuel solution, a non-destructive measuring device to measure plutonium nitrate solutions after fission product separation, and a storage monitoring device for plutonium nitrate solution products, have now been implemented at the Tokai Reprocessing Plant on a regular basis.

The Hexapartite project was organized with the participation of related countries and institutions for the purpose of studying safeguards approaches for centrifuge uranium enrichment facilities. The results of the discussions held during the meetings are reflected in limited frequency unannounced access (LFUA) to the Ningyo Pilot Plant and the Rokkasho Commercial Plant.

The recent international forum on large scale reprocessing plant safeguards, named LASCAR, may also be mentioned. Aimed at the exchange of information on safeguards technological development, which will become necessary for commercial scale reprocessing plant slated for the start of operation at the end of the 1990s, this project was funded by our country's special contribution to the IAEA. Participants in the project were France, Germany, Japan, the United Kingdom, the USA, Euratom and the IAEA. Discussions over five years led to the conclusion that a wide range of safeguards techniques, available now and in the near future, could be effective in such reprocessing plants. With the co-operation of the IAEA, concrete studies are currently being implemented on Japan's Rokkasho Reprocessing Plant concerning the safeguards approaches based on the above conclusion.

Since 1993, Japan has supported the Information Treatment Assistance Programme (ITAP), a programme concerning the IAEA's handling of information, through extra-budgetary contributions. Programme activities include the collection of information related to nuclear energy that is publicized in various countries through news articles, papers at academic meetings and other contributions to strengthening IAEA safeguards by analysing information on imports and exports of nuclear materials and on the export of sensitive equipment and non-nuclear materials for nuclear use. These data are expected to provide assistance to the strengthening of safeguards being considered by the IAEA Board of Governors.

In addition, Japan has been conducting a support programme for IAEA safeguards (JASPAS) since 1981. By 1992, there were a total of 49 tasks, including completed and ongoing ones. In fiscal 1992, fifteen tasks were in operation, financed
by a total fund of about 87 million yen. Some of the achievements include facility specific safeguards equipment which was developed in connection with the facility, as well as general purpose inspection equipment such as improved Cerenkov viewing devices (ICVDs) and the compact monitoring surveillance system, COSMOS. These devices are expected to help increase the efficiency of IAEA safeguards in the future.

6. EXPECTATIONS FOR FUTURE IAEA SAFEGUARDS IMPLEMENTATION

Recently, the IAEA Secretariat informed the Board of Governors that it had initiated an internal programme called ‘93+2’ in order to strengthen the effectiveness of the international safeguards system. The Government of Japan will endorse effective execution of the projects by the IAEA and expects that fruitful results will be obtained for the establishment of a highly reliable safeguards system.

Measures for strengthening IAEA safeguards for detecting undeclared nuclear activities, for example the execution of environmental monitoring, have become one of the essential subjects in recent years. The Government of Japan has actively participated in discussions concerning such measures.

With regard to the early provision of design information, the Government of Japan has applied measures to co-operate with the IAEA to the extent appropriate. For instance, the first preliminary design information for the Rokkasho Reprocessing Plant was submitted to the IAEA Secretariat in January 1993, and additional information should follow successively when detailed designs are fixed.

The Government of Japan supports the reporting scheme on the export and import of nuclear material and on the export of specified equipment and non-nuclear material. It is important not only for States with full scope Safeguards Agreements but for all States, including the nuclear weapon States and the States with INF-CIRC/66 type Safeguards Agreements, that they take steps to co-operate in the scheme on a voluntary basis. The Government of Japan already began to report such information in 1993. In addition, the Government has made a special contribution to supporting the strengthening of the information treatment and evaluation system for IAEA safeguards through the ITAP.

Arising from the requirement of strengthening IAEA safeguards aiming at the early detection of undeclared nuclear activities and facilities, increasing efforts to implement new safeguards facilities and savings in the IAEA budget would lead to the necessity of streamlining the current IAEA safeguards, whilst maintaining their effectiveness. It is vital to make the fullest use of existing resources.

The Government of Japan would like to seek streamlining of the IAEA safeguards, e.g. by the extensive use of SSAC, the extensive and timely application of results obtained through Member States’ Support Programmes, and the development and application of unattended mode containment and surveillance systems. These ideas were proposed to the IAEA Board of Governors in May 1992.
Accomplishments to date include the introduction of inspection equipment for material acceptance verification without the presence of IAEA inspectors, and the introduction of automatic inspection equipment in areas that are not easily accessible. It should be noted that the introduction of this equipment enabled a reduction in the number of initially anticipated inspection person-days. This is expected to contribute greatly to reducing the IAEA's inspection workload.

Japan has a variety of experience in introducing unattended mode verification systems such as the one installed in the PFPF and in the difficult-to-access areas in JOYO and MONJU. The Government of Japan continues to encourage the installation of such systems.

Recently, a plan has been drawn up for Japan and the IAEA to share the use of inspection equipment in the hope of increasing the efficiency of the safeguards implementation. This includes the sharing of responsibilities for preparing inspection equipment, joint use of non-destructive measuring devices, common monitoring devices and specific seal types.

7. CONCLUSION

Nuclear energy activities in Japan are strictly limited to peaceful uses in accordance with the Atomic Energy Basic Law and the NPT, and a strict SSAC for nuclear materials has been established and implemented in order to secure the nation’s basic principle. It has been the fundamental policy of the Government firmly to establish the indigenous fuel cycle in Japan. In this respect, the MONJU FBR is to be brought to criticality soon. Further, the Rokkasho Reprocessing Plant, which should play a key role in the national fuel cycle system, is now under construction. In this regard, given the scope of peaceful uses of nuclear energy, maintenance of a credible SSAC and further strengthening the system by employing newly developed advanced technology should become an increasingly important issue.

Various techniques to be applied to the Rokkasho Reprocessing Plant are the subject of intensive development, with the guidance of the STA, as discussed earlier. Experiences and results acquired during the past years at MONJU and PFPF should be fully utilized for further efficient and effective implementation of IAEA safeguards. Various advanced techniques, including the examples given previously and subjects to be developed in the near future, should play an important role in optimizing the IAEA safeguards responsibilities in harmony with each country's SSAC.

Japan is ready to co-operate with the IAEA in the area of a competent safeguards regime.
THE ROLE OF A REGIONAL ORGANIZATION IN THE APPLICATION OF SAFEGUARDS

The example of ABACC

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Abstract

THE ROLE OF A REGIONAL ORGANIZATION IN THE APPLICATION OF SAFEGUARDS: THE EXAMPLE OF ABACC.

The Argentine Republic and the Federative Republic of Brazil began nuclear co-operation in the 1960s, and this has increased significantly since 1980, when an Agreement on the Peaceful Uses of Nuclear Energy was signed. In 1991 the Governments of the two countries created a Common System of Accounting and Control of Nuclear Materials, and the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC). In the same year the two countries signed with ABACC and the IAEA a Quadripartite Agreement for the application of safeguards on existing nuclear material in Brazil and Argentina. All nuclear material in the two countries is now under either ABACC or IAEA safeguards. The level of implementation achieved within such a relatively short time shows the possibility of successfully establishing regional systems for the application of safeguards.

1. INTRODUCTION

The Argentine Republic and the Federative Republic of Brazil together constitute a region in South America covering over 11.2 million square kilometres, with some 200 million inhabitants, with a trade of some US $7000 million annually. The gross domestic product (GDP) of the region has risen to over US $540 000 million, which represents 50% of the GDP of Latin America and the Caribbean; its population accounts for 35% of the total in this geographical area [1]. Both countries belong to the Mercosur South American Common Market which, together with Uruguay and Paraguay, is in the process of implementation, supported by a broad-ranging spirit of co-operation among the Parties. Nuclear co-operation between Argentina and Brazil in this region began during the 1960s and remains in full force today.
Although during the 1960s and 1970s this co-operation was not as intensive as could be wished, it nevertheless grew strongly after 1980, when the political conditions established by the resolution of controversies over the use of water resources led to the signature of an Agreement on the Peaceful Uses of Nuclear Energy between the two countries [2].

Implementation of this Agreement includes joint developments in various fields of nuclear energy, including the production of radioisotopes by cyclotron, the development of isotopic standards, radiological protection and nuclear safety as well as recycling of fuel elements.

As a natural outcome of this co-operation and the wish to endow their nuclear programmes with transparency, many commitments on the exclusively peaceful uses of nuclear energy have been undertaken by both nations.

These commitments were formulated in various joint declarations on nuclear policy by the Presidents of Brazil and Argentina: Foz de Iguaçu, 1985; Brasilia, 1986; Viedma, 1987; Iperó, 1988; Ezeiza, 1988; and the Joint Statements of Buenos Aires, 1990, and Foz de Iguaçu, 1990 [3—9].

The policies outlined in these declarations finally led to the signature on 18 July 1991 of a Bilateral Agreement on the Exclusively Peaceful Uses of Nuclear Energy [10].

This Bilateral Agreement entered into force on 12 December of the same year, after ratification by the Congresses of both countries.

It should be noted that this ratification implies the Agreement's promulgation with the force of law and that this law imposes mandatory common compliance on both countries.

The Bilateral Agreement sets up the Common System of Accounting and Control of Nuclear Materials (SCCC), and the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC).

On the basis of this Bilateral Agreement, a Quadripartite Safeguards Agreement was signed in December 1991 by the Republic of Argentina, the Federative Republic of Brazil, the ABACC and the IAEA [11].

2. THE BILATERAL AGREEMENT

The basic undertakings of the Bilateral Agreement are:

(a) To use nuclear materials and facilities under the signatories' jurisdiction or control exclusively for peaceful purposes.

(b) To prohibit and to prevent in the signatories' respective territories, and to abstain from carrying out, promoting or authorizing directly or indirectly, or from participating in any way in:

— the testing, use, manufacture, production or acquisition by any means of any nuclear weapon; and
the receipt, storage, installation, deployment or any other form of possession of any nuclear weapon.

(c) Considering that currently no technical distinction can be made between nuclear explosives for peaceful purposes and those for military purposes, the Parties also undertake to prohibit and prevent in their respective territories, and to abstain from carrying out, promoting or authorizing, directly or indirectly, or from participating in any way in the testing, use, manufacture, production or acquisition by any means of any nuclear explosive device while the above mentioned technical limitation exists.

As a basic control undertaking, the Parties agreed to submit all the nuclear material in all nuclear activities carried out in their territories or anywhere under their jurisdiction or control to the SCCC.

The Agreement also establishes that any serious non-compliance by either of the Parties gives the other Party the right to withdraw from the Agreement, with the obligation to notify the Secretary-General of the United Nations and the Secretary-General of the Organization of American States of this fact.

3. THE COMMON SYSTEM OF ACCOUNTING AND CONTROL OF NUCLEAR MATERIALS

The Agreement establishes the SCCC with the purpose of verifying that nuclear materials in all nuclear activities of the Parties are not diverted to uses not authorized under the terms of the Agreement.

The SCCC consists of the General Procedures and the Application Manuals for each installation.

The General Procedures specify the basic criteria and the requirements of the SCCC. Chapter 1 contains the criteria and the provisions for the starting point, exemption and termination of safeguards. It also includes the general rules for establishing adequate levels of accounting for and control of nuclear material, and shall later be detailed in the Application Manual for each facility or other locations, taking into account the usual parameters (nuclear material category, conversion time, inventory or annual throughput). Chapter 2 establishes the requirements at the State level for licensing nuclear facilities and other locations, and the requirements regarding relevant information for the SCCC (records, physical inventory and traceability of the measurement systems). Chapter 3 describes the procedures for the application of the SCCC at the State level.

The provisions for the application of the SCCC and ABACC are in Chapter 4. It includes the requirements of the relevant information that shall be provided to ABACC (technical questionnaires (TQ), inventory change report (ICR), material balance report (MBR), physical inventory list (PIL), and notification of transfers
from, to or between the States party). Additionally, Chapter 4 describes in a general way the purposes of the inspections, the scope of the inspections, the access for and the notice given for inspections. The general provision for the evaluation of the shipper-receiver differences and the material unaccounted for (MUF) are also included in this Chapter.

The General Procedures are supplemented by two Annexes giving the Report Forms and the Basic Schedule for Routine Communications.

4. THE BRAZILIAN-ARGENTINE AGENCY FOR ACCOUNTING AND CONTROL OF NUCLEAR MATERIALS

For the purpose of applying the SCCC in both countries, the Agreement also establishes the ABACC.

This Agreement endows the ABACC with the characteristics of an international agency, and its employees assume the status of international staff. Their privileges and immunities are established in an additional protocol to the Agreement, in the corresponding Headquarters Agreement signed with the Government of Brazil, and in a special Agreement signed with the Government of the Argentine Republic [10, 12, 13].

The organs of the ABACC are the Commission, its governing body, consisting of four members appointed by the Parties, each Government appointing two, and the Secretariat — its executive body.

The main functions of the Commission are:

— to take care of the functioning of the SCCC
— to supervise the functioning of the Secretariat
— to appoint the staff of the Secretariat
— to prepare a list of qualified inspectors from among those proposed by the Parties
— to inform the Parties regarding any non-compliance with the Agreement.

The decisions of the Commission may only be taken by a unanimous vote of its members.

Any anomaly noted as a result of the inspections or through the appraisal of the national records should be reported by the Secretariat to the Commission, which should urge the Party concerned to rectify the situation.

The Secretariat has the following functions:

— to implement the directives and instructions issued by the Commission
— to perform the necessary activities for the implementation and administration of the SCCC
— to act as the representative of the ABACC
— to designate and instruct the inspectors who will carry out the inspection
— to receive and evaluate the inspection reports
— to inform the Commission immediately of any discrepancy in the records of either of the Parties which emerges from the evaluation of the inspection results.

The Secretariat consists of a Secretary and a Deputy Secretary (whose nationalities alternate every year) and a staff of, currently, six senior Technical Officers (three for each Party), two Administrative Officers, four Administrative Auxiliaries, and about sixty part-time inspectors provided by the Parties (thirty from each country) who only report to the ABACC while carrying out their inspection duties.

The Agreement establishes that the Brazilian installations must be inspected by Argentine inspectors and vice versa.

The inspectors are experts usually working for the national authority or some other official organization in each country, and they are convoked by the ABACC's Secretariat whenever necessary. It should be emphasized that the inspection staff comprises not only experienced people who perform inspections at a national level but also experts in several areas of safeguards (destructive testing, destructive analysis and testing, design and nuclear installation operations, etc.).

The organic structure of the Secretariat consists of a Technical Unit and an Administrative-Financial Unit. The former includes the following areas:

— accounting of nuclear materials
— planning and evaluation
— operations
— technical support.

The annual budget of the ABACC is around US $2 million; this does not include the wages of the inspectors and consultants, which are borne directly by the countries; nor does it include the purchase of equipment, which is carried out under special arrangements.

The two Governments contribute equally to the ABACC's financial support; this is a legal obligation for both Parties, enforceable by law.

5. STATE OF IMPLEMENTATION

Regarding the state of implementation of the SCCC and ABACC, during the first months of 1992 efforts were devoted to obtaining the resources (premises, staff, financial support, etc.) and to preparation of the necessary regulations for its operation.

The corresponding Headquarters Agreement between the Federative Republic of Brazil and the ABACC was signed on 27 March 1992. The Secretariat of the ABACC began its operations in its headquarters in the City of Rio de Janeiro in July that year.
The statements of initial inventories of both countries were received in September 1992 and have been systematically updated since then.

Considering that both countries have at present nuclear material under IAEA safeguards (INFCIRC/66 Agreements), the Secretariat decided to assign priority to the control of the nuclear materials submitted only to the SCCC.

The actions corresponding to this priority were complemented in December 1993 by the checking of all the design information questionnaires of the installations, the verification of the full initial inventory in those installations, as well as discussion of the Application Manuals, which are at an advanced stage of execution.

It may thus be said that all nuclear material in both countries is now under ABACC or IAEA safeguards.

In order to achieve this objective, the following technical activities were implemented:

(a) Accounting: a database was developed which allowed the registration of the initial inventory and all changes thereafter.

(b) Inspectors: the inspection system was implemented successfully. Up to December 1993, 56 inspection were carried out in the two countries. To date, ABACC inspection efforts have totalled 30 person-days per month.

(c) Training of inspectors: two seminars for training inspectors were held in 1992, one in Brazil and the other in Argentina. Furthermore, with the support of the ABACC, the Argentine national authority organized a one month inspectors' course, held in June 1993. Inspectors of both nationalities participated in the course.

(d) A programme of approximately US $1.5 million was established for the purchase of equipment. A first stage, covering US $150 000, was completed, and a second one for US $500 000 is presently being executed. Funds for a third stage have been included in the 1994 budget.

Furthermore, all necessary arrangements were made for carrying out calibration and maintenance of the equipment, as well as for preparing and recording ABACC seals.

(e) Chemical and isotopic analysis of samples: In order to analyse samples, a programme to implement a network of Brazilian and Argentine laboratories was started, together with its corresponding intercomparison project.

The policy of the ABACC establishes that all samples taken from Argentine installations must be analysed in Brazil, and vice versa.

6. THE QUADRIPARTITE AGREEMENT

The Agreement for the Exclusively Peaceful Uses of Nuclear Energy was complemented by a Quadripartite Agreement between the two Governments, the
ABACC and the IAEA, concluded on 13 December 1991 in Vienna, by virtue of which the IAEA also assumes the responsibility of applying full scope safeguards in both countries.

The basic undertakings of this Agreement are the acceptance by the States party of safeguards, in accordance with the terms of the Agreement, on all nuclear materials in all nuclear activities within their territories, under their jurisdiction or carried out under their control anywhere, for the exclusive purpose of verifying that such material is not diverted to nuclear weapons or other explosive devices.

Furthermore,

(a) The IAEA shall have the right and the obligation to check that safeguards will be applied, in accordance with the terms of the Agreement, on all nuclear materials in all nuclear activities within the territories of the States party, under their jurisdiction or carried out under their control anywhere, for the exclusive purpose of verifying that such material is not diverted to nuclear weapons or other explosive devices.

(b) The ABACC undertakes, in applying its safeguards on nuclear material in all nuclear activities within the territories of the States party, to co-operate with the IAEA in accordance with the terms of the Agreement, with a view to ascertaining that such nuclear material is not diverted to nuclear weapons or other explosive devices.

(c) The IAEA shall apply its safeguards in such a manner as to enable it to verify, in ascertaining that there has been no diversion of nuclear material to any nuclear weapon or other nuclear explosive device, the findings of the SCCC. Verification shall include, inter alia, independent measurements and observations conducted by the IAEA, in accordance with the procedures specified in the Agreement. The IAEA, in its verification, shall take due account of the technical effectiveness of the SCCC.

Furthermore,

— The States party, the ABACC and the IAEA shall co-operate to facilitate the implementation of the safeguards provided for in the Agreement.

— The ABACC and the IAEA shall avoid unnecessary duplication of safeguards activities.

The Quadripartite Agreement shall enter into force on the date upon which the IAEA receives from the ABACC and the States party written notification that their respective requirements for entry into force have been met.

The General Part of the subsidiary arrangements to the Quadripartite Agreement has already been discussed by the Parties and will enter into force at the same time.

The Quadripartite Agreement assigns ABACC an active role in the implementation of safeguards. This role is fully described in the General Part of the Subsidiary Arrangements.
The fact that the date when the SCCC came into force and the ABACC was implemented was after signature of the Quadripartite Agreement provided for future relationships to be taken into consideration as well as for the necessary complementary action between the ABACC and the IAEA, for the application of safeguards arising therefrom.

7. CONCLUSIONS

The description of the efforts made by Brazil and Argentina to set up a Common System of Accounting and Control of Nuclear Materials, the conception of the ABACC to administer this, and the level of implementation achieved over the short period of time available show the possibility of successfully establishing regional systems for the application of safeguards.

The signature of the Quadripartite Agreement and the progress achieved with the IAEA in preparing for the implementation of this Agreement suggest the feasibility of these systems playing a major role, and contributing to the efficiency, effectiveness and success of the universal safeguards system to which we all aspire.

REFERENCES


STATUS OF PREPARATIONS FOR SAFEGUARDS IMPLEMENTATION IN UKRAINE

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Abstract

STATUS OF PREPARATIONS FOR SAFEGUARDS IMPLEMENTATION IN UKRAINE.

At the 36th session of the IAEA General Conference, Ukraine was ready to put all the Ukrainian nuclear power plants (NPPs) and research reactors under IAEA control, and during the 37th session Ukraine declared its readiness to conclude a Safeguards Agreement with the IAEA and apply safeguards to all nuclear materials used for peaceful purposes under Ukrainian jurisdiction and control. However, from the political point of view the main problem is the nuclear weapons inherited by Ukraine from the former Soviet Union. According to the last agreement concluded between the Russian Federation, the United States of America and Ukraine, in January 1994, all the nuclear weapons in Ukraine will be shipped for dismantlement to the Russian Federation in exchange for nuclear fuel for Ukrainian NPPs. However, there is a problem as to whether the type of agreement to be concluded between Ukraine and the IAEA should be on the basis of INFCIRC/153 or INFCIRC/66. Independently of the political aspects the technical preparations have already commenced. The Ukrainian State Committee on Nuclear and Radiation Safety (UkrSCNRS) is officially responsible for implementation of domestic and international safeguards issues. The paper outlines the main areas of UkrSCNRS activity and discusses other regulations and agreements.

As was stated at the 36th session of the IAEA General Conference, Ukraine was ready to put all the Ukrainian nuclear power plants (NPPs) and a research reactor under IAEA control, and during the 37th session Ukraine declared its readiness to conclude a Safeguards Agreement with the IAEA and apply safeguards to all nuclear materials used for peaceful purposes under its jurisdiction or control. Table I lists Ukrainian nuclear facilities and IAEA technical visits in 1992-1993.

However, from the political point of view, the main problem is the nuclear weapons inherited by Ukraine on its territory from the former Soviet Union. According to the last agreement concluded between the Russian Federation, the United States of America and Ukraine in January 1994, all the nuclear weapons located in Ukraine will be shipped for dismantling to the Russian Federation in exchange for nuclear fuel for Ukrainian NPPs.

However, there was a question regarding the type of agreement to be concluded between Ukraine and the IAEA: should the Agreement be on the basis of INFCIRC/153 or INFCIRC/66?
In January 1994 the IAEA Secretariat officially forwarded to the Permanent Mission of Ukraine to the International Organizations in Vienna the Draft Agreement between Ukraine and the IAEA for the application of safeguards to all nuclear material in all peaceful nuclear activities of Ukraine. The document INFCIRC/153 is to be used as the basis for the text of the Agreement, except for some articles to be taken from the Argentina/Brazil Agreement. It is anticipated that this Agreement will remain in force until superseded by an agreement between Ukraine and the IAEA in connection with Article III of the Treaty on the Non-Proliferation of Nuclear Weapons. The text of the Draft Agreement has been preliminarily approved by representatives of five Member States of the United Nations Security Council.

As was stated during the last meeting of the Board of Governors, Ukraine will work diligently with the IAEA to develop the final text of the Agreement to be presented to the next Board of Governors, meeting in June 1994. For this purpose the Ukrainian side is developing proposals, comments and questions to be discussed during the official negotiations on the Agreement. The matters to be clarified are associated with inspection activities, financial matters, liability, etc. The same concerns the technical questions to be included in Subsidiary Arrangements.

It should be emphasized that there were no analogous practices anywhere, where a State having such a large nuclear programme started to apply comprehensive safeguards to all its peaceful nuclear activities within a short period of time. In the former Soviet Union IAEA safeguards were applied formally, as allowed for a nuclear weapon State, to two selected reactors. Both of these are located in the Russian Federation. Moreover, the State System of Accounting and Control (SSAC) of nuclear materials did not exist in the Soviet Union; each facility used its own internal accountancy system based generally on manual procedures that were not compatible with the systems installed on the other facilities. There was no standard national system.

An important action to be taken is the establishment of a strong and cost effective safeguards system that complies with the IAEA requirements. This will require the establishment of a nuclear material accounting and reporting system and a verified inventory of the material at all Ukrainian facilities covered by the Agreement. A significant detail to be resolved is the relative timing of the notification of the Agreement being put in force and the readiness of the Initial Report for IAEA inspection activity.

Independently, political and organizational aspects of the technical preparations have already been implemented. The Ukrainian State Committee on Nuclear and Radiation Safety (UkrSCNRS) is officially responsible for implementation of domestic and international safeguards issues, and according to its Statute: "... keeps the State accountancy of nuclear materials and controls their storage, transportation and utilization" as well as "... represents the interests of Ukraine at the IAEA, and develops drafts of international agreements" [1].
TABLE I. LIST OF FACILITIES

<table>
<thead>
<tr>
<th>Nuclear power plants</th>
<th>Units in operation</th>
<th>Type of unit</th>
<th>IAEA technical visits undertaken in 1992–1993</th>
</tr>
</thead>
<tbody>
<tr>
<td>Zaporozhye NPP</td>
<td>5</td>
<td>WWER-1000</td>
<td>2</td>
</tr>
<tr>
<td>Rovno NPP</td>
<td>3</td>
<td>WWER-440 (2)</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>WWER-1000 (1)</td>
<td></td>
</tr>
<tr>
<td>Khmelnitskij NPP</td>
<td>1</td>
<td>WWER-1000</td>
<td>2</td>
</tr>
<tr>
<td>South Ukraine NPP</td>
<td>3</td>
<td>WWER-1000</td>
<td>3</td>
</tr>
<tr>
<td>Chernobyl NPP*</td>
<td>2</td>
<td>RBMK-1000</td>
<td>3</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Research reactors</th>
<th>Type of reactor</th>
<th>Capacity</th>
<th>Type of material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kiev Nuclear Research</td>
<td>WWER-M</td>
<td>10 MW(th)</td>
<td>LEU/HEU&lt;sup&gt;b&lt;/sup&gt;</td>
</tr>
<tr>
<td>Institute</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sevastopol Naval College</td>
<td>IR-100</td>
<td>200 kW(th)</td>
<td>LEU/HEU</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Research institute</th>
<th>Type of material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kharkov Institute of Physics and Technology</td>
<td>LEU/HEU bulk form</td>
</tr>
</tbody>
</table>

<sup>a</sup> Unit 4 destroyed in 1986; Unit 2 shut down in 1991.

<sup>b</sup> LEU = low enriched uranium, HEU = highly enriched uranium.

The main areas of UkrSCNRS activity in safeguards implementation are:

- creation of the legislative basis;
- establishment of the SSAC;
- defining basic requirements for the SSAC;
- issuing the regulations and standards, including procedures of recording and reporting systems for the State and facility levels;
- identifying the facilities and nuclear materials subject to accounting and control;
- creation of the basic information system;
- improving the operator’s measurement system;
- preparation of technical information.
"The Act of Ukraine on Nuclear Energy Utilization and Radiation Protection" has been drafted and adopted by the Presidium of the Ukrainian Parliament and now is lying on the 'waiting list' for action by the complete Parliament. Another set of regulations that have been issued and put into force is: the Statute on SSAC, regulations on recording and reporting systems for the State and facility levels. Some technical issues have already been prepared, including Design Information for each facility. At the same time as the manual reporting system was established, activities to create a computerized system were also implemented.

There are some real prospects within the framework of a co-operative Agreement [2] with the Swedish Nuclear Power Inspectorate (SKI) which provides practical assistance to the UkrSCNRS in training of State and facility level staff, including software to be used at both levels, as well as delivery of hardware.

Some other co-operative projects are to be implemented in the near future. One of these is the Agreement between the UkrSCNRS and the US Department of Defense concerning development of State systems of control, accounting and physical protection of nuclear materials to promote the prevention of nuclear weapons proliferation from Ukraine, concluded last December within the framework of the Safe Secure Dismantlement Project.

The latest proposal is a voluntary initiative of the Finnish Centre for Radiation and Nuclear Safety to support the Ukraine safeguards programme. It promises to be very helpful, i.e. there are several units in both countries having very similar design, construction and nuclear fuel.

These and some other support programmes need not to be duplicated and should be implemented on the basis of the Co-ordinated Technical Support Plan developed recently by the IAEA and a number of countries which have expressed their willingness to assist Ukraine in establishing a safeguards system.

REFERENCES


STATUS OF SAFEGUARDS IMPLEMENTATION IN THE REPUBLIC OF KOREA

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Abstract

STATUS OF SAFEGUARDS IMPLEMENTATION IN THE REPUBLIC OF KOREA.

The Republic of Korea (ROK) has been building up its nuclear power industry since its first power reactor went into operation in 1978. There are now eight PWRs and one CANDU unit in operation, accounting for 43% of the total electricity supply. The ROK became an IAEA Member State in 1957 and ratified the Non-Proliferation Treaty in 1975. A full scope Safeguards Agreement was concluded in the same year. Since then, a further 13 peaceful use nuclear facilities have been subject to IAEA safeguards, and two PWRs and one multipurpose research reactor are expected to be included as soon as they are completed. The legal authority for all nuclear activities in the ROK is primarily based on the Atomic Energy Act, revised in 1986 to accommodate the safeguards clauses concluded with the IAEA in 1976. So far, the State System of Accounting and Control (SSAC) of nuclear materials has been centred solely in the Atomic Co-operation Division of the Ministry of Science and Technology, but recently a new Nuclear Control Division was created. The Korea Institute of Nuclear Safety manages safeguards information collected from each facility and assists IAEA inspectors to perform their mission efficiently. A computer program called KAERI has been developed to help manage facility-level accounting data and also State-level data. The output is sent to the IAEA on diskette. Research projects have been concentrated on a material accounting system and non-destructive assay measurements on fresh fuel material. The Government is planning to establish a Nuclear Material Safeguards Centre, which will be responsible for the development of safeguards technology. This should come to fruition in the course of 1994. When the Centre is working to capacity, in a few years’ time, it is expected to help provide increased reliability and transparency in the peaceful use of nuclear materials in the ROK.

1. INTRODUCTION

The Republic of Korea (ROK) has been building up its nuclear power industry since its first nuclear power reactor, Kori-1, went into operation in 1978. Fifteen years later the ROK has now eight PWRs and one CANDU unit in operation, from which it generates 7616 MW(e), accounting for 28% of the total generation capacity and 43% of the total electricity supply.
With this rather rapid growth in its nuclear power programme, the ROK now ranks tenth in the world in terms of nuclear power generation capacity and it is the only country to have both CANDU reactors and PWRs.

Looking to its future development, two PWRs (1000 MW(e) each) and one CANDU unit (600 MW(e)) are now under construction, to be completed by 1997, and two more PWRs (1000 MW(e) each) and two CANDU units (600 MW(e)) are at the design stage, with completion planned by the end of this century.

In addition, the ROK has a long term plan to have six PWRs and one CANDU unit by the year 2006 so that nuclear power generation could account for 43% of the nation's total power supply.

2. SAFEGUARDS IN THE REPUBLIC OF KOREA: A HISTORICAL REVIEW

Since the ROK became an IAEA Member State in 1957, the first noteworthy encounter with in relation to safeguards came in 1968, when the ROK concluded the trilateral ROK–USA–IAEA agreement as it introduced its first research reactor.

<table>
<thead>
<tr>
<th>TABLE I. FACILITIES SUBJECT TO IAEA SAFEGUARDS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Name of the facility</td>
</tr>
<tr>
<td>Triga Mark II and III</td>
</tr>
<tr>
<td>Kori-1 (PWR)</td>
</tr>
<tr>
<td>Kyung-Hee University</td>
</tr>
<tr>
<td>CANDU Fuel Fabrication Plant</td>
</tr>
<tr>
<td>Wol-sung (CANDU) 1</td>
</tr>
<tr>
<td>Kori-2 (PWR)</td>
</tr>
<tr>
<td>Kori-3 (PWR)</td>
</tr>
<tr>
<td>Kori-4 (PWR)</td>
</tr>
<tr>
<td>Post-irradiation examination facility</td>
</tr>
<tr>
<td>Yeong-Gwang-1 (PWR)</td>
</tr>
<tr>
<td>Yeong-Gwang-2 (PWR)</td>
</tr>
<tr>
<td>Ul-Jin-1 (PWR)</td>
</tr>
<tr>
<td>Ul-Jin-2 (PWR)</td>
</tr>
<tr>
<td>PWR Fuel Fabrication Plant</td>
</tr>
<tr>
<td>KMRR</td>
</tr>
<tr>
<td>Yeong-Gwang-3 (PWR)</td>
</tr>
<tr>
<td>Yeong-Gwang-4 (PWR)</td>
</tr>
</tbody>
</table>
FIG. 1. Status of reports and records sent to the IAEA.

FIG. 2. IAEA person-days of inspection.
**TABLE II. STATUS OF SAFEGUARDS R&D WORK**

<table>
<thead>
<tr>
<th>Year</th>
<th>Project</th>
<th>Field application</th>
<th>Measurement system</th>
</tr>
</thead>
</table>
| 1988 | • Enrichment measurement by passive gamma ray analysis | • U-235 enrichment  
— UF₆ cylinders (depleted)  
— UO₂ drums (depleted and natural) | • Gamma measurement system  
— Hardware: Ge(Li), MCA, PC, plotter, printer  
— analysis program: Spectran-AT |
| 1989 | • Enrichment measurement by passive neutron analysis for UF₆ cylinders | • U-235 content  
— UF₆ cylinders (depleted and natural) | • Neutron measurement system  
— Hardware: SNAPᵃ, thermal printer, digital cassette drive  
— Analysis program: SNAP-II, HP-41CX |
| 1990 | • Field application of segmented gamma scanning system for accounting for nuclear materials in low density solid wastes from facilities  
• Modelling of computerized material balance system for CANDU fuel fabrication facility | • U-235 content  
— Solid waste drums | • SGSSᵇ  
— Hardware: HPGe, MCAᶜ, PC, printer, plotter  
— Software: Spectran-AT |
<table>
<thead>
<tr>
<th>Year</th>
<th>Development of spent fuel verification technique (I)</th>
<th>Spent PWR fuel verification (qualitative analysis)</th>
<th>Gamma ray detection system</th>
</tr>
</thead>
<tbody>
<tr>
<td>1991</td>
<td>• Development of spent fuel verification technique (I)</td>
<td>• Spent PWR fuel verification (qualitative analysis)</td>
<td>• Gamma ray detection system</td>
</tr>
<tr>
<td></td>
<td>• Spent PW R fuel verification (qualitative analysis)</td>
<td>• Gamma ray detection system (- Hardware: HPGe, MCA, PC, printer, plotter)</td>
<td>- Software: Spectran-AT</td>
</tr>
<tr>
<td></td>
<td>• Gamma ray detection system (- Hardware: HPGe, MCA, PC)</td>
<td>- Software: Spectran-AT</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Year</th>
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<td>1992</td>
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<td>• Spent PWR fuel verification (qualitative analysis)</td>
<td>• Gamma ray detection system</td>
</tr>
<tr>
<td>1992</td>
<td>• Development of material balance evaluation technique in fuel fabrication facility</td>
<td>• Material balance evaluation in bulk handling facility</td>
<td>- Hardware: HPGe, MCA, PC</td>
</tr>
<tr>
<td>1992</td>
<td>• Material balance evaluation in bulk handling facility</td>
<td></td>
<td>- Software: Spectran-AT</td>
</tr>
</tbody>
</table>

\* SNAP = Shielded neutron assay probe.
\* SGSS = Segmented gamma scanning system.
\* MCA = Multichannel analyser.
The NPT came into force in 1970; the ROK signed and ratified it in 1975 and in the same year the Safeguards Agreement with the IAEA was concluded, whereby the ROK accepted full scope safeguards. 

As a result of this agreement, two research reactors, TRIGA II and III, were the first nuclear facilities to which IAEA safeguards were applied and, accordingly, the first physical inventory listing (PIL) was produced as an initial report to the IAEA in 1976. This was the first time that the ROK sent an official report of this kind to the IAEA.

Since then, a further 13 peaceful use nuclear facilities have been subject to IAEA safeguards, and two PWRs and one multipurpose research reactor are expected to be included when they are ready (Table I).

As was noted earlier, the ROK nuclear power plant programme has grown rather rapidly and therefore IAEA safeguards related activities have been greatly increased to keep up with safeguards requirements.

As can be seen in Table I, there was no net increase in nuclear facilities from 1988 to the present, but nuclear activities, on the other hand, have increased significantly (Fig. 1).

IAEA on-site verification activities have also greatly increased, as shown in Fig. 2.

3. STATUS OF SAFEGUARDS IMPLEMENTATION IN THE REPUBLIC OF KOREA

3.1. Background

Article 7 of the ROK–IAEA Safeguards Agreement states: "the Government of the Republic of Korea shall establish and maintain a system of accounting for and control of all nuclear material subject to safeguards under this Agreement" and, later, "The Agency, in its verification, shall take account of the technical effectiveness of the Republic of Korea's system".

In order to meet this requirement the Government has been taking the necessary measures. The current status of safeguards R&D work is shown in Table II.

It is worth noting, however, that even though the Government is at the top and the centre of the system in terms of hierarchy and national role, it is mutual cooperation with every organization and institution involved that have made safeguards work successfully to date.

3.2. Legal framework

The legal authority for all nuclear activities in the ROK is primarily based on the Atomic Energy Act, which was revised in 1986 to accommodate the safeguards
clauses concluded with the IAEA in 1976. Before this, safeguards were carried out by directly applying the ROK-IAEA Safeguards Agreement.

In the Atomic Energy Act, 'internationally controlled materials' are defined as such materials (nuclear and non-nuclear), equipment and plant imported into the ROK from the nuclear supplier countries. In addition, the Act requires each facility operator to appoint a licensed safeguards officer responsible for both facility safeguards and radiation safety and also requires a safeguards manual to be set up prior to the operation of a facility.

In the Presidential Order, which ranks below the Act in terms of legal authority, but which is very important in the actual enforcement of the Act, mainly due to the detailed procedures and descriptions that it contains, a nuclear facility operator is required to submit a draft safeguards manual to the Ministry of Science and Technology (MOST) for approval.

The detailed descriptions of internationally controlled material and the contents of the safeguards manuals are specified in the related Regulations.

In order to make such laws and orders effective, appropriate penal provisions are attached.

Study is going on to find out whether more legal support is needed to help the safeguards system work more effectively, and there is a strong body of opinion that there should be a new clause introducing a domestic inspection regime (other than the IAEA inspection) which would enable ROK experts to inspect their own nuclear facilities prior to the IAEA inspection, with the aim of showing the international community the reliability and transparency of the ROK's use of nuclear materials.

4. ORGANIZATIONS

So far, in the Government, the working organization of the State System of Accounting for and Control (SSAC) of nuclear materials has been centred solely on the Atomic Co-operation Division of the MOST, but recently a new division, the Nuclear Control Division, was created to intensify the SSAC. The SSAC is now jointly supported by these two divisions (Fig. 3).

To help the Government carry out safeguards work, the Korea Institute of Nuclear Safety (KINS) is managing safeguards information collected from each facility and assisting the IAEA inspectors to perform their mission efficiently in the ROK.

Each facility or material balance area (MBA) has its own designated independent safeguards officers, as it was strongly recommended by the Government to strengthen the SSAC with a workforce specialized in safeguards work.

It is felt that more people need to be involved in safeguards at each facility because of the rapid expansion of ROK nuclear energy capacity on the one hand and
the Republic's intention to show the international community its reliability and transparency in dealing with nuclear material on the other.

The Government supports that safeguards should have higher priority in nuclear policy than previously, considering the recent development of international opinion as regards the NPT.

5. COMPUTERIZED SAFEGUARDS DATA MANAGEMENT SYSTEM

A computer program called KAERI has been developed to help manage facility-level accounting data and also State-level data (Fig. 4). The output from this
program, a PC diskette, eventually became the official medium for sending reports to the IAEA. The characteristics of the program are as follows:

- it can be run by any IBM compatible PC (XT and above)
- it includes an extensive error checking routine
- it is capable of updating all facility records
- it is capable of database management at State level
- it is capable of producing an official report by the IAEA.

FIG. 4. Management of computer system for nuclear material accountancy and control.
Besides this main accounting program, two other computer programs have been developed for safeguards purposes; one is for plant inventory verification (PIV) raw data treatment and the other is for material unaccounted for (MUF) evaluation for a facility similar to the IAEA's MUF evaluation process.

6. R&D PROGRAMME AND MANPOWER TRAINING IN SAFEGUARDS

Research and development projects related to safeguards started in 1985 when an independent Safeguards Department was formed for the first time in the Korea Atomic Energy Research Institute (KAERI).

So far, research projects have been concentrated on a material accounting system and non-destructive assay (NDA) measurement techniques on fresh fuel material. These projects are expected to be extended further to chemical analysis and measurement and accounting of irradiated nuclear material before spent fuel is handled in the ongoing ROK/Canada/USA joint project, direct use of PWR spent fuel in CANDU reactors (DUPIC).

7. CONCLUSION

As the share of nuclear energy in the ROK is becoming greater in its long term energy development programme, whilst at the same time efforts towards the peaceful use of nuclear energy in the international community are gaining momentum, the ROK Government is planning to establish a Technology Centre for Nuclear Control (TCNC) which will have the sole responsibility for the development of safeguards technology and its application in order to meet ever-increasing safeguards needs.

If all goes according to plan, the Centre will come to reality by the first half of 1994 as an institution attached to KAERI.

When the Centre operates at full capacity in a few years' time, it is expected to conduct efficient work towards increased reliability and transparency in the peaceful use of nuclear materials in the Republic of Korea.
ASSISTANCE TO NEWLY INDEPENDENT STATES IN ESTABLISHING STATE SYSTEMS OF ACCOUNTING AND CONTROL OF NUCLEAR MATERIAL

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Department of Safeguards,
International Atomic Energy Agency,
Vienna

Abstract

ASSISTANCE TO NEWLY INDEPENDENT STATES IN ESTABLISHING STATE SYSTEMS OF ACCOUNTING AND CONTROL OF NUCLEAR MATERIAL.

Nuclear trade and co-operation among States are essentially dependent upon effective and credible safeguards. The disintegration of the former Soviet Union has resulted, inter alia, in the emergence of a number of newly independent States (NIS). With one exception, all the NIS have declared their intention either to become or to remain non-nuclear-weapon States, but many of them have nuclear programmes. However, the nuclear infrastructure on which those programmes once rested is no longer in place and needs to be reconstructed. The paper outlines work under way among the IAEA, its Member States and the NIS relating to the establishment and development in the NIS of State Systems of Accounting and Control (SSAC) of nuclear material. The paper describes IAEA activities in the NIS, including fact-finding missions and technical visits, the successful attempts to find donor States providing voluntary funding and expertise, and the co-ordination of technical support between the IAEA and the donor States.

1. GENERAL

Safeguards depend for their effectiveness largely on the extent to which governments ensure that operators keep accurate, precise and complete records; promptly send the IAEA the required reports; employ reliable and accurate equipment for measurement and analysis; take inventories of nuclear material at the prescribed intervals; and determine, at each inventory taking, the amount of nuclear material unaccounted for.

In Safeguards Agreements pursuant to the NPT the State is required to establish and maintain a State System of Accounting and Control (SSAC) of nuclear material within its territory, jurisdiction or control.
TABLE I. NUCLEAR FACILITIES IN THE NEWLY INDEPENDENT STATES

<table>
<thead>
<tr>
<th>Country</th>
<th>Facilities and Activities</th>
</tr>
</thead>
<tbody>
<tr>
<td>Armenia</td>
<td>2 WWER reactors</td>
</tr>
<tr>
<td>Belarus</td>
<td>2 critical assemblies&lt;br&gt;Fresh and spent fuel storage</td>
</tr>
<tr>
<td>Estonia</td>
<td>Uranium ore refining plant&lt;br&gt;(2 training reactors)</td>
</tr>
<tr>
<td>Georgia</td>
<td>2 research reactors (IRT, TTR)&lt;br&gt;R&amp;D facility&lt;br&gt;Critical assembly</td>
</tr>
<tr>
<td>Kazakhstan</td>
<td>FBR (BN-350)&lt;br&gt;4 research reactors (WWR, pulse graphite, IWG, RA)&lt;br&gt;LEU fuel fabrication&lt;br&gt;R&amp;D facility&lt;br&gt;Critical assembly&lt;br&gt;Uranium mining, ore refining plants</td>
</tr>
<tr>
<td>Kyrgyzstan</td>
<td>Uranium mining plants</td>
</tr>
<tr>
<td>Latvia</td>
<td>Research reactor (IRT)&lt;br&gt;Critical assembly</td>
</tr>
<tr>
<td>Lithuania</td>
<td>RBMK reactors (2)</td>
</tr>
<tr>
<td>Tajikistan</td>
<td>Uranium mining and refining plants</td>
</tr>
<tr>
<td>Ukraine</td>
<td>4 RBMK reactors (separate SF storage)&lt;br&gt;16 WWER reactors&lt;br&gt;2 research reactors (WWR, training)&lt;br&gt;R&amp;D facility&lt;br&gt;Critical assembly&lt;br&gt;Uranium ore refining plant</td>
</tr>
<tr>
<td>Uzbekistan</td>
<td>2 research reactors (WWR, pulse type)&lt;br&gt;Several mining and refining plants</td>
</tr>
</tbody>
</table>

All the newly independent States (NIS) of the former Soviet Union, with the exception of the Russian Federation, have declared their intention either to become or to remain non-nuclear-weapon States (NNWS). As of now, nine of the NIS — Armenia, Azerbaijan, Belarus, Estonia, Georgia, Kazakhstan, Latvia, Lithuania and Uzbekistan — are Parties to the NPT. Additionally, by the terms of the Lisbon Protocol of 23 May 1992, Ukraine agreed to accede to the NPT “as a NNWS Party in the shortest possible time and begin immediately to take all necessary actions to this end, in accordance with their constitutional practices”.

Many of the NIS have nuclear programmes including uranium mining and refining. States in this category are Armenia, Belarus, Estonia, Georgia, Kazakhstan, Kyrgyzstan, Latvia, Lithuania, the Russian Federation, Tajikistan, Ukraine and Uzbekistan. The scope of the nuclear programmes in the NIS is summarized in Table I. The facility in Ukraine with several significant quantities (SQ) of highly enriched uranium (HEU), is identified in Table I as an R&D facility.

2. IAEA ACTIVITIES IN THE NEWLY INDEPENDENT STATES

The IAEA has a direct interest in matters relating to international safeguards and non-proliferation. To support the NIS NNWS in meeting national and international obligations, either assumed or in prospect, in the field of nuclear non-proliferation, the IAEA embarked in 1992 on a number of activities aimed at helping them to establish and/or further develop their SSACs.

For the above mentioned purpose, the Director General Established an Inter-Departmental Co-ordinating Group within the IAEA, chaired by the Assistant Director General for External Relations, M. ElBaradei.

As in many countries, SSACs are also charged with responsibilities in physical protection, import/export control and regulatory matters. It became logical to incorporate these topics in the support activities to SSACs for the NIS where such assistance would be required. In order to cover all of these topics, extensive support from Member States (donor States) became essential.

The work consisted of and continues to consist of:

1. carrying out fact finding missions/technical visits;
2. finding interested donor States; and
3. co-ordinating technical support.

This paper reviews the results to date of the three major activities by the IAEA and donor States.

2.1. IAEA fact finding missions/technical visits

In 1992, fact finding missions were carried out in Belarus, Kazakhstan and Ukraine. In 1993, such missions took place in Armenia, Estonia, Kyrgyzstan, Latvia, Lithuania and Uzbekistan. Further missions to Azerbaijan, Georgia and Tajikistan are planned for 1994 (see Table II). The objectives of fact finding missions are, inter alia, to:

- enquire about the timing of likely accession to the NPT and acceptance of technical visits to prepare for the NPT;
- identify relevant contact persons and organizations; and
TABLE II. IAEA FACT-FINDING MISSIONS AND TECHNICAL VISITS TO NEWLY INDEPENDENT STATES

<table>
<thead>
<tr>
<th>Newly independent State</th>
<th>Fact-finding missions/technical visits</th>
</tr>
</thead>
<tbody>
<tr>
<td>Armenia</td>
<td>2</td>
</tr>
<tr>
<td>Azerbaijan</td>
<td>—</td>
</tr>
<tr>
<td>Belarus</td>
<td>3</td>
</tr>
<tr>
<td>Estonia</td>
<td>1</td>
</tr>
<tr>
<td>Georgia</td>
<td>—</td>
</tr>
<tr>
<td>Kazakhstan</td>
<td>6</td>
</tr>
<tr>
<td>Kyrgyzstan</td>
<td>1</td>
</tr>
<tr>
<td>Latvia</td>
<td>1</td>
</tr>
<tr>
<td>Lithuania</td>
<td>1</td>
</tr>
<tr>
<td>Tajikistan</td>
<td>—</td>
</tr>
<tr>
<td>Ukraine</td>
<td>8</td>
</tr>
<tr>
<td>Uzbekistan</td>
<td>2</td>
</tr>
</tbody>
</table>

No known nuclear programmes in Moldova and Turkmenistan.

— identify and list needs as regards co-ordinated technical support plans for each individual NIS.

In 1993, technical visits were conducted to most major facilities in Armenia, Belarus, Ukraine, Kazakhstan and Uzbekistan. The objectives of technical visits are to:

— obtain information about the operator’s nuclear material flows, quantities, categories and measurement system;
— further define needs lists for co-ordinated technical support;
— identify safeguards equipment requirements; and
— demonstrate the verification of nuclear material and equipment used.

Technical discussions and demonstrations also took place. These visits have been a valuable means of supplying the basic information needed for the Department of Safeguards to initiate preparations for implementing safeguards in the respective
countries, and to familiarize the relevant State and facility officials with the IAEA's procedures and requirements.

Those facilities which may become subject to safeguards have been identified and associated information concerning nuclear material flow quantities and categories has been obtained. On the basis of this information, safeguards equipment needs have been identified, budgeted for, and initial purchases made for longer lead time items. Inspection resource requirement estimates have been made based upon draft safeguards approaches prepared or updated for all major NIS facility types.

In discussions with facility operators, IAEA inspection procedures have been reviewed in detail, the associated equipment demonstrated, and technical requirements documented for surveillance equipment installation. At some sites, the nuclear material measurement capabilities of the facilities have been reviewed with IAEA experts and compared with international standards to help define 'equipment needs'. Through these interactions, the needs of the individual State for assistance with infrastructure, development and equipment have also been identified.

Discussions at all levels have also helped to identify hardware and training needs in the basic SSAC infrastructure, including computers and software for nuclear materials accounting, communication systems, and instrumentation used by State inspectors. Consultations are continuing about the legal aspects of the NPT and Safeguards Agreements, and on facility design verification procedures.

2.2. Finding donor States

From the outset, for reasons already stated above, the IAEA recognized that it could not complete this extensive work alone and would have to rely heavily on voluntary funding and expertise from Member States. The IAEA consequently compiled and sent to potential donors needs lists, as discussed with the recipients, i.e. requirements in the NIS relating to non-proliferation. Being aware that some countries had already initiated support activities to one or more NIS or were in the process of establishing bilateral agreements, the IAEA expected that donor States would consider granting additional funds and harmonize their support activities in a co-ordinated manner. On the basis of suggestions from some Member States, a meeting was called in Vienna on 27-28 May 1993 where representatives from Australia, Belgium, Canada, Finland, France, Hungary, Japan, South Africa, Sweden, the United Kingdom, the United States of America, the Commission of the European Communities and the Organisation for Economic Co-operation and Development expressed their readiness to help the NIS improve their SSACs in a co-ordinated manner. So far, seven countries have made funding available and have been active or are expected soon to participate actively in actual support activities. Additional countries are ready to be called upon to assist if the need arises.
2.3. Co-ordinated Technical Support

'Co-ordinated Technical Support' is a term used to describe the co-ordinated support by the IAEA and donor States to the NIS in order to support SSACs at facility and State levels, including physical protection and export/import. The 'Co-ordinated Technical Support Plan' is approved by the State concerned and is intended to be a main tool in helping both State and facility authorities in meeting their responsibilities.

Co-ordination of efforts has since focused on the preparation and subsequent implementation of Co-ordinated Technical Support Plans for each individual NIS. These plans identify the needs to be addressed, the time-scale over which the associated activities are to be conducted and the areas of intended contribution from each of the donor States. The plans contain a phased approach to the support.

Phase I addresses immediate needs, with emphasis on support to existing authorities in improving legislative infrastructure and on SSAC requirements, in particular in relation to concluding and implementing a Safeguards Agreement with the IAEA.

Phase II will include completion of the legal infrastructure, improving operators' measurement systems and other components of material control and accounting, physical protection and export/import control systems. Training is recognized as an important element in the successful transfer of donor support and is included with each planned technical activity.

To date, such plans have been agreed for Belarus, Kazakhstan and Ukraine; donor States have been identified, and Implementation/Co-ordination Committees with responsible persons from each of the donor States and the IAEA have been or are being set up. Representatives of donor States are presently making facility visits in order to familiarize themselves with the areas, as identified in the Co-ordinated Technical Support Plan, for which they have accepted responsibility. Funds have been made available to implement the complete Co-ordinated Technical Support Plan for these States. Co-ordinated Technical Support Plans for the remaining NIS are in the process of being set up. One of the donor States has offered its assistance to help the remaining NIS in setting up the basic infrastructure of their SSACs; such assistance would contribute to achieving a certain degree of compatibility among all NIS. Other donor States have expressed willingness to contribute to the implementation of identified elements of the Co-ordinated Technical Support Plans in one or more of the NIS. The IAEA will contact potential donor States to solicit funding and expertise in order to cover the complete Co-ordinated Technical Support Plans for the NIS.
<table>
<thead>
<tr>
<th>Event</th>
<th>Location</th>
<th>Date</th>
<th>Organizer</th>
</tr>
</thead>
<tbody>
<tr>
<td>SSAC Seminar</td>
<td>Kiev, Ukraine</td>
<td>December 1992</td>
<td>Ukrainian State Committee on Nuclear and Radiation Safety and the IAEA</td>
</tr>
<tr>
<td>Reprocessing Seminar</td>
<td>Sellafield, UK</td>
<td>December 1992</td>
<td>Department of Trade and Industry</td>
</tr>
<tr>
<td>Management Seminar</td>
<td>Stockholm, Sweden</td>
<td>December 1992</td>
<td>Swedish Nuclear Power Inspectorate</td>
</tr>
<tr>
<td>Safeguards Seminar</td>
<td>Stockholm, Sweden</td>
<td>March 1993</td>
<td>Swedish Nuclear Power Inspectorate</td>
</tr>
<tr>
<td>Reprocessing Seminar</td>
<td>Sellafield, UK</td>
<td>April 1993</td>
<td>Department of Trade and Industry</td>
</tr>
<tr>
<td>Safeguards Seminar</td>
<td>Springfield and Dounreay, UK</td>
<td>April 1993</td>
<td>Department of Trade and Industry</td>
</tr>
<tr>
<td>Safeguards Seminar</td>
<td>Stockholm, Sweden</td>
<td>April 1993</td>
<td>Swedish Nuclear Power Inspectorate</td>
</tr>
<tr>
<td>Training Course on Physical Protection</td>
<td>Santa Fe, NM, USA</td>
<td>May 1993</td>
<td>Department of Energy</td>
</tr>
<tr>
<td>Training Course in the Implementation of SSAC</td>
<td>Los Alamos, NM, USA</td>
<td>May 1993</td>
<td>US Support Programme to the IAEA</td>
</tr>
<tr>
<td>Nuclear Material Accounting and Control</td>
<td>Novosibirsk, Russian Federation</td>
<td>May 1993</td>
<td>Gosatomnadzor of Russia</td>
</tr>
<tr>
<td>Seminar on the Organization of SSAC</td>
<td>Alma-Ata, Kazakhstan</td>
<td>June 1993</td>
<td>Atomic Energy Agency of the Republic of Kazakhstan and the IAEA</td>
</tr>
<tr>
<td>Nuclear Material Accountancy at WWERs</td>
<td>Paks, Hungary</td>
<td>November 1993</td>
<td>Hungarian and Swedish Support Programmes to the IAEA</td>
</tr>
<tr>
<td>Seminar on Nuclear Law</td>
<td>Leiden, Netherlands</td>
<td>September 1993</td>
<td>OECD Nuclear Energy Agency and the IAEA</td>
</tr>
<tr>
<td>Reprocessing Seminar</td>
<td>Sellafield, UK</td>
<td>October 1993</td>
<td>Department of Trade and Industry</td>
</tr>
<tr>
<td>Safeguards for Uranium Processing and Breeder Reactors</td>
<td>Springfield and Dounreay, UK</td>
<td>November 1993</td>
<td>Department of Trade and Industry</td>
</tr>
</tbody>
</table>
3. ADDITIONAL ASSISTANCE

Additional support by donor States and the IAEA to the NIS has included contributions to SSAC training activities organized by donor States and seminars on the organization of SSACs in Ukraine and Kazakhstan (see Table III). These enjoyed broad NIS participation.

The IAEA and donor States have also given legislative assistance to the NIS. This is aimed at establishing, within each of them, a comprehensive framework of nuclear law covering all areas of nuclear activity. Examples of recipients of such assistance are Kazakhstan, Belarus and Ukraine. In September 1993, the IAEA also co-sponsored, with the Nuclear Energy Agency in Leiden (Netherlands), a Training Seminar for Lawyers and Regulators.

4. CONCLUSION

Soundly established SSACs are fundamental to a State’s ability to benefit fully from the peaceful uses of nuclear energy. Nuclear trade and co-operation among States are essentially dependent upon effective and credible safeguards. These, in turn, rely heavily on SSACs. The disintegration of the former Soviet Union has resulted, inter alia, in the emergence of a number of new States, many with substantial nuclear programmes. Consequently, the nuclear infrastructure on which those programmes once rested is no longer in place. Bearing this in mind and drawing on the wishes of these NIS, the nuclear infrastructure needs to be reconstructed if the new States are to derive maximum benefit from the peaceful exploitation of the atom. Against this background, and since 1992, the IAEA, together with donor States, has been helping the NIS to build up their respective nuclear infrastructures. Much of the activity relates directly to the establishment of reliable SSACs. The IAEA and donor States are therefore fulfilling a vital function in the NIS. Much remains to be done, but much is already under way.

The co-operative spirit of all parties involved is very much in evidence; the NIS have been very open and have provided IAEA and donor States access to their nuclear programmes and, to a large extent, to facilities without a Safeguards Agreement in place. Also, donor States, recognizing the common interest in strengthening the SSAC infrastructure in the NIS, have responded in a very positive manner. Recently, a new donor State joined this work, offering both expertise and many millions of dollars as well.

It is therefore appropriate to end this paper by thanking both the recipient States for their co-operative attitudes and the donor States for their generous and positive help.
MATERIAL ACCOUNTANCY

(Session 3)

Chairman

A.N. PRASAD

India
EFFECTIVE NUCLEAR MATERIALS CONTROL AND ACCOUNTANCY FOR COMMERCIAL MIXED OXIDE FUEL FABRICATION PLANTS IN THE UNITED KINGDOM

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G.P. SNAPE
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Windscale, Seascale, Cumbria

United Kingdom

Abstract

EFFECTIVE NUCLEAR MATERIALS CONTROL AND ACCOUNTANCY FOR COMMERCIAL MIXED OXIDE FUEL FABRICATION PLANTS IN THE UNITED KINGDOM.

A mixed oxide (MOX) fuel demonstration facility (MDF), constructed under a joint project between British Nuclear Fuels plc (BNFL) and AEA Technology, is now operating satisfactorily, qualification programmes are complete and production is under way. BNFL now intends to build a full size production plant called the Sellafield MOX Plant (SMP). Achieving effective and efficient nuclear materials control and accountancy (NMCA) has been an important consideration during the formulation of these projects and throughout the detailed design work. The goal is to achieve the highest NMCA and safeguards credentials in a flexible and highly commercial manufacturing environment.

1. INTRODUCTION

The production of mixed oxide (MOX) fuel is not a new process; many tonnes of MOX have been safely and successfully loaded into reactors around the world. The construction of MOX plants at Sellafield to handle recycled plutonium, coupled with the capability of BNFL's Oxide Fuel Complex (OFC) to process recycled uranium, effectively closes the fuel cycle. Such facilities are important elements of a responsible fuel management strategy and serve the needs of customers who express a wish to utilize their valuable fissile materials recovered from reprocessing.
The siting of a large commercial MOX plant at Sellafield (SMP) builds on the experience gained in plutonium fuel fabrication for fast reactors and has the advantage that MOX fuel can be fabricated without the plutonium produced in the reprocessing plants leaving the facility. SMP will be built immediately adjacent to the thermal oxide reprocessing plant (THORP) to form an integrated unit with direct transfers within the same complex.

2. BASIC PRODUCTION MODEL

MOX and low enriched uranium (LEU) fuel fabrication plants share many of the same production processes, but the presence of plutonium in MOX requires more shielding and high levels of automation. Increased shielding and containment in many respects are an aid to achieving effective material control. Figure 1 shows the basic production flow model.

![Diagram of basic production flow model](https://via.placeholder.com/150)

**FIG. 1. Basic production flow model.**

2.1. Weighing areas
(see Fig. 1)

A small store for plutonium dioxide cans and a uranium dioxide drum store provide the principal feeds to the pelleting process area. The process manufactures sintered pellets from a blend of uranium, plutonium and recycled MOX powders. Weighing determines the amount of powder fed from the hopper and dispensing system into the high energy ball mill (attritor mill). The milling process is rapid and
produces a highly homogeneous powder blend which is then conditioned in a spheroidizer. This milling and conditioning process is known as the short binderless route and utilizes gravity flow to transfer materials. The granules from the spheroidizer are pressed into green pellets, which are placed in boats and sintered in a furnace. Sintered pellets are ground to size, using a dry grinding process, and then inspected. Samples of sintered pellets are taken for destructive analysis and resinter testing. For transferring pellets individually, a cushion transfer system is used.

Good sintered pellets are placed in trays and transferred into an in-line pellet store to await analytical results. Trays of pellets of the required plutonium enrichment are subsequently withdrawn from the store and transferred to the rod assembly area.

Rod assembly commences by loading a pre-weighed pellet stack into an empty tube. Once the end components have been fixed, the tube forms a basic rod. Some fuel designs require stacks to include a mix of uranium and MOX pellets, and in these cases the uranium pellets will be manufactured and provided separately either by the customer or by BNFL Springfields.

There are no wet processes, and clean sintered residues are recovered, sampled and returned to the powder feed stage. All batches have unique identities and are produced in campaigns for each plutonium enrichment. High levels of shielding in MDF and SMP allow the plants to cater for higher burnup and longer aged plutonium oxide, and both plants can manufacture fuel containing up to 10% fissile plutonium.

2.2. Item counting areas
(see Fig. 1)

Rods emerging from stack insertion, tube pressurizing and scaling are followed and accounted for on an item basis, utilizing the unique identity marked on each rod. Rod assembly continues with inspection (including some rod destructive testing) and enrichment checking using an X ray fluorescence technique. Certified rods are inserted into a matrix, known as a magazine, which is transferred to a magazine store. Reject rods may be stored in designated magazines in a magazine store before being transferred to a route for recovery of the special nuclear materials, which can then be returned to the pelleting process area.

Full magazines are removed from the store and transferred to the fuel assembly area. The orientation within the loaded magazine matches the required orientation of the fuel rod array in the fuel assembly. Once aligned, the magazine's full load of rods can be pushed or pulled directly into the fuel skeleton. Other assembly components are secured to the skeleton, and the fuel is then inspected and transferred to a finished fuel store. Some MOX fuel designs may require uranium-only rods; again, these will be manufactured and provided separately either by the customer or by BNFL Springfields.
3. FEATURES OF THE FULL SCALE SELLAFIELD MOX PLANT

The MDF is housed in an existing building and has a design throughput of 8 tonnes of heavy metal per year (t HM/a) (approximately 20 PWR assemblies). SMP's scale of operation will require a new building, with civil construction programmed for 1994 and operation for 1997. The plant will be immediately adjacent to THORP and will share services and systems to form an integrated complex. The design throughput of 120 t HM/a will be achieved by more frequent process operation, parallel process lines and the ability to handle larger batch sizes. SMP multiprocess lines cater for fabrication of both PWR and BWR fuels.

The criteria for dose uptake have increased the automation of SMP in relation to MDF. Plutonium dioxide cans will be automatically transferred from the THORP store via enclosed ducting directly into SMP. Can emptying operations will be carried out automatically, as will the transfer of pellet trays into the pellet store and that of finished fuel into the fuel store.

The introduction of a powder bulking stage allows SMP to have reduced analysis requirements. Routine samples, taken from sintered pellets and recycled material, will be transferred to the Sellafield laboratories (including the Euratom on-site laboratory) using the THORP pneumatic transfer system.

All sintered scrap in SMP will be pneumatically transferred back for milling and homogenization in a recycle process. Pressed green pellet residues can be sintered and recycled by the same route. Reject rods will be transferred to a stripping and crushing area. Pellets will not normally be salvaged, since this would increase operator dose uptake and complicate traceability. Bagless transfers will be used extensively in order to reduce waste arisings, particularly at the powder feed stages.

4. MODERN MANUFACTURING

Manufacturing excellence techniques such as manufacturing resource planning (MRP2) and just in time (JIT) are in use at the LEU fabrication facility at Springfields and are being considered for the MOX fabrication facilities. These techniques, coupled with statistical process control (SPC) and a total quality management philosophy, all aim at increasing throughput, quality, flexibility, responsiveness and the elimination of all forms of 'waste'. The practical effects can be a shortening of lead times, faster changeover, reduction in buffer stocks, flexible campaign sizes, reduced stock, lower reject rates and higher process efficiency. Cumulatively, they can significantly reduce work in progress (hold-up) and cut the level of residue arisings and hence the uncertainty associated with their accountancy.

High levels of automation and manufacturing excellence will be complemented by nuclear materials control and accountancy (NMCA) arrangements which are non-intrusive and integrated closely with the plant control and material tracking systems.
Extensive material tracking, open to authentication, will help preserve the quality of these very valuable products by reducing the need to interrupt flow for physical verifications. Records are open to external independent audit and are maintained online in a fully computerized system. Arrangements to make these plants intrinsically safeguardable involve the provision of space for likely dedicated safeguards verification equipment. There are sufficient points in the natural flow of the process to site dedicated safeguards equipment so that their passive presence has no effect on the production lead time. The level of assurance generated by these safeguards enabling arrangements, together with other monitoring and authentication features, will reduce the likelihood of counter-flow movements or significant stock handling. Storage areas will be in frequent use and, consequently, the potential for containment by seals is limited.

5. A COLLABORATIVE PROCESS

Before the capital funding for SMP and MDF was approved, a prerequisite was to guarantee that effective material accountancy and control would be implemented. Catering for these requirements at the conceptual stage of these projects minimizes the potential for high cost and disruption arising from any retrofitting. Successful collaboration on MDF, THORP and OFC safeguards arrangements have demonstrated the benefits of a number of organizations working towards a common goal. SMP, whilst building on the sound MDF model, also incorporates lessons from THORP. The success of this collaborative approach stems from the following activities:

(a) Utilizing a multidisciplinary team of Company safeguards specialists and designers at the conceptual stage to consider requirements relating to NMCA, timeliness, continuity of knowledge and independent verification.

(b) Notifying the safeguards inspectorate of our intentions at an early stage and forming joint working groups and plenary meetings to co-ordinate the detailed arrangements years before plant operation. SMP, for example, was notified in 1993 (four years before operation), and formal meetings with Euratom began in February 1994.

(c) Employing an open policy for the provision of information far in excess of that required in the formal regulations.

(d) Providing continuous specialist staff support throughout the design and commissioning phases. This develops an in-depth understanding and provides stable lines of communication.

(e) Involving the material accounting staff and the production operators at an early stage.
The cornerstone of these activities is good interactive communication, carried out in a spirit of co-operation to build confidence and provide a high level of assurance and transparency. Transparency is further increased by

— providing data access and timely data transfer
— providing on-site laboratory and inspector facilities
— eliminating unnecessary security constraints
— educating all personnel involved by a system of training programmes.

6. MATERIALS CONTROL AND ACCOUNTANCY

Notwithstanding the needs of international safeguards, the plant operators have an obligation to control, protect and account for all nuclear material in their care. This stems from customer, regulatory and public acceptability requirements. Nuclear materials control and accountancy, whilst being of key significance, is one of a number of integrated measures employed by the operator to carry out effective and efficient plant management.

At the same time, it is essential to recognize the need to facilitate effective safeguards arrangements, which in turn will also depend heavily on high quality verifiable material accountancy. Such a high quality regime, utilizing joint Euratom and IAEA inspection, is applied to Sellafield's plutonium product stores. Continuity of knowledge from these stores guarantees verified, high quality materials accountancy on the plutonium feed to the MOX plants. The quality to the NMCA in the rest of the MOX process stems from containment and the use of complex software and flow control systems to provide a high level of traceability, transparency and timeliness.

The MDF process, being less automated and having lower throughputs, is verified by conventional safeguards techniques of flow verification and interim inventories. SMP, on the other hand, will employ many of the features found in THORP, such as near real time materials accountancy (NRTMA), direct hold-up estimation, secure containment and surveillance, provision for in-line NDA, DA monitoring and logging, signal branching and data authentication features. The safeguards authorities can utilize these features to develop a more appropriate, yet high standard of safeguards verification and material monitoring, which relies less on plant run-out and monthly interim inventories.

Both plants employ a system of subdivision into a number of balance areas. MDF has accountancy balance areas which reflect the physical layout of the plant and the requirement to support criticality clearance. SMP has eliminated the potential for criticality by designing its process equipment to be geometrically favourable in terms of size and shape, and will be divided into works accountancy areas (WAAs) to maximize control and to minimize inventory difference (MUF).
Conventional accountancy will be maintained using the MOX plant information computer (MPIC), which will mirror much of the data processing structures and functionality of the THORP chemical plant information computer (CPIC). MPIC transaction logs and running balances will be updated as each accountancy transaction is completed. All material transactions across WAA boundaries will naturally have a unique batch identification, many of which will travel with the material using 'bar code' type technology. The material tracking and control system activities will be verified against these bar codes as part of movement validation.

Running balances by weight (grams) of MOX, Pu, U and $^{235}\text{U}$ will be available for each WWA (and material balance area), and detailed running inventory records will be held for significant storage areas. Running inventories of the plutonium can store, the uranium drum store, the pellet tray store, the rod store and the fuel store will be recorded in detail. These data will include information on isotopics and include relevant trace data to track a 'batch' through its constituent parts, back to its powder parents. As with all running balance systems, nominal values will be utilized to maintain a true and timely reflection of physical reality until better data are available. This would typically apply to transactions and stocks awaiting analytical results. The correction system will use 'delete' and 'add' transactions (including such 'corrections' produced when nominals are updated). All transaction events will be date and time logged, and this will allow past points to be reproduced and will provide a full transaction correction history. Measurement error contributions will be recorded against each transaction for use in a measurement control programme.

Accountancy for stores will be relatively simple and founded on data from the automatic systems. The BNFL security physical protection systems utilizing containment and surveillance, amongst other systems, will prevent unauthorized access, and the material tracking system will provide full data on batches in stores and their locations. The automatic store keeping system for placing and retrieving batches will be verified against independent automatic identity (bar code) checks. All stores will be able to hold differing enrichment and can retrieve material selected at random for verification outside the store. Transfers in and out of the pellet tray store will be weighed, and the powder, rod and fuel stores will utilize batch data following.

Receipts from other BNFL plants of plutonium or uranium will maintain continuity of the batch data by utilizing the shippers' data (provided electronically). Verification of receipts will therefore be on a batch item basis. Any weight differences which arise when the powders are used will instigate a reconciliation process but will not generate shipper-receiver differences. Emptying of cans will be fully automatic and will be repeated until a can is accepted as being empty against the shipper's data criteria. Empty cans will be transferred directly into waste containers via a bagless transfer system. This is the main source of plutonium contaminated material and will be measured both by empty can weighing and by NDA measurements on the waste.
containers. It is not foreseen that routine powder sampling will be required in order to check powder quality and so, whilst powder sampling will be possible, it has not been designed as an in-line automated process.

The pellet process accountancy will be based on direct measurements of the input and output streams. Special care will be taken to account for uranium pellets/rods, especially at stack make-up and in the residue recovery route. Whilst pellets will not be strictly item followed, the pellet transfer system includes a number of weighing points to confirm that no pellets are dislodged or removed during automatic transfer. Material present as process hold-up will be minimized by clean-out (or extensive run-out) for annual inventory taking. Many of the plant vessels have direct weighing, but hold-up in equipment such as the mills and spheroidizers may require estimation by non-intrusive NDA techniques. Sintered MOX residues recovered by crushing and milling will be homogenized, weighed and sampled. The material balance in the pelleting and recovery area will be the target for application of NRTMA techniques.

The item accounting processing and storage areas for rods and fuel are again relatively straightforward. Nuclear transformations to account for isotopic decay are required before products can be dispatched. Customer accountancy, enrichment campaign balancing and regulatory reporting complete the accountancy facilities.

8. VERIFICATION

Features enabling effective and efficient safeguards have been considered (see Section 5 (a)), and, whilst discussions with the inspectorate have only just begun, it is possible to indicate some of the considerations:

(a) Design information verification of the basic technical characteristics and the plant design is anticipated, especially as many areas will have very limited human access.

(b) A substantial number of surveillance points are available, and double cabling will be used to cater for safeguards cameras, connected via a protected link to an inspector’s office.

(c) Space for siting possible NDA equipment will be provided:
   - at the entrance to the plutonium can store,
   - at the entry and exit points to the pellet tray store,
   - at rod loading into magazines,
   - on the magazine carrier in the rod store,
   - on the lifting gear in the fuel store.

(d) Access to data will be provided by a special user account; access will be given to tracking and weighing data; and there will be timely data provision by electronic transfer.
(e) Logging equipment will allow for independent processing of raw signals (e.g. from weighing).

(f) There will be access to NRTMA.

9. NEAR REAL TIME MATERIALS ACCOUNTANCY

BNFL began its development of NRTMA in the early 1980s and is now commissioning an operational system on the THORP CPIC. NRTMA relies upon the ability to determine the in-process plutonium inventory at frequent intervals during the operation of the plant. To be effective, either as an operator’s materials control technique or for safeguards, the method of establishing an in-process inventory must be rapid and reliable, and must intrude as little as possible into plant operation.

Conventional accountancy on its own cannot meet the timeliness goals for control of materials such as MOX without frequent intrusive plant shutdowns for inventory taking. The current approach favours the adoption of NRTMA to maintain operational flexibility. For SMP, the NRTMA system would reside on the MPIC and could use the statistical software modules already tested on THORP with minimal need for change.

NRTMA will utilize installed instrumentation to rapidly collect and process data and will include NDA measurements and mathematical models for hold-up in areas where materials cannot be directly measured. Inventory differences will be calculated for each balance period and subjected to statistical analysis. Sequential analysis over a number of balance periods will detect apparent discrepancies in near real time. Not only can NRTMA indicate events quicker than conventional accountancy, but it can also provide the operator with a powerful management tool to aid investigation of the possible cause(s) of the anomaly. The NRTMA technique is a powerful tool and can be applied to any large processing complex which requires frequent inventory monitoring.

10. CONCLUSIONS

The safeguards arrangements for THORP [1] and MDF were developed in an excellent spirit of co-operation between BNFL and Euratom, and the end result was a sound system which ensured control under international safeguards to provide a high degree of assurance. The challenge in SMP is to achieve the same high safeguards credentials and to have effective NMCA arrangements which derive maximum benefit from the high levels of automation and integrate transparently with modern flexible manufacturing techniques.
REFERENCES

DEVELOPMENT OF AN INTEGRATED SYSTEM FOR NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL AT JAERI

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Abstract

DEVELOPMENT OF AN INTEGRATED SYSTEM FOR NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL AT JAERI.

The paper describes the design concept and the current status of an integrated system for nuclear material accountancy and control which is under development at Japan Atomic Energy Research Institute (JAERI). JAERI has decided to update the current system for material accountancy and control and to develop the integrated new system with a consolidated database in order to augment transparency, credibility, promptness and flexibility of the system in accordance with the strengthening and streamlining of IAEA safeguards, to realize prudent control of obligations required by bilateral nuclear co-operation agreements, and to provide information on the physical protection, safe handling, property control and cost effective use of nuclear material, also for public relations. The system is composed of two workstations using the UNIX operating system — one for implementation and the other for development purposes — and many terminals, located at the headquarters, administrative offices, and research facilities and laboratories. The system is also connected with a mainframe computer. There are many files on the database to record inventory changes, book and physical inventories, and statistics on material balances. These files are controlled by a commercially available database management system which permits access to data on the files with a simple query language, spreadsheet type software or application programs.

1. INTRODUCTION

It is desirable for an organization’s system of accounting and control of nuclear material to provide more transparent, credible, prompt and flexible environments under a new safeguards regime with strengthening and streamlining measures. On the other hand, bilateral nuclear co-operation agreements require prudent control of obligations provided in the agreements, and the organization itself also needs to control nuclear material not only as property but also as regards its physical protection, safe handling and cost-effective use. Furthermore, the organization should pay attention to public relations. In order to respond to these various requirements in an efficient manner, the Japan Atomic Energy Research Institute (JAERI) decided to update the current system for materials accounting and control and to develop an integrated new system with a consolidated database usable for the purposes intended.
2. OUTLINE OF JAERI'S ACCOUNTANCY AND CONTROL

JAERI was founded on 15 June 1956 and was associated with the Japan Nuclear Ship Research and Development Agency from 31 March 1985. Currently it has five establishments: Tokai Research Establishment, Takasaki Radiation Chemistry Research Establishment, Oarai Research Establishment, Mutsu Establishment and Naka Fusion Research Establishment. Nuclear material was first received on 28 May 1957 for Japan’s first nuclear research reactor, JRR-1, which first achieved criticality on 27 August 1957. The current status of nuclear facilities at JAERI is summarized as follows:

*Research reactors:* JRR-1 (decommissioned in 1969), JRR-2, JRR-3, JRR-4, JPDR (under decommissioning), NSRR, JMTR, Nuclear Ship Mutsu (under decommissioning) and HTTR (under construction);

*Critical assemblies:* TCA, VHTRC, FCA, JMTRC and SCF (construction being completed as a part of the Nuclear Fuel Cycle Safety Engineering Research Facility, NUCEF);

*R&D facilities:* JAERI-Tokai R&D (one material balance area (MBA) with three inventory key measurement points (KMPs), which includes five special areas for obligation control and two KMPs composed of 30 and 11 internal sub-KMPs, respectively), JAERI-Oarai R&D (one MBA with five inventory KMPs, which includes three special areas for obligation control), JAERI-Naka JT-60 (one MBA with one inventory KMP) and outside facilities (located at Naka and Takasaki).

JAERI has nuclear material such as enriched uranium, plutonium, $^{233}$U, natural uranium, depleted uranium and thorium. The enriched uranium is divided into two categories: highly enriched uranium (HEU), with enrichment greater than or equal to 20%, and low enriched uranium (LEU) with enrichment less than 20%, controlled accordingly. At the end of March 1993, the total weight of nuclear material was 94 t, 96.6% of which was in Tokai Research Establishment, 0.7% in Oarai Research Establishment and 2.8% in Mutsu Establishment. The amount of nuclear material in other research establishments was negligibly small. The total number of batches was about 9000. During fiscal 1992 (from April 1992 to March 1993), JAERI received nuclear material 88 times and shipped 37 times. The number of inventory changes was about 1000, which includes transfer of nuclear material within an MBA.

Inspections by IAEA inspectors started on 3 December 1964, first to JPDR, and then were extended to other facilities. The INFCIRC/153 type Safeguards Agreement entered into force on 2 December 1977 and since then IAEA safeguards have been uniformly applied to all nuclear material at JAERI. During fiscal 1992, 120 person-days were spent on JAERI's facilities by IAEA inspectors, while 119
person-days were spent by inspectors of Japan's State System of Accounting and Control (SSAC) of nuclear material.

JAERI's system of accounting and control of internationally controlled nuclear material, other material and equipment (JSAC) has been established on the basis of the fundamental rules and regulations for accountancy and control of such materials (including nuclear material, other material and equipment), and has been implemented basically at each establishment. Responsibility for the accounting and control of the materials themselves rests upon a director who controls the facility where the materials are located. On the other hand, the Nuclear Fuel Division, in the case of Tokai Research Establishment, and the Accounts Divisions, in the case of other establishments, are responsible for administrative and safeguards related matters, and the Office of Nuclear Fuel and Radioactive Waste Management Programme of the Headquarters is responsible for planning and implementing JSAC as a whole, including acquisition of the materials.

3. CURRENT INFORMATION SYSTEM FOR JSAC

Internationally controlled equipment and material other than heavy water are controlled on a manual basis and the treatment of heavy water is similar to that of nuclear material. Therefore, only nuclear material is described hereafter.

Nuclear material is controlled manually in three establishments: in the case of Takasaki Radiation Chemistry Research Establishment and Naka Fusion Research Establishment, owing to the very small amount being handled; and in Mutsu Establishment, because no inventory change occurred. A single system is used for both Tokai Research Establishment and Oarai Research Establishment. Therefore the description of the current system is restricted to that of Tokai Research Establishment.

Only two facilities, FCA and NUCEF, have their own computerized systems for material accountancy and control. In other facilities nuclear material is managed on a manual basis. All facilities, however, report any changes in their inventory to a responsible office, the Nuclear Fuel Division, in three forms: a nuclear fuel receipt form, a shipment form and a form for transfer within the establishment. This office manually inputs relevant data from the documents into the central information system.

The current system works on a mainframe computer, FACOM M-780, dedicated to administrative purposes. The database (file) management system is the VSAM system and the database consists of one source file, which records inventory changes, and a few auxiliary files defining organization, codes and MBAs. The source data file consists of the following data elements:
VSAM REFERENCE KEY DATA
- Area No., Batch name, Nuclear material No., Element code, MBA, KMP, Job, Room, Regulation, Inventory change year-month-day-No., Type of record, Correction (official) year-month-day.

BATCH DATA
- Inventory change type for official use and Code for JAERI's use, Ownership, Receipt from/Shipmetn to MBA, Carrier, Material description with regard to physical and chemical forms, Containment and irradiation, Measurement basis code, Isotope code, Container for transport, Supplier country code for JAERI's use.

OBLIGATION CONTROL
- Country which transferred the nuclear material in question, Name(s) of country/countries which transferred nuclear material, equipment, or components used to produce the nuclear material in question. Other obligation, New/Old Japan–USA agreement, Neutron contribution to nuclear production.

ACCOUNTANCY DATA
- Number of items, Weight of element, Weight of fissile isotopes, Enrichment, Weight of plutonium produced, Weight of fissile plutonium produced.

BEFORE/AFTER CHANGE
- Batch name, Nuclear material No., Element code, MBA, KMP, Job, Room, Regulation.

DATA PROCESSING
- Input year-month-day-hour-minute-second, Correction (internal) year-month-day, Succeeding changes in the current period, Zero or non-zero inventory, Batch partition for measurement, Processing No.-year-month.

All these data, which relate to a batch, are given as input when the batch is received. The minimum data, however, are necessary when the batch is transferred within the research establishment.

As for the data elements cited above in the category of BEFORE/AFTER CHANGE, an accounting balance area similar to the MBA is explicitly established by duplicating the related input record and by using one record as a shipping record from the shipping area and the other as a receiving record at the receiving area. This procedure makes it easy to take a material balance over a specific accounting balance area.

Since there is only one major file on the database, as stated earlier, it is easy to maintain the database because inconsistencies in data between different files do not occur. As for an incorrect record caused by clerical errors, such a record is replaced by the correct one on the database. This means that the logic in a data processing program could, in general, be substantially simplified. Another feature is the erasure of the records of batches which no longer exist from the database once every few years, with the old file being kept on a magnetic tape. This makes it
possible to search a record efficiently because the number of records on the file is reduced.

Although the current system has many of the features mentioned above, it has become obsolete as regards the following points because it was designed and developed in the 1970s.

(a) **No consideration of bulk accounting:**

Material accountancy in JSAC is basically item accounting. Material can be traced back and forth on a batch basis following the changes of an attribute, e.g. location and material description. The material can be divided into several parts using the rebatching procedure, but the total amount of the material is assumed to be the same with or without measurements. Even if the material has a bulk form, the quantity of the material is not updated on a periodical basis by remeasurement because the quantity itself is very small. Such practices cannot be continued because the operation of NUCEF will start soon, when a large quantity of bulk material will be handled. Since NUCEF will measure the material, powder or ceramic, when received or dissolved, a shipper-receiver difference (SRD) occurs. As the physical inventory is required to be measured periodically, its value will change owing to measurement errors or for other reasons, resulting in a non-zero material unaccounted for (MUF) value.

(b) **No alternative method for the use of data:**

Access to the database can be gained only by a computer program. If a new request arises, then a new program must be developed. Furthermore, even if the access procedure is improved in future, it is not easy to obtain information on inventory or on statistics because some data processing is needed based on the inventory change file.

(c) **Reliance on documents:**

There are no files on the database for inventory change reports (ICRs), physical inventory listings (PILs), material balance reports (MBRs) and obligation control reports (OCRs). These reports are kept in a document. Another case is a record which has some errors in it. Therefore, paperless, full automation is not attainable.

(d) **No consideration of networking:**

The use is restricted to the staff of the Nuclear Fuel Division (and of the Oarai Accounts Division). It is desirable, however, to have a network with the systems of facilities and with Headquarters' personal computers (PCs) making a direct access possible.

(e) **Others:**

Only one staff member of the Information System Centre of JAERI has been dedicated to the development of a program and the maintenance of the current system. As a result, the documentation of the system is not adequate and there will be difficulties in taking over his duties when he retires. Also, the current
system shows limited flexibility when an improvement is attempted in response to various new requirements.

4. CONCEPT OF AN INTEGRATED SYSTEM FOR NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL (NEW SYSTEM)

The fundamental concept of an integrated system for nuclear material accountancy and control is to use the favourable features of the current system as far as possible and to resolve all the problems mentioned in Section 3.

The new system is composed of two workstations using the UNIX operating system, one installed in the Nuclear Fuel Division for implementation and the other installed in the Safeguards Technology Laboratory for development purposes, with many terminals which are PCs or workstations located at Headquarters, administrative offices, and research facilities and laboratories. These workstations and terminals are connected with one another by principal and local network systems and are also connected with a mainframe computer. The configuration of the system is shown in Fig. 1.

The implementation environment is separated from the development environment first so that the implementation is not disturbed by the development work, which, it is estimated, will last a few years, and secondly so that the development can be freely carried out without any consideration of the implementation. The new system is connected with the mainframe computer, first because the current system is working on the mainframe and the initial load data should be transferred from it; secondly a fast line printer can be used for large amounts of printout; thirdly the FCA's system works on it; and finally because we want to use it for the long term storage of archive records.

At JAERI, there is a wide range of facilities and laboratories, in terms of size, and of nuclear materials with many chemical and physical forms and with various qualities and quantities. Since it is difficult to standardize all source data in a form, we have defined two levels of source data. The first level of source data consists of miscellaneous information and is kept at a facility or a laboratory on its PC or workstation. The second level of source data is composed of information with standardized formats or with a document-type form including tables and figures. These are all registered on the new system at the Nuclear Fuel Division, or are reported to the new system through the network system from the facility or laboratory system, making almost real time accountancy and control possible. It is noted that the standardized forms of information are the improved versions of the nuclear fuel receipt form, shipment form, and the form for transfer within the research establishment, which was designed so as to fulfil the Safeguards Agreement, bilateral agreements and other obligations.

The database management system is a commercially available system called Informix. Since the NUCEF's own system works on a workstation with the database
FIG. 1. Network of computers and terminals for JAERI's integrated system of accountancy and control of nuclear material, other material and equipment.

managed by the same Informix, direct access to the NUCEF database is possible from the new system. Using Informix together with other software packages, quick and easy access to various data on the database is made possible, for example, by a simple language, SQL, spreadsheet type software, Wingz, and by application programs which are written in C and SQL.

The database consists of such files as source data files for inventory changes and physical inventories; report files for an ICR, PIL, MBR, OCRs, material and equipment reports and report headers; inventory and statistics files for book inventories, monthly inventory listings and monthly material balances; and other files, e.g. for error records, design information, organization and for image data.
Data elements, a subset of which constitutes a file and should be stored on the database, have been re-examined in comparison with those of the current system. Deletion from and addition to those data elements are as follows:

**DELETION**
- Carrier, Input hour-minute-second, Processing No.-year-month

**ADDITION**
- Organization, Facility, Period covered by report from—to, Date of inventory, Number of line entries for accountancy data and concise note data, Signature, Type of report or file, report No., entry No., Continuation, ICR/PIL Report No. and Entry No. referred to, Entry name, Origin of material, Unit, Concise note code, Correction to — Report No. and Entry No., Record No., Category of statistic, End use code, Number of batches, Weight of element reported, Total weight of element over a batch, Weight of fissile isotopes reported, Total weight of fissile isotopes over a batch, SQ, Ekg (effective kg), Pu status — separated or contained.

As in the current system, material accountancy and control is carried out on a batch basis with batch follow-up possible. Additional procedures applied to the bulk material of NUCEF are as follows: SRDs are added as inventory change. For the bulk material which exists in tanks and in the process area, a batch, called a bulk batch, is defined so as to cover the whole material. Any increase and decrease of nuclear material in the bulk batch is followed up by the rebatching procedure. When a physical inventory is taken, the inventory is measured for each tank which constitutes a batch for PIL, called a tank batch. Since the tank batch cannot be followed up with regard to the quantity of nuclear material, this record is kept on a different file, the source-data-2 file, instead of the source-data-1 file for inventory changes. For the bulk batch, a MUF value is given to the system as another inventory change. In keeping a book inventory of the bulk batch, the MUF value is reduced from the book inventory, converting it to the physical inventory. In order to avoid any inconsistencies, the system automatically compares the physical inventory thus obtained with the summation of inventories over the tank batches.

As for the obligation control, an adjustment, called an $M^+/M^-$ procedure in the SSAC of Japan, is made to the weight of element which corresponds to an obligation if two batches of different isotopic compositions with different obligations are completely mixed as a solution batch. This procedure is needed because the weight of fissile isotopes is conserved for each obligation before and after mixing. SRDs and MUF are automatically divided into fractions in proportion to the obligation-wise quantities of nuclear material in the batch related to a SRD or in the bulk batch in the case of MUF. In the case of the tank batch, such division is not required. Instead, the obligation-wise inventory over an MBA, which is obtained from the book inventory of the bulk batch, is requested to be reported to the national authority on an OCR form.
Input to the new system can be prepared with minimum effort by opening windows and selecting a suitable one among a list of codes and parameters. If a facility or a laboratory has its own system and can provide the input, then it is possible to send it directly to the new system. If the receipt, shipment or transfer form is prepared as a document, a responsible person of the Nuclear Fuel Division prepares the input based on the document.

Output from the new system is such reports as ICRs, PILs, MBRs, OCRs, book inventory listings and other reports used for internal management or for safeguards purposes. An interface between the new system and the in-field support system (IFSS) of the IAEA may be provided through a floppy disk. As for the image data, the new system can provide us with a photographic view of the research establishment, a facility, its layout, drawing of a fuel structure, a table or figure of statistic on nuclear material, an outline of safeguards and so on. This information can be used for visual presentation of the new system, with the aim of helping the staff engaging in material management and the general public who visit JAERI to better understand the material accountancy and control at JAERI.

For the system backup, the same database is maintained in both workstations and a dump of its contents is carried out periodically.

5. CURRENT STATUS OF THE SYSTEM DEVELOPMENT

Two workstations, Fujitsu S-4/10 model 40, have been installed at the Nuclear Fuel Division and the Safeguards Technology Laboratory, respectively. Fundamental software such as the operating system, Solaris, Japanese version 1.1, the database management system, Informix, and the spreadsheet-type software, Wingz, has been installed and the database has been constructed by defining its structure and by transferring the initial data from the current system. As for the application programs, basic programs for input, quality control and output have been developed, as well as programs for the system presentation. These programs are being used on a test basis.

It is planned that a part of the new system should go into the implementation phase around October 1994 while the development of additional functions is continued.

6. CONCLUDING REMARKS

We have started a long term project to develop an integrated system for material accountancy and control with the aim of making the system more transparent, credible, quick to respond, flexible in developments and efficient in staff utilization. As the first phase of the project, we have been developing a network system
connecting headquarters, administrative offices and facilities and laboratories. We believe that the system, if completed, will make the management of nuclear material more transparent to everyone and also make it more flexible to respond to requests from the safeguards authorities. As the second phase, we shall incorporate in the system such functions as acquisition control, proprietary control and document control, together with demonstration and training as an integrated part of the system.

Although the adoption of a new system concept provides us with possibilities to do many new things, it creates new problems. For example, having many interrelated files on the database makes it necessary to enhance the quality control function in order to avoid any inconsistencies among data on these files. This is true especially if a correction to data is carried out. Another example is that, since the system is made accessible to more users, access control for an individual user should be strictly observed. We consider that these problems are inevitable, and appropriate measures should be applied in each case.
LA GESTION INFORMATISEE DES MATIERES NUCLEAIRES AU CENTRE D'ETUDES NUCLEAIRES DE GRENOBLE

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Abstract—Résumé

COMPUTERIZED MANAGEMENT OF NUCLEAR MATERIAL AT THE GRENOBLE NUCLEAR RESEARCH CENTRE.

Under French law the production, possession, transfer, use and transport of nuclear material are subject to prior authorization by the Ministry of Industry and to control with a view to preventing losses, theft or diversion of such material. The holder of the authorization is required, in particular, to take measures for the monitoring and accounting of the nuclear material. For this purpose, the Grenoble Nuclear Research Centre, in collaboration with the Central Directorate for Safety and the Institute for Nuclear Protection and Safety (IPSN) of the Commissariat à l'énergie atomique, has designed a computerized network which can be used for automation and centralization of nuclear material monitoring and accounting at the Centre and for interfacing with the IPSN's national centralized accounting system. The paper describes this network with the possibilities of its use in real time and emphasizes the advantages offered by automated centralized nuclear material management.

LA GESTION INFORMATISEE DES MATIERES NUCLEAIRES AU CENTRE D'ETUDES NUCLEAIRES DE GRENOBLE.

La réglementation française impose que l'élaboration, la détention, le transfert, l'utilisation et le transport de matières nucléaires soient soumis à une autorisation préalable délivrée par le Ministère de l'industrie et à un contrôle dont l'objet est d'éviter les pertes, vols ou détournements de ces matières. Pour le titulaire de l'autorisation, les obligations comportent notamment des mesures de suivi et de comptabilité des matières nucléaires. A cet effet, le Centre d'études nucléaires de Grenoble a élaboré, en collaboration avec la Direction centrale de la sécurité et l'Institut de protection et de sûreté nucléaire (IPSN) du Commissariat à l'énergie atomique, un réseau informatisé permettant l'automatisation et la centralisation du suivi et de la comptabilité des matières nucléaires au niveau du Centre, et d'assurer l'interface avec le système national de centralisation de la comptabilité de l'IPSN. L'objet de la présente communication est de décrire le réseau réalisé avec ses possibilités d'utilisation en temps réel et de souligner les avantages de la gestion informatisée centralisée des matières nucléaires.
1. INTRODUCTION

1.1. Aspect réglementaire général

La réglementation en vigueur en France impose que l'élaboration, la détention, le transfert et l'utilisation des matières nucléaires soient soumis à une autorisation préalable délivrée par le Ministère de l'industrie ainsi qu'à un contrôle par l'exploitant et les pouvoirs publics dont l'objet est d'éviter les pertes, vols ou détournements de ces matières. Pour le titulaire de l'autorisation, les obligations comportent essentiellement des mesures de suivi et de comptabilité des matières nucléaires détenues.

A cet effet, des dispositions spécifiques doivent être prises afin de:

a) Connaître de façon précise, en quantité et qualité, toutes les entrées et sorties de matières nucléaires de l'établissement ou des installations.

b) Assurer le suivi des matières nucléaires présentes à quelque titre que ce soit dans l'établissement ou dans les installations, c'est-à-dire connaître leurs localisations, usage, mouvements et transformation.

c) Décéler sans délai les anomalies éventuelles concernant le suivi des matières nucléaires et transmettre aussitôt l'information au Ministre de l'industrie.

d) Vérifier par des inventaires périodiques que la situation physique des matières nucléaires détenues est conforme à la comptabilité tenue dans l'établissement et, en cas d'anomalie, transmettre aussitôt l'information au Ministère de tutelle.

e) Prévenir immédiatement les services de Police ou de Gendarmerie lorsque les matières nucléaires paraissent avoir été volées, perdues ou détournées.

Enfin, le suivi et la comptabilité des matières nucléaires doivent être organisés de manière à permettre au Ministre de tutelle d'en vérifier l'efficacité et la fiabilité, de centraliser la comptabilité des matières et, le cas échéant, d'être informé sans délai de la nature et de la quantité des matières manquantes.

1.2. Aspect réglementaire fixant les conditions techniques du suivi et de la comptabilité des matières nucléaires

L'autorisation délivrée par le Ministère de l'industrie fixe, en tant que de besoin, les ensembles et sous-ensembles techniques qui doivent être différenciés pour

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1 Loi n° 80.572 du 25.07.1980 sur la protection et le contrôle des matières nucléaires.

2 Décret n° 81.512 du 12.05.1981 relatif à la protection et au contrôle des matières nucléaires.

3 Arrêté du 24 juin 1982 fixant les conditions techniques du suivi et de la comptabilité des matières nucléaires.
les besoins de la comptabilité à l'intérieur d'un établissement ou d'une installation. Ainsi les enregistrements comptables sont établis séparément pour chacune des catégories de matières nucléaires suivantes:

- thorium,
- uranium appauvri,
- uranium naturel,
- uranium enrichi à moins de 10% en uranium 235,
- uranium enrichi à 10% ou plus mais à moins de 20% en uranium 235,
- uranium enrichi à 20% ou plus en uranium 235,
- uranium 233,
- plutonium.

La comptabilité des matières doit comporter l'enregistrement chronologique sur un livre journal ou tout autre support équivalent, de chacune des variations affectant quantitativement le stock de matières nucléaires ou sa répartition dans les catégories explicitées ci-dessus, à savoir:

- Mouvement externes: réception et expédition.
- Opérations internes: utilisation, mouvement et transformation.
- Corrections et ajustements, résultant de mesures, de calculs et estimations.
- Ecarts constatés en cours d'inventaires physiques.

Les variations de stock doivent être enregistrées le jour même où elles ont eu lieu ou ont été déterminées. Le stock comptable en fin de mois est calculé et enregistré pour chacune des catégories de matières énumérées ci-dessus. Les informations minimales suivantes doivent être enregistrées pour chacune des variations:

- date de la variation et de la prise en compte;
- date de correction (écriture modifiant une écriture précédente);
- nature de la variation enregistrée;
- quantité de matières nucléaires;
- quantité d'uranium 235 (pour l'uranium enrichi seulement);
- forme physico-chimique;
- identification du lot et nombre d'articles contenus, le cas échéant;
- identification de la campagne de traitement, le cas échéant;
- référence des justifications techniques telles que fiches de pesée, procès-verbaux d'échantillonnage, fiche d'analyse, enregistrements d'appareils de mesure, calculs d'établissement des productions ou des consommations, programme de contrôle de la qualité des mesures utilisées.

Enfin, le titulaire d'une autorisation doit transmettre les données comptables relatives aux variations affectant le stock de matières nucléaires qu'il détient à l'IPSN.
2. DESCRIPTION DU RESEAU ELABORE

2.1. Généralités

A partir des conditions techniques du suivi et de la comptabilité des matières nucléaires rappelées en 1.2., le Centre d'études nucléaires de Grenoble (CENG) a élaboré en collaboration avec la Direction centrale de la sécurité et l'IPSN, un réseau informatisé permettant l'automatisation et la centralisation de la comptabilité des matières nucléaires dénommé réseau MANU (MAtières NUcléaires) dont l'objet est :

a) la comptabilisation des mouvements par matière, par unité comptable appartenant à un établissement ou une installation ;

b) l'enregistrement chronologique des mouvements ;

c) l'édition de la fiche de stock matière établie pour les différentes entrées et sorties ;

d) l'inventaire comptable d'une unité pour le contrôle national ;

e) l'inventaire mouvement et stock final pour le contrôle Euratom ;

f) l'édition de différentes listes de la base de données : unités comptables, comptes extérieurs, produits, unités étrangères, annuaire des exploitants, les différents types de mouvements et la concordance entre les codes d'engagement de contrôle ;

g) l'édition du livre journal.

2.2. Schéma synoptique du réseau MANU

Le réseau MANU est conçu autour d'un serveur de fichiers et de communications géré par l'administrateur de réseau. La figure 1 fournit la configuration générale adoptée. Les stations locales implantées dans les unités détentrices de matières nucléaires du Centre sont connectées sur une étoile optique reliée au serveur de fichiers. L'administrateur de réseau transmet quotidiennement les mouvements effectués à l'Institut de protection et de sûreté nucléaire. La configuration spécifique à la base de données centralisée est représentée sur la figure 2.

2.3. Le Bordereau de Déclaration d'Opérations sur les Matières Nucléaires (BDOMN)

Toutes les informations concernant les mouvements de stocks de matières nucléaires devant être obligatoirement transmises à l'Institut de protection et de sûreté nucléaire (voir paragraphe 1.2.), un document standard multi-fonctions est imposé, document appelé BDOMN édité par le logiciel de gestion des matières
FIG. 1. Réseau MANU.
FIG. 2. La base de données.
nucléaires (MANU) pour chaque type d’opérations effectuées regroupées en six familles:

a) expéditions de matières nucléaires,
b) réceptions de matières nucléaires venant de l’étranger ou d’un compte extérieur français,
c) accusé de réceptions de matières nucléaires,
d) variations de stocks,
e) mises aux déchets de matières nucléaires avec possibilité d’actions correctives relatives aux annulations pures et simples de déclaration, aux remplacements de déclarations ainsi qu’aux annulations et remplacements d’anciennes déclarations.

2.4. Exemple d’application

Les tableaux I et II donnent des exemples d’application de comptabilité centralisée et, notamment, une opération de réception de matières nucléaires par le CENG ainsi que l’édition de l’enregistrement chronologique des mouvements de matières au sein d’une unité de gestion du Centre.

2.5. Constitution du réseau

Le réseau est de type Ethernet géré par 3 + Open sous système OS/2. Les liaisons entre les unités détentrices de matières et le serveur de fichiers est un AT 486 avec 9 Mo de mémoire additionnelle. Le logiciel MANU est développé en langage Nantuket avec un compilateur Dbase.

3. CONCLUSIONS

L’utilisation d’une gestion informatisée des matières nucléaires permet au responsable d’un établissement de connaître en temps réel la situation comptable par matière, par produit et par installation (en situation normalé et en cas de crise).

Le réseau MANU constitue un outil informatique utile aux gestionnaires de matières pour enregistrer tous types de mouvement, éditer les documents officiels s’y rapportant, effectuer les contrôles obligatoires et uniformiser les documents comptables.
**TABLEAU I. EXEMPLE D'APPLICATION DE COMPTABILITE CENTRALISEE.**

**OPERATION DE RECEPTION DE MATIERES NUCLEAIRES**

Feuillet / Bordereau de déclaration d'opération sur les matières nucléaires

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<th>N° Bordereau</th>
<th>Emission</th>
<th>Opération</th>
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<th>Modifie le bordereau</th>
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<th>Destinataire</th>
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<td>Des Co Code Date J</td>
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**Transport**

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</thead>
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</tr>
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</table>

**Comptabilité**

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**Gestion**

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<th>Fournis</th>
<th>Client</th>
<th>RD</th>
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</thead>
<tbody>
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</tbody>
</table>

**Commentaire émetteur bordereau**

17 élémens reçus n° 1 A-17, pas d'erreur par rapport aux données de l'expéditeur
### TABLEAU II. MOUVEMENTS DES MATIERES NUCLEAIRES DU 01/01/93 AU 31/12/93 (POUR ZBM EGALE A FSLE)

<table>
<thead>
<tr>
<th>BDOMN n°</th>
<th>Date émission</th>
<th>Matière Code CEA</th>
<th>Date de mouvement</th>
<th>Code Mvt.</th>
<th>Masse élé.</th>
<th>Masse iso.</th>
<th>Expéditeur destinataire</th>
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<td>24/06/93</td>
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<td>08/07/93</td>
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1. INTRODUCTION

The United Kingdom Atomic Energy Authority (UKAEA), now trading as AEA, has functioned for over 40 years as the organization principally responsible for research and development of nuclear energy in the UK. In that time, it has built up expertise in all aspects of the fuel cycle, from thermal to fast reactors, reprocessing, enrichment, fuel fabrication, residue recovery and waste management. In recent years the organization has undergone major changes in its function and structure, in response to the slowing down of the development of nuclear energy and reduction in government funding. As a result, AEA has found it necessary to move further into commercial activities, accelerating a process which started several decades ago. Many laboratories throughout the world now find that they are also facing similar trends towards greater commercialization. We summarize here the experience of AEA in continuing to meet its safeguards obligations under the new circumstances.

2. SAFEGUARDS IN THE RESTRUCTURED AEA

The present structure of AEA includes a number of businesses, in two groups, nuclear and non-nuclear, supported by a corporate services organization, in a number of locations throughout the UK. It is planned to restructure AEA into three divisions: Government Division, Commercial Division and Services Division. The Services Division will exist for one year only, during which time its functions have to be subsumed in the other two divisions or divested to the private sector. This latest change is being made in readiness for a possible privatization of the commercial part.
The Government Division will be responsible for most of the nuclear material currently owned by AEA, and its main function will be to manage the historic liabilities resulting from the development of nuclear energy by the UKAEA.

Supervision of nuclear materials accountancy, control and safeguards is currently carried out by Nuclear Materials Management Division (NMMD) within the Corporate Safety Directorate; the central Headquarters of NMMD are at Harwell, and there is a department on each nuclear site (Dounreay, Windscale, Harwell and Winfrith) with responsibility for monitoring and servicing local activities. This arrangement ensures that the NMMD personnel responsible for monitoring nuclear materials accountancy and control in AEA are independent of the personnel having custody of, and using, nuclear materials. NMMD is also responsible for the day-to-day management of the UK Safeguards R&D Programme in support of IAEA safeguards [1], and both this Programme and the corporate activities derive benefit from the common management.

2.1. Meeting safeguards obligations

An important aspect of the new structure is the establishment of a strong means of meeting the safeguards obligations. Nuclear materials accountancy and control have many parallels with safety management in the way they are regulated and implemented. The emphasis of the new arrangements in AEA is on supporting the businesses so that they can operate in a commercial environment, in which materials are controlled in a proper manner at a cost that does not hamper successful commercial operation.

Comprehensive documentation on accountancy and control procedures is the basis of the system, complemented by quality assurance. Systems audit is used as a key tool for ensuring that accountancy and control procedures are adequate to meet the statutory requirements; recommendations from audits are followed up to ensure that they have been implemented. These measures help to give AEA customers confidence that nuclear material entrusted to the AEA will be satisfactorily looked after.

2.2. Measuring and improving performance

NMMD has defined performance parameters so that improvements in the service provided can be demonstrated by comparison with quantitative scales. In the context of nuclear materials, the parameters include material unaccounted for (MUF) by the material balance area and the material type. Business and financial efficiency are covered by other parameters.

For many years, AEA, together with British Nuclear Fuels plc (BNFL), has published its annual MUF figures for each form of material at each site. This policy was established in order to demonstrate the overall outcome of nuclear materials accountancy and control in the two organizations. While the figures gave rise to comments by the media in the early years, there has generally been little reaction in
recent years. Indirectly, this custom of publishing the MUF data could well have provided an additional incentive to achieve good accountancy.

2.3. Computerized nuclear materials accountancy systems

In view of the past programmes of the UKAEA, computerized nuclear materials accountancy systems have evolved to meet the individual needs of the various sites. Consideration is now being given to investigating possible economies in the cost of running a number of systems, particularly as the nature of the nuclear work changes and its volume is reduced.

2.4. Fuel cycle facilities

A new plant, the mixed oxide (MOX) fuel demonstration facility [2], a joint BNFL–AEA project, is currently being commissioned, and discussions on the safeguards approach, involving the relevant UK Government Department, have been held with Euratom. These discussions began two years prior to active commissioning, thus meeting the recent amendment to Commission Regulation (Euratom) 3227/76, requiring notification of the basic technical characteristics at least 200 days before the first consignment of nuclear materials was due to be received. This co-operation between plant operators and the regulatory authorities demonstrated the value of early discussions to both sides.

Owing to the complex nature of some fuel cycle operations, accountancy anomalies occur occasionally. It is important that these are detected at an early stage so that special investigations can be set up, the problems can be identified and corrective action can be taken quickly. Then, information on the experience gained is disseminated so that other facilities can benefit from this.

On the technical front, while many of the accountancy measurements in the fuel cycle are based on destructive analysis, there is increasing reliance on non-destructive analysis, where necessary and appropriate. There is emphasis, particularly in reprocessing plants, on control of scrap and waste, so that the amounts and therefore the cost of recovery and disposal, as well as uncertainty within the accounting scheme are minimized.

Good accountancy is central to the control of nuclear materials, but complementary physical security measures are also important in providing additional assurance. Close liaison is maintained with those groups responsible for security within the AEA.

REFERENCES


1. INTRODUCTION

The feasibility of using the running book inventory (RBI) as a near real time indicator of loss or diversion of plutonium fuel from a nuclear fuel reprocessing plant has been of interest in safeguards for some time. A prior study determined that the RBI, when adjusted by a measurement of a partial inventory (which is highly variable yet conveniently measured), can be used to develop viable indicators. An application of particular interest is to the head-end of a large scale reprocessing plant. In a recent study, this problem was explored in depth, using a flow sheet typical of such a facility.

The acronym for the running book inventory of a material balance area adjusted by a measured partial inventory is ARBI. The partial inventory measured is presumed to be that which is highly variable but conveniently measurable. Consequently, even in the presence of no measurement bias and short of total clean-out, the ARBI has a non-zero expected value and a variance characterized by the process variation and the imprecision of the measurement systems. For selected indicators based on the ARBI, statistical tests have been derived and their properties explored as they relate to IAEA sensitivity and timeliness criteria.
A previous study [1] indicated that, dependent upon the variability of the measured inventory, a considerable fraction of the total inventory can be left unmeasured without a substantial loss of test detection sensitivity when the loss indicators are carefully selected so as to attenuate the process variance of the unmeasured inventory. As an example, a statistic as simple as the difference between the averages of the lower third and the top third of a sequence of ARBIs was found to be particularly effective. This indicator, labelled the B-Statistic and patterned after Bartlett's test, has other desirable properties and, when it is applied to rolling datasets, it can be applied to both protracted and abrupt diversion scenarios. The simplicity of the statistic, together with the ease of application, and the fact that the test can be started or restarted anywhere in the sequence make it particularly attractive as a loss indicator in an ARBI sequence.

Some additional conclusions reached in earlier studies involving loss indicators based on the ARBI are the following: (1) There is no 'one best' indicator; the choice of indicators in a given application depends in part on the magnitudes of the process variance and the measurement error. (2) There is a trade-off between detection time and loss rate for a protracted diversion that is 'detectable', i.e. it takes longer to detect a small diversion rate than it takes to detect a large one. (3) A plant with large systematic measurement errors will require frequent inventories to meet the IAEA sensitivity criteria. (4) A key element in deriving appropriate statistical tests is the assumption made in relation to the correlations that exist between elements of the database. (5) Application of derived tests to historical databases, such as those reported at Dounreay [2], suggests that systematic errors (correlations) are often of very short duration and thus should not be propagated as long term errors unless there is considerable justification for doing so [3].

2. DISCUSSION

A description of the head-end of a typical large scale reprocessing plant was developed from unpublished data obtained from France, Japan, the US Nuclear Regulatory Commission and the IAEA [4]. This conceptual facility utilizes one continuous dissolver of the type developed in France, has a nominal throughput capability of 4 t U/d for PWR fuel, and has input accountability tank (IAT) batch sizes that are believed to be similar to those being used in, or planned for, the Hague and the Rokkasho Mura reprocessing plants. This gave an IAT weekly output of six to seven batches containing about 274 kg Pu and a measured inventory of about 23 kg Pu,
which varied from 3 to 38 kg Pu. The unmeasured inventory was a nominal 5 kg Pu, with a process standard deviation of about 0.5 kg Pu (10%). Thus, the inventory of the head-end area is much smaller than that used in studies of the process area. This artifact of the plant design caused the variance of ARBI due to the inventory to be very small compared with that of the transfer measurements. Thus, the capability of the B-Statistic to attenuate the process variance of unmeasured inventory could not be fully realized. Consequently, it was necessary to concentrate on other indicators and means to reduce the effects of the large variances in the reactor calculated inputs to the head-end area.

The Pu/U ratio method [5] was used to distinguish between an SRD and an MUF in the head-end area for plutonium based on the hybrid К-edge densitometer (HKED) [6] measurement at the IAT applied to the reactor calculated uranium values for each fuel assembly entering the chopping cell. Thus, the head-end area for plutonium was split into two balances: (1) the SRD balance between the reactor calculated plutonium and the Pu/U ratio estimate of plutonium entering the chopping cell, and (2) the ARBI balance between the Pu/U ratio estimate of plutonium entering the chopping cell and the IAT measurement. This resulted in a substantial reduction of the effect of measurement errors, particularly the systematic error that is typical in burnup calculations, on the detection sensitivity for the ARBI material balance indicator.

A series of parametric studies were applied to the uranium and plutonium ARBI data to determine whether the ARBI-Statistic (comparable to a CUMUF test), the D-ARBI-Statistic (based on differences) and the B-Statistic could detect a diversion within two different IAEA timeliness criteria (50% probability of detecting a diversion of 8 kg Pu, or 800 kg U, within 4 weeks or, alternatively, within 12 weeks) [7]. The parameters included inventory frequency, ARBI data subset lengths, input measurement methods, correlation assumptions, detection criteria, size of the unmeasured inventory, and increases in the process variance of the unmeasured inventory. The SRD-Statistic was simulated for the plutonium balance in accordance with the IAEA criteria to evaluate whether the sum of all SRDs during the material balance period was greater than 0.1 significant quantity.

The correlation assumptions with regard to the individual measurements have a substantial impact on the apparent sensitivity of a test utilizing ARBI data. Improved equations for calculating the systematic error variance of ARBI were developed on the basis of the number of periods that are correlated for individual input, output and waste measurements. All simulations and associated tests presumed that only measurements within a given period (one week) are correlated. This choice was made to correspond to essentially one reactor discharge batch being processed in each period. This study describes a procedure for testing an observed ARBI data sequence to determine whether the correlation assumptions are realistic. An approximate method for estimating multiple test thresholds is described which does not require the Monte Carlo technique.
3. CONCLUSIONS

As a result of the study, it was determined that:

(a) A test based on uranium ARBI is the most sensitive indicator of loss/diversion and should be given high priority in the material balance evaluation for the head-end of a large scale reprocessing plant, since the plutonium cannot be separated from the uranium at this stage of the process.

(b) For plutonium, the use of the Pu/U ratio method for determining the input to the chopping cell improves substantially the sensitivity of the B-Statistic test, the ARBI-Statistic test and the D-ARBI-Statistic test to detect a loss or diversion.

(c) The uranium and plutonium ARBI balances and related tests should be consistent with each other and would differ only if the uranium or plutonium HKED measurements were preferentially biased with regard to each other; this is an unlikely event if the isotope dilution mass spectrometry method is used for calibration and verification of IAT batches.

![Graph](attachment:image.png)

*FIG. 1. B-Statistic test, ARBI-Statistic test, D-ARBI-Statistic test and SRD test as a function of time for the model facility with abrupt diversion of 8 kg Pu in period 13 (multiple test thresholds).*
(d) For plutonium, the SRD test, based on the reactor calculated input to the spent fuel pool minus the Pu/U ratio estimate of plutonium entering the chopping cell, provides an effective method of isolating input biases from potential head-end losses.

(e) A combination of tests are needed to protect against all types of diversions:

- The ARBI-Statistic test for uranium and plutonium provides the best protection against diversions initiated in the early periods;
- The B-Statistic and the D-ARBI-Statistic tests provide the best protection for diversions initiated in later periods;
- The B-Statistic test will provide better protection than the D-ARBI-Statistic test when inventory and process variances become larger compared to transfer variances;
- The SRD test for plutonium will isolate reactor calculation bias; as long as the uranium ARBI is not significant, a significant plutonium SRD is of little consequence.

Figure 1 is an example of all four tests with abrupt diversion of 8 kg Pu in period 13, using the approximate procedure for determining the multiple test thresholds. For the SRD test, zero bias in the reactor calculated input for plutonium was assumed.

REFERENCES


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SARP-II:
SAFEGUARDS ACCOUNTING AND
REPORTS PROGRAM, REVISED*

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1. INTRODUCTION

A computer code, SARP (safeguards accounting and reports program), which will generate and maintain at-facility safeguards accounting records, and generate IAEA safeguards reports based on accounting data input by the user, was completed in 1990 by the Safeguards, Safety, and Nonproliferation Division (formerly the Technical Support Organization) at Brookhaven National Laboratory as a task under the United States Programme of Technical Assistance to IAEA Safeguards [1]. The code was based on a State System of Accounting and Control (SSAC) of nuclear material for off-load refuelled power reactor facilities, with a model facility and safeguards accounting regime as described in the IAEA safeguards publication STR-165 [2]. Since 1990, improvements in computing capabilities, as well as comments and suggestions from users have engendered revision of the original code. The result is an updated, revised version, called SARP-II.

* Work performed under the auspices of the United States Programme of Technical Assistance to IAEA Safeguards (POTAS).
SARP-II should be most useful to States party to the NPT, with small to medium nuclear activity and who wish to transmit IAEA reports on computer media. The user is assumed to be completely familiar with IAEA (NPT) safeguards and with nuclear materials accounting, record keeping and reporting as implemented at the user facility.

The program consists of a database foundation, with additional codes for user interface, file management, printing, etc., and was originally written in dBase III+, compiled with CLIPPER and based on guidelines given in the IAEA Safeguards Information Treatment PC Systems Development Handbook. SARP-II uses CLIPPER 5.01. Minimum computer hardware required for using SARP-II is an IBM PC (or compatible) with at least 512 kb of RAM, a hard disk, a floppy disk drive, a colour monitor and a printer. The User Manual [3] and the Technical Reference Manual [4], prepared for SARP, are being updated.

Most of the principal functional and organizational features of SARP-II are identical with the original, i.e. the system is divided into seven main modules:

— Tutorial
— Initialization
— Accounting Transactions
— Record Printout
— Report Printout and Concise Note Preparation
— Archival of a Material Balance Period
— Corrections.

However, significant changes have been made in a number of areas, as will be described in the following sections.

2. UPGRADES AND REVISIONS OF THE ORIGINAL SARP

SARP-II results from the upgrades and revisions of the original SARP as described in the following sections. It will help the reader in understanding several points in the discussion, to point out that the Accounting Transactions module, which represents the core of the program, covers a total of nine possible fuel assembly movements, as indicated in Fig. 1.

2.1. Code and communications changes

All the changes described here are, of course, effected through changes in the program code. In addition to improvements in the performance and the user friendliness of the system, however, the program code has been updated to a newer version of CLIPPER (to CLIPPER 5.01). Further, for several instances (different types of accounting transactions) in which the accounting calls for identical sequences of data
FIG. 1. Possible fuel assembly movements and corresponding accounting transactions.
manipulation, code has been written as 'functions', which are then called by the program as needed. In the earlier version of SARP, the program was written in such a way that each accounting transaction had its own extended segment of code, which was called into play when the user elected to use that transaction. The newer code version is thus 'integrated', with a resultant decrease in program length.

As with the original version, SARP-II is menu driven. Special colour-coding has been introduced which reminds the user of the particular module of the program being run. For example, the Initialization module screens are all with orange background and blue print, the Accounting Transactions module screens are all dark blue with light blue print, the Record Printout module screens are green with black print, etc. Error messages to the user are always red with yellow print.

The main-menu default choice has been set at Accounting Transaction selection, since this is expected to be the most frequently used module. Additional streamlining has been accomplished by allowing the user to 'block' select multiple items. The user is often given lists of fuel assembly batch identification numbers or of record identification numbers, etc. Rather than using arrow keys plus 'enter' to select multiple items one at a time, the user can 'block' a group, either by the use of keystrokes or by the use of the mouse. This option eliminates repetitive keystroke sequences.

A listing of fuel assemblies by key measurement points (KMP) has also been made available to the user and can be called up at a number of relevant times during the program.

2.2. Improvements to the 'Corrections' module

SARP allowed the user to make corrections to input data at the time of on-line entry. It also allowed limited subsequent corrections in which fuel assembly data could be changed on the basis of a 'correcting accounting entry' which was retained as a record in itself. Should very few corrections be needed, this approach would probably be sufficient. However, to accommodate the eventuality that numerous corrections are needed, the 'Corrections' module of the program has been changed.

In SARP-II the first method of correction is still allowed (at the time of on-line data entry). For correction of errors which are not recognized at the time of on-line data entry, however, an analysis of the types of expected errors was performed. On the basis of this analysis, it became clear that most errors would result from the 'incorrect' use of an accounting transaction. For example, if the user realized that incorrect data for a fuel assembly had been entered as part of performing transaction (6) (spent fuel movement from the reactor core to the spent fuel storage) and that the error had been later perpetuated through the user performing transaction (7) (spent fuel transfer from storage to shipping) for the same fuel assembly, a correction would be made by first 'reversing' transaction (7) and then by 'reversing' transaction (6). Corrections are thus based on the transactions. They are performed
'stepwise', backward from the accounting situation when the error was recognized to the situation when the error was first introduced.

2.3. Messages to the user and quality control checks

Throughout the program, an effort has been made to increase and clarify messages to the user, notably in two main areas. The first series of messages facilitate smooth use of the program, i.e. they give 'on-line' indication of the next step the program will automatically take, once the user has hit 'enter' or indicated completion of a given step.

The second series of messages indicate errors to the user, i.e. the messages follow from built-in quality control checks performed by the program. Many of these checks relate to the requirements of the IAEA Code 10 (e.g. certain letter codes must be used to indicate the type of nuclear material involved). Other checks relate to the logic of the accounting. An example of a user message resulting from quality control checks could involve a case in which the user attempts to perform a transaction in which a fuel assembly is moved from one key measurement point to another (KMP-a to KMP-b). The user inputs a batch identity number for a fuel assembly which is already in KMP-b. The program checks all batch identity numbers in KMP-a for a match, fails to find one, and therefore sends the user a message that the user-input batch identity number does not exist in KMP-a, giving an option to retry.

2.4. Sizing of fresh fuel, spent fuel and reactor core

The model facility on which the program is based has a spent fuel pond which is divided into two areas, one for fresh fuel and one for spent fuel. The user is given the opportunity to input dimensions of the whole pond (x rows by y columns, with rows defined by numbers and columns defined by letters). Then the user is asked to specify the area reserved for fresh fuel (e.g. A-1 through G-10). The remainder is automatically designated spent fuel storage. In a similar manner, the user is asked to input dimensions of the reactor core (z rows by q columns).

For any transactions involving fuel storage areas or the reactor core, the user is given the opportunity to 'view' the relevant area map, in which the current fuel assembly occupant is indicated.

2.5. Tutorial upgrades

The tutorial has been upgraded to reflect the menu colour-code, to provide example data to the user, and to allow the 'learner' to participate in an extended exercise in which all of the main program modules are used with a given set of data. This means that the learner initializes the program, which includes the following steps: sizing and identifying the reactor core, the spent fuel pond and the fresh fuel section;
inputting the current uranium, \(^{235}\text{U}\), and plutonium inventory in the general ledger and in all relevant subsidiary ledgers; and inputting all fuel assembly data (fuel assembly history cards) for each KMP. Once the initialization is complete, the learner starts performing accounting transactions wherein:

- ten fuel assemblies are received;
- five fuel assemblies are moved to the fresh fuel storage;
- two fuel assemblies are treated as faulty and are moved to storage for shipping;
- four fuel assemblies are moved from the fresh fuel storage into the reactor core;
- three fuel assemblies are moved within the core;
- two fuel assemblies are moved from the core to the spent fuel storage;
- one fuel assembly is moved from the spent fuel storage to shipping;
- one fuel assembly is moved from the spent fuel storage back to the core.

Throughout these transactions in the tutorial, the learner sees material transfer forms, inventory change documents, fuel assembly history cards, general and subsidiary ledger updates, and fresh fuel, spent fuel and core status updates. Finally, the learner is allowed to print out a 'mock' set of tutorial records and the relevant 'mock' IAEA reports, i.e. the Physical Inventory Listing, the Material Balance Report and the Inventory Change Report. All data entered during the tutorial are 'mock' data and all relevant files are purged upon exiting the tutorial module.

2.6. SARP-II interface to SARP

To allow users to change their nuclear safeguards accounting system at a given facility from SARP to SARP-II, a code has been introduced to copy SARP files into the appropriate locations in SARP-II. This interface would be needed, for example, if a user were changing from the earlier system to the newer version at some time other than at the beginning/end of a material balance period.

3. SUMMARY

SARP-II represents an updated, more user accommodating, integrated version of the original safeguards accounting and reports program. Changes made include revision to CLIPPER 5.01, extended quality control checking with accompanying messages to the user, restructuring of the approach to 'corrections', menu colour-coding, data 'blocking' options, and expansion of the tutorial.
REFERENCES


NUMIS — INTEGRATED NUCLEAR MATERIALS MANAGEMENT
A strategic business tool

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1. INTRODUCTION

The Fuel Division of British Nuclear Fuels plc (BNFL) is a bulk handling facility for the conversion of uranium ore concentrate to natural UF₆ or uranium metal fuel, and the conversion of enriched UF₆ to fuel or intermediate products. In the early 1980s, the Fuel Division embarked on the development of a comprehensive nuclear materials management (NMM) system in collaboration with the CEC Joint
Research Centre, Ispra. The resulting Nuclear Material Information Service (NUMIS) was designed as an integrated system satisfying strategic business needs and accommodating a wide range of requirements. The philosophy adopted during conceptual design was that the demands on an NMM system "for financial, commercial, safety and quality assurance purposes are far more stringent than those for Safeguards" [1]. Therefore, it was believed that a high quality safeguards reporting system could be integrated into the NUMIS system with relatively little overhead on either the database or the processing functions. We discuss here the benefits gained from this integrated approach and the successful role that NUMIS plays in supporting key business objectives.

2. SAFEGUARDS APPLICATIONS

NUMIS generates automatically all the safeguards data for the facility, i.e. inventory change reports (ICR), physical inventory listings, lists of inventory items and material balance reports. The system has enabled the facility to reduce the costs of fulfilling its statutory safeguards obligations at a time of increased inspection frequency.

The Fuel Division currently has ten material balance areas, and two more are planned. The ICRs can be obtained on line, at any frequency required by the Euratom inspectors (daily, if desired). The monthly ICR is transmitted to Euratom on a floppy disc; a typical ICR contains 15 000 lines of data. Transactions are recorded at the lowest batch level, i.e. a drum of material, providing greater transparency for inspectors. A dictionary of eight character safeguards batch numbers is referenced to 17 character NUMIS batch numbers, which provides full traceability.

A powerful enquiry programme permits on-line interrogation of the ICR data, with many possible permutations and selection criteria. Inspectors may request ICR extracts and summaries for any type of transaction, e.g. a list of all new measurements declared during a specified time period. The ICR data, or the ICR enquiry subsets, can be printed locally or output to discs for transfer to personal computers.

A useful feature for routine weekly inspections is the ability to list transactions in a time sequenced order (to one hundredth of a second!) with a cumulative balance. This facilitates the reconciliation of the book balance with the physical stock while movements are going on.

A customized report calculates measured discards in effective kilogrammes for any specified date range. This is used by the inspectors to confirm adherence to the requirements in the particular safeguards provisions.

3. SOURCE DOCUMENT GENERATION

NUMIS generates export documents for all main product streams. Electronic data interchange is also under development.
All internal transfer documents are generated by the system at the point of transfer.
Packing sheets are generated for fuel boxes.

4. MATERIAL CONTROL

NUMIS facilitates good material control in various ways:

— Invalid moves for criticality reasons are not permitted;
— Invalid moves for quality control reasons are not permitted;
— On-line material balances for individual plants or processes;
— Material balances by production campaign;
— Material balances by customer;
— Material balances by works accountancy areas;
— Store location control;
— Chemical analysis data are automatically transferred onto the system to update batch details;
— Full batch traceability;
— Generation of transfer source documents;
— Generation of lists of inventory items for physical verification by operators, safeguards inspectors or auditors;
— Provision of a ‘what-if’ facility to plan UO₂ powder blending.

5. MAIN FEATURES OF THE SYSTEM

5.1. Data entry and update

— On-line transaction input with shop floor data capture;
— Alternative batch transaction input;
— Extensive validation of data;
— Automatic generation of material accounts;
— Automatic generation of summary records;
— Automatic generation of inventory difference records;
— Automatic retrospective update of uranium weights.

5.2. Reports and enquiries

— Many formats are supported;
— Many selection criteria;
— All enquiries are on-line;
— Generation of dispatch documents and internal transfer documents;
— Automatic safeguards reporting;
— Full analytical data are available on-line;
— Reports are printed in many locations;
— Personal computers are supported.

5.3. Size of the system

— Number of registered users: 800;
— Number of physical movements per year: 250 000;
— Number of material accounts: 3500.

REFERENCE

SAFEGUARDS FOR
URANIUM ENRICHMENT AND
FUEL FABRICATION FACILITIES

(Session 4)

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DEVELOPMENTS IN SAFEGUARDS AS APPLICABLE TO URENCO'S ENRICHMENT PLANTS
An operator's perspective

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Abstract
DEVELOPMENTS IN SAFEGUARDS AS APPLICABLE TO URENCO'S ENRICHMENT PLANTS: AN OPERATOR'S PERSPECTIVE.

A Continuous Enrichment Monitor (CEMO), developed under the United Kingdom Safeguards Support Programme for the IAEA, has been tested on the Urenco centrifuge enrichment plants at Capenhurst and Almelo. A development model has been evaluated since July 1992, monitoring independently eight cascades at the Capenhurst plant. It has shown itself capable of accurately and reproducibly measuring the enrichment level in the product pipe, confirming the production of low enriched uranium and detecting periods when UF₆ operation was interrupted. A short term test at Almelo confirmed that a similar instrument could be used successfully on the combined output of several cascades, to monitor the product assay level. The installation requirements for the CEMO present no problems for the latest generation of Urenco technology installed at Capenhurst and Almelo and, once the CEMO has been installed, no plant operator involvement in its operation is necessary.
1. INTRODUCTION

The Urenco centrifuge enrichment plants at Almelo, the Netherlands, Capenhurst, United Kingdom, and Gronau, Germany, are under safeguards according to the principles agreed at the Hexapartite Safeguards Project (HSP) meetings (1980–1983). These plants are subject to materials accountancy, containment and surveillance (C/S) and limited frequency unannounced access (LFUA) inspections. At the time of the HSP meetings, Urenco had successfully operated three pilot plants and had completed commissioning its first two 200 tSW/a enrichment plants at Almelo (SP3) and Capenhurst (E21). Construction of two larger plants (SP4 and E22) at the same sites was in progress. Since that time, the 200 tSW/a enrichment plant at Capenhurst has been closed after 15 years of operation, the E22 plant at Capenhurst has been completed with a capacity of 850 tSW/a, while the plant at Almelo is nearing completion with a capacity of around 1200 tSW/a. In addition, a new 1000 tSW/a enrichment plant is currently under construction at Gronau in Germany. First capacity of this plant came on line in 1985 and current capacity is around 500 tSW/a.

Since the HSP meetings, centrifuge and the related plant technology has also been extensively developed. Urenco is currently installing the fourth generation of commercial centrifuges, and the development of a fifth generation machine is on schedule for deployment in the second half of the decade. These developments have resulted in uranium enrichment plants which are much more efficient and flexible than the earlier designs in the production of low enriched uranium (LEU) for use in fuel for commercial reactors.

As foreseen at the time of the HSP meetings, considerable development programmes [1–4] have been undertaken on non-destructive assay (NDA) measurement techniques for potential use during LFUA visits to centrifuge enrichment plants. The experience gained with the development of these techniques for the intermittent application of NDA to centrifuge cascades formed the basis for the development of a Continuous Enrichment Monitor (CEMO) within the UK Safeguards Support Programme for the IAEA [5], funded substantially by the UK Department of Trade and Industry. Development trials of this monitor have been performed on the latest tranches of capacity installed in the Capenhurst enrichment plant, with additional tests at Almelo. In summary, the CEMO has been tested successfully under various combinations of pressures and pipe dimensions present during normal operation of Urenco plants.

Urenco fully supports the development of equipment for the continuous monitoring of the product enrichment assay in centrifuge enrichment plants.
2. DEVELOPMENTS IN URENCO TECHNOLOGY

Following the Treaty of Almelo, which enabled the establishment of the Urenco organization, Urenco operated three pilot plants based on technology developed independently in the three participating countries. The later tranches of these pilot plants already incorporated improvements, resulting from the interchange of development experience between the three countries. A complete assessment of the performance and future development potential of all areas of the centrifuge and plant technology was undertaken and two design lines were agreed upon for further development.

The first generation centrifuges installed in the two subsequent 200 tSW/a plants built in parallel had a low separative power. As a consequence, even with many hundreds or thousands of centrifuges installed per cascade, there were many replicated cascades in both plants. The area of the cascade halls was also large, typically e.g. 100 m × 100 m in SP3. Through the development of the next generations of centrifuges the separative power was continually improved by increasing both the length and the speed of rotation of the rotor. The separative power of the currently installed fourth generation centrifuge is an order of magnitude higher than that of the first generation machines, with a further significant increase already demonstrated for the next generation. As a consequence, capacity which originally required many tens of cascades can be constructed now using only a few cascades occupying typically 25 m × 50 m in SP4. Centrifuge mounting and thermal management technology encompassing both single and block mounted concepts has similarly developed through the generations.

Because of the low growth rate of nuclear power generation since the 1970s, Urenco's plant expansion, which is based on matching capacity with firm orders, has been slower than originally expected. As a result, the two larger enrichment plants at Almelo and Capenhurst have incorporated the latest generation technology as it has become available. The E22 plant at Capenhurst contains all four generations of centrifuge technology. First generation centrifuges remain in operation in the SP3 plant at Almelo and second to fourth generation centrifuge technology is installed in SP4. A trial cascade for the latest machine has also recently been commissioned at Almelo.

3. CURRENT SAFEGUARDS REGIME

All Urenco centrifuge enrichment plants are subject to safeguards based on the conclusions of the HSP meetings. Procedures include monthly inspection visits covering material accountancy with associated C/S measures, including container seals and, in selected areas, video surveillance and an annual physical inventory verification (PIV). As agreed in the HSP negotiations, visits to the centrifuge cascade
halls are an integral part of the safeguards regime. At Capenhurst this has included the use of an off-line NDA measurement instrument, the Cascade Header Enrichment Monitor, to confirm the presence of LEU in cascade pipework. The LFUA visits to the cascade area are to verify that no changes have been made to the installation of the process equipment by checking against design drawings/photographs kept on the site under Inspectorate seal. This verification has been assisted by the essentially repetitive nature of the cascade pipework and the large number of similar cascades. These procedures have been in operation for about nine years and have enabled the Inspectorates to confirm that the Urenco plants have only been producing LEU and that safeguards criteria have been met.

However, as the centrifuge plant technology has developed, commercial pressures have led to continuing improvements in the flexibility and efficiency of the production of LEU for commercial reactors and to larger enrichment plants. When the SP4 plant at Almelo is completed, it will have a capacity nearly 50% higher, which is installed in 20% fewer cascade halls than originally planned. The increasing number and size of nuclear facilities subject to international safeguards also lead to the requirement for increasingly efficient safeguards. A continuous enrichment monitor, which would reliably confirm the continuous production of LEU in a centrifuge enrichment plant on a 'go/no go' basis, would thus seem to be a significant benefit to the Inspectorates. Reliability is, however, essential because any 'false' indication that LEU production was not confirmed would lead to significant effort by operators and Inspectorates to resolve the apparent anomaly, and would raise unwarranted doubts on the use of the enrichment plant.

4. ENRICHMENT MONITORING

There has been considerable research undertaken over the last decade into systems for the 'intermittent' measurement of the enrichment in centrifuge cascade pipework on a 'go/no go' basis (LEU confirmation). The instrumentation has generally been limited by the need for liquid nitrogen cooling of high resolution detectors, and the performance has been deleteriously affected by the small diameter process gas pipework in the early centrifuge plants and by extensive UF$_6$ deposits present in some of those pipes. The end result has been the need for long measurement times, complicated measurement techniques and considerable inspector (and associated operator supervision) time to implement measurements. The latest centrifuge cascade pipework diameter is significantly larger than that of much of the earlier technology, thus improving the ratio of the gas signal to the deposit signal and providing potentially more suitable conditions for NDA monitoring.
5. THE CONTINUOUS ENRICHMENT MONITOR

The CEMO detects the total mass of $^{235}$U in the pipe monitored ($\gamma$) by measuring the 185.72 keV $\gamma$ rays emitted by that isotope. The gas pressure ($p$) is determined from the measurement of the absorption of Ag K X rays (22.25 keV) emitted by a $^{109}$Cd source. These two measurements permit the enrichment ($E$) to be determined from $E = \gamma \times K/p$, where $K$ is a constant. Both measurements use a common low resolution scintillation detector fitted with a local high voltage generator. The detector is gain stabilized using 88 keV $\gamma$ rays from the same $^{109}$Cd source. This complete package, which can be sealed by the Inspectorates, including lead shielding for the detector, is free mounted on an appropriate length of pipework, requiring only a short length of accessible straight pipework and approximately 350 mm clearance in one direction. The connection from this unit, and from up to seven other units, to the central control unit consists solely of screened low voltage cable. Each unit has its own 256 channel multichannel analyser. The central control unit performs all calculation for determining and displaying the mass of $^{235}$U, the pressure in the pipe and the enrichment, and the upper (3$\sigma$) statistical bound for enrichment, as well as logging such data and background counts. The complete central control unit, electronics and associated support equipment can be installed in a cabinet (approximately 1 m $\times$ 1 m by 2 m high) located up to 100 m from the detector heads and which can be sealed by the Inspectorates.

For the operator the most important points of the design, apart from the small spatial requirements, are the minimal services required. The operator only needs to provide a 'highly reliable' power supply, as the CEMO has a permanent battery backup that covers any short loss of mains power. The gain stabilization of the detector compensates for the temperature dependence of the detector, permitting operation in an uncontrolled (temperature) environment. The low voltage connection between the detector and the control unit presents no safety hazard and the low resolution detector requires no liquid nitrogen. Thus, once the CEMO is installed, the operator should effectively have no further commitments to its routine operation.

5.1. Location

The possible location for the installation of the CEMO within the centrifuge enrichment plant is determined by a number of factors, including:

(a) Unhindered access by the Inspectorates to both the control module and the detector heads, the latter for checking seals and routine maintenance. This implies a location for the detector head outside the LFUA area.

(b) The availability of a length of suitable straight pipework of acceptable diameter and wall thickness, in which deposits and process gas pressures are of an acceptable level.
(c) Potential take-off points between the cascades and the CEMO monitoring point, and the suitability of C/S measures, if present, to cover such situations.

Urenco considers that suitable locations and conditions exist for the latest technology installed in the enrichment plants at Capenhurst and Almelo. (The latest technology has not yet, but will be, installed in the plant at Gronau). At Capenhurst this location is on the product header connected to each cascade after it leaves the cascade hall and before the output from several cascades is combined and collected in an assay unit. At Almelo this location is on the product header, after the output of several cascades has been combined within the cascade hall in an assay unit header, also after it leaves the cascade hall.

Although in both cases the pipework is made of aluminium, the pipe diameters and pressures vary widely. External pipe diameters range from 120 to 220 mm, wall thicknesses are between 4 and 5 mm, and operating pressures range from one hundred to several hundred Pa.

5.2. Operation

Centrifuge enrichment plants are designed for continuous operation and operate under stable conditions during a production campaign to produce product and tails to given assays defined by customer requirements. A change in assay within the flexibility of the cascade will require new cascade settings and the cascade will come to a new equilibrium over a period of hours. Typically, the Urenco centrifuge technology is designed to operate for more than ten years without maintenance, so no cascade 'downtime' for routine planned maintenance is expected. Interruptions to production can occur, however, e.g. owing to power cuts, support systems failures, etc. These outages are typically very short, mostly lasting minutes rather than hours, and are infrequent. Cascades are on line for around 99% of the time. Provided that the pipe dimensions and normal operating pressure of the pipe monitored are suitable, the CEMO should thus be able to provide confirmation of LEU production leaving the cascade hall continuously. In addition, when mounted on a cascade basis, the CEMO would indicate when and for how long the cascade was off line.

5.3. CEMO plant experience

5.3.1. Capenhurst

Plant experience with an early prototype CEMO was gained in 1989 when a four-detector variant was first mounted on cascade pipework at Capenhurst. The equipment has been continuously evaluated and development continued since that time. In July 1992 an eight-detector head development model CEMO was installed.
FIG. 1. Gamma ray emission (\(^{235}\text{U}\)) and X ray absorption (pressure) data from one CEMO detector head mounted in Capenhurst, for the period after July 1992, and the resultant calculated enrichment +3 \(\sigma\) (%).
on the last eight cascades commissioned and it has monitored the product from these cascades to the present day. This period of operation has included a joint programme by the developers: AEA Technology (Harwell Instruments) and the IAEA/Euratom Inspectorates. The results have been discussed with the UK Department of Trade and Industry and with Urenco. Not only did the results confirm LEU production whilst the cascades were on line, but they also showed that the accuracy of the enrichment measurement under normal operating conditions was extremely high and that, even with a $3\sigma$ statistical spread, the 'LEU not confirmed' limit will not be exceeded erroneously. Similarly, the pressure measurement has reliably detected those process gas outages that occurred.

The detector on the eighth cascade was installed when the cascade was in vacuum before UF$_6$ was introduced. During UF$_6$ commissioning of the cascade, the 22 keV X ray transmission was reduced, as expected, by a combination of both a deposit buildup and the process gas. Later gas outages indicated that the deposit signal was significantly higher (more than twice as high) than the gas signal. Consequently, the CEMO calibration constants had to be adjusted after commissioning. With the equipment accessible to the inspectors at all times, such corrective measures could be taken during their next visit. Typical traces from a detector showing the $\gamma$ ray emission ($^{235}$U), the X ray absorption (pressure) and the resultant enrichment levels are shown in Fig. 1.

5.3.2. Almelo

The cascade hall containing the latest technology in the Urenco SP4 plant at Almelo is constructionally different from that of the E22 plant in Capenhurst. A suitable location for the detector was identified on the header pipe where the combined output of several cascades left the cascade hall. The application of CEMO on a group of cascades, rather than on individual cascades, has the potential to reduce capital investment in equipment, but is less advantageous if there are possibilities for the removal of the process gas before the monitor. In this case, additional C/S measures would be necessary to cover these possibilities in order to gain the full benefit of the CEMO. The pressure at this location is significantly lower, but the pipe has a larger diameter and a thinner wall than at Capenhurst. The lower pressure is detrimental to the accuracy of the pressure measurement, but the pipe dimensions are beneficial. Urenco sponsored a short term trial of a CEMO with the same software and basic hardware (though with an adapted source/detector housing) on the plant at Almelo. To gain information on the sensitivity of the measurements while not interrupting the production campaign in progress, the CEMO was mounted on both the tails header (nominal assay 0.3% $^{235}$U) and the product header (nominal assay 4% $^{235}$U), which at this location have identical pipe geometries and material. The pressure in the tails header is approximately twice that in the product header. The results are summarized in Fig. 2. Although these were short term tests, the indications are that the spread
FIG. 2. Gamma ray emission ($^{235}$U) and X ray absorption (pressure) data from the single CEMO detector mounted on tails and product pipes at Almelo for one and two days, and the resultant calculated enrichment (%). Runs 1-9 are measurements made on the tails pipe (10 = average); runs 11-27 are measurements made on the product pipe (28 = average).
FIG. 3. High resolution spectra from the background and the product pipe containing ex-oxide reprocessed re-enriched material.

in measurements (over two days) was similar to that expected from counting statistics and thus erroneous indications of LEU not confirmed should not arise. The tests also confirmed the benefits of larger diameter, thinner wall thickness pipes to the performance of the CEMO.

During these tests at Almelo, a check could be made to determine if the use of UF\textsubscript{6} from reprocessed ex-oxide material in cascades monitored by CEMO would adversely affect the measurements. A high resolution germanium semiconductor detector was mounted on the pipe set to detect $\gamma$ rays with energies up to 2000 keV. The background spectrum and the spectrum from the product pipe indicate no peaks that are liable to interfere with the operation of the CEMO (Fig. 3).

6. CONCLUSIONS

Long term trials of the CEMO on individual cascades of the latest generations of technology installed in the Urenco centrifuge enrichment plant at Capenhurst have confirmed that the instrument will reliably confirm LEU production while the plant is operating under normal conditions. The instrument will also detect reliably instances when process gas is removed from the cascade. Initial tests suggest that a similar conclusion with respect to the ‘confirmation of LEU production’ can be drawn for the Urenco enrichment plant at Almelo, with CEMO monitoring an assay unit (group of cascades).
Apart from requiring a short length of suitable pipework, a small cabinet and a reliable power supply, the CEMO makes no demands on the operator during routine operation.

Urenco would welcome the use of the CEMO by the Inspectorates as a reliable addition to the current range of safeguards techniques applied to the latest generations of technology installed in current and future Urenco centrifuge enrichment plants.

REFERENCES


[3] Van de MEER, K., "Enrichment verification on UF$_6$ in low pressure process pipes by the two geometry method", ibid., p. 177.


OUTLINE OF THE SAFEGUARDS IMPLEMENTED AT THE ROKKASHO URANIUM ENRICHMENT PLANT

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Presented by Y. Morikami

Abstract

OUTLINE OF THE SAFEGUARDS IMPLEMENTED AT THE ROKKASHO URANIUM ENRICHMENT PLANT.

The Rokkasho uranium enrichment plant is Japan's first commercial scale enrichment facility. Construction began in October 1988, and normal operations were launched in March 1992 with an initial capacity of 150 t separative work units per annum. Facility Attachment negotiations with the IAEA concerning material accountancy and the scope of inspections at the Rokkasho plant began in October 1990. An agreement was reached in September 1991. Regulations on material accountancy were also approved by the Government in September of the same year. The main features of material accountancy in the Rokkasho plant are: (a) two material balance area (MBA) structures, consisting of a storage MBA and a process MBA; (b) in-operation physical inventory taking by simultaneous switch-over of feed, product and tails flows; (c) evaluation of material balance by the uranium element balance method, which does not hamper operations at the plant. The Rokkasho plant has adopted several safeguards relevant designs to enable efficient and effective limited frequency unannounced access inspections in the cascade area.

1. OUTLINE OF THE FACILITY

The Rokkasho uranium enrichment plant is located 30–60 m above sea level in a hilly region in the Oishitai District of Rokkasho Mura in Kami-Kita County in the southern part of the Shimokita Peninsula, in the north-eastern section of Aomori Prefecture. The plant is about 80 km from the city of Aomori and 40 km from the city of Misawa.

The plant produces enriched uranium through the gas centrifuge process. Development of this enrichment technique began in 1972 as a national project mainly undertaken by the Power Reactor and Nuclear Fuel Development Corporation. Using domestically developed technologies that resulted from this project, the construction of the plants began in October 1988. After a series of inspections and tests, the plant began operation in March 1992 with an initial capacity of 150 t separative work units (SWU) per annum. The plan calls for gradually increasing capacity in units of 150 t SWU/a, eventually to achieve 600 t SWU/a by the end of 1994 (see Table I).
TABLE I. FACILITY OUTLINE

<table>
<thead>
<tr>
<th>Item</th>
<th>Content</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design capacity</td>
<td>600 t SWU/a</td>
</tr>
<tr>
<td>Design capacity of operation unit</td>
<td>150 t SWU/a</td>
</tr>
<tr>
<td>No. of operational units</td>
<td>4</td>
</tr>
<tr>
<td>Area of plant grounds</td>
<td>Approx. 3.4 million m²</td>
</tr>
<tr>
<td>Building area</td>
<td>Approx. 30 000 m²</td>
</tr>
<tr>
<td>Throughput of feed material</td>
<td>Max. 1150 t U</td>
</tr>
</tbody>
</table>

Incoming shipment of feed material
Temporarily stored in storage

Feed material

48Y cylinder

Feed cylinder vessel

Material evaporation

Cascade

Product

Cold trap

Trapping/collection of products

Intermediate product cylinder vessel

Withdrawal of product

Homogenization vessel

Homogenization by liquefaction

Enrichment measurements

Blending to adjust enrichment as required

Intermediate product (Intermediate product cylinder)

Product cylinder vessel

Refill

Product

30B cylinder

Store in storage

Shipment of uranium product

Compressor

Tails cylinder vessel

Withdrawal of tails

Tails

48Y cylinder

Store in storage

FIG. 1. Schematic diagram of major processes.
The main buildings in the plant include a uranium enrichment building, a UF$_6$ storage building, a waste storage building and an auxiliary building. Major facilities and equipment, such as cascade facilities, a UF$_6$ handling facility, and homogenization and blending equipment, are installed inside the uranium enrichment building. Major processes employed at the Rokkasho uranium enrichment plant are shown in Fig. 1.

2. HISTORY OF SAFEGUARDS

The safeguards implemented at the Rokkasho uranium enrichment plant are based on the Japan–IAEA Safeguards Agreement and the Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors. Owing to the special nature of the plant, the implementation of safeguards largely reflects the conclusions reached in the Hexapartite Safeguards Project, which involved six organizations concerned with gas centrifuge enrichment facilities.

Facility Attachment (FA) negotiations on the material accountancy method, inspections and other issues began in October 1990. Agreement was reached in September 1991. Regulations on material accountancy were also approved by the Government in September 1991.

3. CURRENT SAFEGUARDS

3.1. Material accountancy

To implement nuclear material accountancy, the plant established a material balance area (MBA) and key measurement points (KMPs), as shown in Fig. 2.

Reports of material accountancy submitted to the Government from January to December 1993 focused on the acceptance of overseas feed materials and standard samples, and intra-MBA transfer. There were a total of 895 inventory change reports (ICRs).

The nuclear substances handled at the plant are controlled in cylinder units, with the exception of nuclear materials inside analysis samples, the chemical trap, the cold trap, and sludge. The weight and volume of nuclear substances in the product cylinders and tails cylinders are confirmed through measurement equipment, with measurement accuracy as shown in Table II.

3.2. Physical inventory taking

Physical inventory taking (PIT), carried out during the normal operation of an enriched uranium plant (‘in-operation PIT’ method), is implemented approximately once every 12 months.
TABLE II. METHODS TO DETERMINE PRODUCT CYLINDERS AND TAILS CYLINDERS

<table>
<thead>
<tr>
<th>Key</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Cylinder receipt/shipment, shipper-receiver difference</td>
</tr>
<tr>
<td>2</td>
<td>Transfer of cylinder</td>
</tr>
<tr>
<td>3</td>
<td>Receipt and shipment of samples</td>
</tr>
<tr>
<td>4</td>
<td>Discards, accidental loss and gain</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Key</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>UF₆ storage</td>
</tr>
<tr>
<td>B</td>
<td>UF₆ feed/withdrawal room, intermediate room and homogenization room</td>
</tr>
<tr>
<td>C</td>
<td>Management and water treatment room, decontamination room and analytical laboratory</td>
</tr>
</tbody>
</table>

Calculation of uranium amount: (Gross weight − tare weight) × U concentration.
Calculation of ²³⁵U: (Gross weight − tare weight) × U concentration × enrichment.
Figures in parentheses show degree of accuracy.
The method determines the physical inventory through the simultaneous switch-over of the operation lines between the feed vessels and the cascade, and the cascade withdrawal equipment for product and tails. The procedure is as follows:

- Using the weighing and process data, determine beforehand the inventory within the cylinder vessel and the product cold trap, which will be used after switching operation lines.
- Determine the inventory of the enriched uranium and tails, using weighing methods, and determine the enrichment of UF₆ in the tails cylinder vessel by estimation methods.
- Transfer to the intermediate product cylinder (IPC) the enriched uranium found inside the detached product cold trap. After transporting the IPC to the homogenization room, carry out homogenization treatment, and determine the inventory through sample analysis and weighing.
- Calculate the uranium gas residue inside each cold trap, using the gas equation. Calculate the inventory inside each chemical trap, using operational data.
- Clean out various cylinders, drums and analysis samples, and determine the inventory.

Implementing PIT through the above method will enable the identification of all individual nuclear materials, as well as minimizing the hampering of plant operations.

In PIT, an evaluation of the material balance is made based on the confirmed value of the physical inventory. The uranium element balance method is used for this evaluation because the amount of tails (depleted uranium) withdrawn at the enrichment plant is several times that of the product (enriched uranium). For economic reasons, homogeneous liquefaction and precise measurement of the enrichment levels for the purpose of material accountancy are not performed at commercial plants.

The uranium element balance method is used to evaluate the material balance. The objectives of safeguards can be fully satisfied by using this method, which accurately measures the total uranium amount by measuring the weight of tails, rather than by measuring the material balance of ²³⁵U with analytical data, as the latter method can lack accuracy. This method has already been agreed upon and implemented at the previously mentioned plants.

3.3. A system to report material accountancy records

The record and report system of the safeguards data of the plant is a subsystem of the Rokkasho uranium enrichment plant management system, which controls the flow of information for the entire plant. Data necessary for material accountancy are obtained from the data of other subsystems, and incorporated in a database.
The system enables the timely provision of information necessary for inspections as well as recording and reporting matters related to safeguards. It also enhances work efficiency.

The material accountancy records reporting flow is shown in Fig. 3.

3.4. Inspections

The main objectives of inspection activities carried out at the Rokkasho uranium enrichment plant are as follows:

— To verify that the flow and inventory of nuclear materials are exactly as declared;
— To verify that production of nuclear materials is within the range of declared enrichment.

To achieve these objectives, routine inspections are implemented by both national and IAEA inspectors. The contents of implemented inspections and responses by the facility are outlined below.

3.4.1. Design information verification (DIV)

Examples of DIV include visual confirmation that the design information provided and the actual arrangement of the facilities and equipment are in accordance
with the declaration and, within the cascade rooms, visual confirmation that the
arrangement, piping and valves of the cascades are in accordance with the
declaration.

In conjunction with this, records of photographs and other materials to be used
for limited frequency unannounced access (LFUA) are drawn up under the agree­
ment with the IAEA.

3.4.2. Interim inspections

Interim inspections are conducted once a month. These include book audits to
check for any inconsistencies between the report and the record, and on-site inspec­
tions (calculation of contained items, item identification, weighing, measuring of
cylinder non-destructive assay (NDA), etc.) to verify the location, identity, quantity
and composition of nuclear materials which will become safeguards targets. The
operator presents the following documents to enhance the efficiency of interim
inspections:

- original inventory change report
- summary of inventory change report
- total inventory list
- cylinder list
- layout of cylinder location in UF₆ storage.

3.4.3. Limited frequency unannounced access

To accommodate LFUA safeguards inspection activities inside the cascade
hall, the following design features have been implemented:

- Simplified piping in the cascade hall has been designed so that visual
  observation alone may suffice. One example is the installation of the complex
  piping portion of the cascade system outside the cascade hall.
- The penetration of piping between the inside and outside of the cascade area
  permits visual inspection.
- The feed, product and tails piping in each cascade is extended directly from
  the cascade hall to the outside intermediate room. This piping arrangement
  makes it possible to carry out NDA measurements inside the intermediate
  room.

LFUA inspection activities by visual observation include the following
actions:

- Verify that the piping and arrangements in the cascade hall are exactly as per
  photographs taken at initial DIV;
— Verify that the penetration pipes to the intermediate room are exactly as per photographs taken at initial DIV;
— Verify that the penetration pipes to the intermediate room are exactly as declared.

The facility records such details as the total number of hours spent inside the cascade hall.

3.5. Future plants

To cope with planned future facility expansion of up to 1500 t SWU/a, the plant management is considering, from a technological standpoint, the most effective system of safeguards implementation which concurrently meets the demands of the IAEA.

4. POSTSCRIPT

So far, the Rokkasho uranium enrichment plant has been implementing safeguards without any major problems. However, the planned expansion of the facility is expected to increase the amount of nuclear materials handled as well as work related to safeguards. To cope with this situation, the authors feel that more effective and efficient safeguards measures will need to be implemented in the future.
DEVELOPMENT OF AN NDA APPROACH FOR VERIFYING THE PROCESS INVENTORY OF A GASEOUS DIFFUSION ENRICHMENT CASCADE

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Abstract

DEVELOPMENT OF AN NDA APPROACH FOR VERIFYING THE PROCESS INVENTORY OF A GASEOUS DIFFUSION ENRICHMENT CASCADE.

An extensive series of measurement experiments are being conducted in the United States of America and in Argentina to evaluate a variety of gamma ray and neutron non-destructive assay (NDA) measurement techniques for measuring and independently verifying the gas phase and the solid phase process inventory in gaseous diffusion uranium enrichment cascades. The nuclear material contained in the process equipment of a gaseous diffusion cascade is a major component of the overall nuclear material inventory at a gaseous diffusion plant. Quantifying and verifying this material is an important element of both domestic and international systems for accounting and control of nuclear materials. To date, two joint US-Argentina field measurement evaluations have been conducted: one at the gaseous diffusion facilities in Oak Ridge, Tennessee, and Portsmouth, Ohio; and a second one at the gaseous diffusion complex in Pilcaniyeu, Argentina. The objectives of these field tests are to compare the performance of various NDA measurement methods for accuracy, repeatability,
intrusiveness and ease of implementation. The results from these measurement evaluations are being used to develop an approach that can be used by the IAEA and the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials to independently verify the process inventory of a gaseous diffusion cascade without compromising the security of proprietary commercial, industrial and technological information. The joint field evaluations are described, results to date are reported, and future work is outlined.

1. INTRODUCTION

In July 1991, Argentina and Brazil signed a bilateral agreement for the exclusive peaceful uses of nuclear energy which established a common system of accounting and control (SCCC) of all nuclear materials in all nuclear activities and created the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC) to apply the SCCC in the two countries for verifying the basic undertaking provided for in the agreement. In December 1991, Argentina, Brazil, ABACC and the IAEA signed a quadripartite agreement for the application of full scope safeguards in Argentina and Brazil. Once the ratification of the quadripartite agreement is complete, all nuclear material in all nuclear activities in both countries will be subject to IAEA safeguards.

The small scale gaseous diffusion plant that is being deployed at Pilcaniyeu, Argentina, will be the first gaseous diffusion plant to be subject to international safeguards. Currently, the IAEA has extensive experience in inspecting commercial enrichment facilities based only on the gas centrifuge process; ABACC has recently conducted the initial verification of inventory and design information at the enrichment facility in Argentina. Although the nuclear material process and handling activities at gas centrifuge and gaseous diffusion plants are quite similar outside of the cascade area, important safeguards relevant differences exist in the design and operating parameters. One important difference is the amount of process inventory associated with each type of enrichment cascade. For a given capacity, the process inventory for a gaseous diffusion plant is substantially larger than that for a gas centrifuge plant because of the larger process equipment, the higher process gas operating pressures, and the physics of the separation process. The nuclear material contained in the process equipment of a gaseous diffusion cascade is a major component of the facility's overall nuclear material inventory. Quantifying and verifying this material is an important element of both domestic and international systems for accounting and control of nuclear material. Conversely, the nuclear material contained within the process equipment of a gas centrifuge cascade is sufficiently small that it is not of safeguards significance. In the current international inspection regime, the in-process inventory for a gas centrifuge cascade is not verified by the IAEA. Thus, an important element of the safeguards approach developed for a gaseous diffusion plant will be the method of independently verifying the process inventory.
To address this safeguards issue, Argentina and the United States of America initiated a co-operative programme in December 1992 to discuss non-destructive assay (NDA) measurement methods being developed by each country to measure the process inventory of a gaseous diffusion cascade. Argentina has been developing and evaluating a variety of NDA techniques for determining gaseous diffusion plant hold-up to meet national safeguards requirements, and in anticipation of IAEA and ABACC safeguards implementation. The USA has been developing and implementing NDA measurement methods for determining gaseous diffusion plant hold-up to meet domestic nuclear materials accounting and criticality safety requirements at US gaseous diffusion facilities [1–4]. To date, two joint US-Argentine field measurement evaluations have been conducted to compare the cost, accuracy, repeatability, intrusiveness and ease of implementation of a variety of NDA techniques for measuring process inventory.

2. GASEOUS DIFFUSION CASCADE

2.1. Equipment configuration

The process equipment associated with gaseous diffusion operations is generally large in size and emits high amounts of heat and noise. A gaseous diffusion plant is typically comprised of a single cascade consisting of many repetitive units, or stages, of process equipment connected in series. Each stage consists of: (1) a large cylindrical vessel, called a diffuser, that contains the diffusion barrier, (2) one or two compressors used to compress the gas to the pressure needed for flow through the barrier, and (3) process piping for stage and interstage connections. In US cascades, diffusers are positioned horizontally; in Argentina, diffusers are positioned vertically. Individual stages are grouped together to form cells, or modules, which are the smallest units of production that can be isolated from the cascade and taken out of service for maintenance or other reasons. Cells usually contain twenty or fewer stages. Typically, individual cells are enclosed in a metal housing to control the environment around the process equipment. During cascade operation, the temperatures inside the housings can reach 70°C or more.

2.2. In-process inventory

The in-process inventory of an operating gaseous diffusion cascade is comprised of uranium hexafluoride (UF₆) process gas that circulates through the process equipment, and solid uranium compound deposits that adhere to the equipment surfaces. The quantity of process gas is dependent upon the operating temperatures and pressures and the volume of the process equipment. Solid uranium deposits result from a variety of mechanisms. A solid film of UF₆ condenses on equipment
surfaces when the cascade is processing gas. The quantity of this adsorbed UF$_6$ is dependent on operating temperatures and pressures, surface area, and materials of construction. Because of the large surface area of the barrier, most of the adsorbed UF$_6$ is in the diffusers. Consumption occurs when gaseous UF$_6$ reacts with the metallic surfaces of the process equipment and with impurities present in the process gas and on the equipment surfaces, resulting in the formation of solid uranium compounds. The metal–UF$_6$ reactions produce a relatively diffuse deposit of uranium fluorides. The most common UF$_6$–impurity reaction is with water vapour that enters into the process system with ambient air or is present on equipment surfaces. The water vapour–UF$_6$ reactions produce deposits of UO$_2$F$_2$ that tend to be localized in the immediate vicinity of leaks. Additionally, localized deposits of solid UF$_6$ may be present in areas where process gas freeze-out (desublimation) occurs because of reduced temperatures and pressures. This condition is easily reversible with the application of heat. When a cell is isolated from the cascade and evacuated, the UF$_6$ gas and the majority of the adsorbed UF$_6$ are removed; solid uranium compound deposits and any UF$_6$ freeze-out remain in the equipment.

3. NDA MEASUREMENT TECHNIQUES

Uranium emits four primary forms of radiation: alpha, beta, gamma and neutron. Only neutron and $\gamma$ ray emissions are useful for inventory determination in a diffusion cascade because alpha and beta emissions do not penetrate the equipment walls. A variety of $\gamma$ ray and neutron measurement techniques were evaluated for their effectiveness in quantifying the amount of $^{235}$U, total uranium, and uranium isotope concentrations of the in-process inventory. The $\gamma$ ray methods are based on the measurement of the 185.7 keV $\gamma$ rays emitted as $^{235}$U decays and the 1001 keV $\gamma$ rays emitted as $^{234}$Pa$^m$ (a $^{238}$U daughter) decays. The 185.7 keV $\gamma$ rays penetrate through thin walled items, such as diffusers and piping, but are severely attenuated by thicker walled items. Because of their higher energy level, the 1001 keV $\gamma$ rays penetrate through thicker walled items, but are still attenuated by larger compressors that contain steel with a thickness of up to 5 cm. The quantity of $^{235}$U is proportional to the intensity of the measured 185.7 keV $\gamma$ ray peak; the quantity of $^{238}$U is proportional to the intensity of the measured 1001 keV $\gamma$ ray peak if the inventory is in secular equilibrium. Because the half-life of $^{234}$Pa$^m$ is $\sim$24 d, the inventory must be at steady state for a minimum of about three months. This allows the $^{234}$Pa$^m$ to sufficiently approach secular equilibrium so that the 1001 keV peak can be used to quantify the amount of $^{238}$U or to estimate enrichment.

The $\gamma$ ray measurement instrumentation used in the evaluation consisted of a variety of collimated sodium iodide (NaI) detectors and high purity germanium (HPGe) detectors coupled to multichannel analysers. Primarily, two sizes of NaI detectors were used: 7.6 cm dia. by 7.6 cm thick and 12.7 cm dia. by 7.6 cm thick.
Multiple lead collimators of various shapes (cylindrical and rectangular prism) and thicknesses (0.6–5 cm) were tested to evaluate their effectiveness in reducing the background $\gamma$ ray signal from uranium contained in adjacent process equipment. Two hand-held sodium iodide detectors (2.5 cm dia. by 1.3 cm thick and 2.5 cm dia. by 5.1 cm thick) were used to measure the diffusers, in an effort to test their applicability for inspector measurements. Measurements are typically conducted with the measurement instrumentation located outside of the cell housings because of the extreme temperatures inside.

The neutron method is based on measurement of neutrons emitted during the spontaneous fission of $^{238}$U and the $^{19}$F($\alpha$,n)$^{22}$Na reaction between uranium and bonded fluorine in $\text{UO}_2\text{F}_2$ (deposits) or $\text{UF}_6$. For uranium that has not been reprocessed, the dominant source of alpha particles is from $^{234}\text{U}$, which has the shortest half-life of the naturally occurring uranium isotopes. Neutrons are highly penetrating and can be measured through even the thick walled compressors. The measured neutron signal is assumed to be proportional to the quantity of $^{234}\text{U}$ present. Neutron measurements were conducted with neutron slab detectors coupled to scaler analysers. The neutron detectors used contain multiple $^3\text{He}$ tubes encased in polyethylene shielding to reduce the background signal.

4. FIELD MEASUREMENT CAMPAIGNS

Field measurement evaluations were conducted at gaseous diffusion facilities in the USA and in Argentina to evaluate the performance of the NDA measurement techniques under a variety of conditions. In August 1993, a three week evaluation was conducted at the gaseous diffusion enrichment facilities in Oak Ridge and Portsmouth. Beginning in December 1993, a two month evaluation was conducted at the gaseous diffusion enrichment complex in Pilcaniyeu. The objectives of these field tests were to compare the performance of the NDA measurement methods for accuracy, repeatability, intrusiveness and ease of implementation.

4.1. Enrichment facilities in the USA

The Oak Ridge cascade was shut down in 1985 and the process gas was evacuated. The current inventory is comprised of the residual solid uranium compound deposits (primarily $\text{UO}_2\text{F}_2$) present at the time of shutdown. Measurements were conducted on a partial cell (five stages) of process equipment. Each item was measured several times with the $\gamma$ ray technique and the neutron technique to evaluate measurement repeatability for quantifying the residual solid deposits contained in the empty process equipment. The process equipment was also scanned using NaI detectors to determine the homogeneity of deposits in larger items and to locate isolated deposits in the piping. Because the cascade has been shut down for several years,
the uranium daughters with short half-lives (less than one month) have reached secular equilibrium.

The cascade at the Portsmouth Gaseous Diffusion Plant is in operation; the separation equipment selected for measurement had been operating at steady state for nine months before the start of the measurements. The inventory in this equipment was comprised of gaseous UF₆, absorbed UF₆ on the inner surfaces of the process
equipment, and solid UO₂F₂ deposits. NDA measurements were conducted on a complete cell (ten stages) of process equipment. As the process equipment was being non-destructively measured, the UF₆ gas phase inventory was determined using process data (pressures, temperatures, volumes, and gas sample analysis). The UF₆ gas was then evacuated from the cell and the emptied process equipment was measured. This sequence was repeated twice.

4.2. The Pilcaniyeu Enrichment Complex

At Pilcaniyeu, measurements were conducted on a mock-up module containing twenty full size diffusers. The diffusers had been refurbished and were initially free of solid deposits. During the evaluation, known quantities of UF₆ were incrementally fed to the module so that consumption and the initial formation of internal solid deposits could be measured as the module approached steady state. UF₆ was introduced into the module on 7 December 1993; the module operated in cascade mode (total reflux) until 7 February 1994. Comprehensive measurement campaigns were conducted at the beginning, middle and end of the evaluation. Before the introduction of UF₆ gas, measurements were conducted using each technique to determine the background levels. Incremental measurements were conducted on specific units as UF₆ was introduced. Various quantities of UF₆ were fed and withdrawn during the evaluation to allow the module to be measured at different gas pressures. The NDA instrumentation was placed on platforms that were specially built to position the instruments at the vertical midpoint of the diffusers. Figure 1 shows the measurement of a vertical diffuser with a heavily collimated 7.6 cm by 7.6 cm NaI detector coupled to a multichannel analyser. The lead collimator has a rectangular face to control the measurement geometry. In the background is a second instrument platform, used to position a neutron slab detector, and heavily collimated 7.6 cm by 7.6 cm and 12.7 cm by 7.6 cm NaI detectors near the midpoint of the diffuser. (Note: the mock-up module does not have a cell housing enclosing the process equipment because the entire module is located in an environmentally controlled building.)

5. RESULTS

The results of the NDA measurements were compared with (1) historical process data, (2) measurements of known quantities, and (3) different computational methods (e.g. Monte Carlo or mechanistic simulations). The results can be summarized as follows:

(a) The 185.7 keV γ ray signal and the neutron flux are proportional to the quantities of uranium (²³⁵U and ²³⁴U, respectively) present in an item at the time it is measured; the measured responses change immediately as the quantity of
uranium changes. The 1001 keV $\gamma$ ray signal is proportional to the quantity of $^{238}\text{U}$ present if the material is at equilibrium; the response does not immediately change when the amount of $^{238}\text{U}$ changes.

(b) For the size of US process equipment measured, the majority of the process inventory in an operating cell is dominated by the gas phase. For the conditions evaluated at Pilcaniyeu, there is a smaller difference between the gas phase inventory and the solid phase inventory; under normal operating conditions the process gas represents about two-thirds of the cell inventory.

(c) For operating process equipment, the inventory (gas phase and solid phase) is generally homogeneously spread throughout the item being measured. In empty equipment containing residual deposits, the deposits may be non-homogeneous. Scanning of the equipment may be required to identify the specific location of the deposits.

(d) Preliminary analysis of the data obtained at Oak Ridge and Portsmouth indicates the following: (i) the average measurement repeatability for the various techniques was $\pm 15\%$; (ii) enrichment measurements of the process gas were within 10% of the declared values; (iii) quantitative U and $^{235}\text{U}$ estimates based on $\gamma$ ray measurements were within 15% and 5%, respectively; and (iv) quantitative uranium estimates based on neutron measurements were within 10%.

\begin{table}[h]
\centering
\caption{Measurement Results at Pilcaniyeu}
\begin{tabular}{lllllll}
\hline
Date & $^{235}\text{U} \ (\text{g})$ & & & & & \\
\hline
 & Declared & $12.7 \ \text{cm} \times 7.6 \ \text{cm}$ & $7.6 \ \text{cm} \times 7.6 \ \text{cm}$ & & Neutron & \\
 & & Nal detector & Nal detector & & detector & \\
\hline
1993-12-14 & 546 & 542 & -0.7 & 500 & -8.4 & 583$^b$ & 6.8 \\
1994-01-11 & 512 & 538 & 5.1 & 537 & 4.9 & 603$^c$ & 17.8 \\
1994-01-12 & 512 & 544 & 6.3 & 552 & 7.8 & 490$^c$ & -4.3 \\
1994-02-07 & 669 & 674 & 0.7 & 649 & -3.0 & 786$^c$ & 17.5 \\
1994-02-08 & 441 & 481 & 9.1 & a & 781$^d$ & 77.1 \\
 & & & & & 448$^b$ & 1.7 \\
\hline
\end{tabular}
\end{table}

\begin{itemize}
\item $^a$ Detector failure.
\item $^b$ Assuming all UF$_6$.
\item $^c$ Assuming UF$_6$ plus UO$_2$F$_2$.
\item $^d$ Assuming all UO$_2$F$_2$.
\end{itemize}
(e) The measurement results for Pilcaniyeu are summarized in Table I. Preliminary analysis of the data indicates the following: (i) the measurement repeatability for an entire module based on measurements of individual diffusers was shown to be extremely high; (ii) the quantitative $^{235}\text{U}$ mass estimates based on 185.7 keV $\gamma$ ray measurements were within 10% of the declared values; (iii) quantitative uranium mass estimates based on neutron measurements were shown to be useful for measurement of an entire module, but were not applicable for measuring a single unit; (iv) gamma spectrometry measurements indicate that 95% of the solid deposits are located in the diffusers; and (v) the growth of solid deposits occurs during the first hours of the initial feeding of clean units.

6. CONCLUSIONS

The following conclusions are based on the experience gained in conducting the evaluations and the initial data analysis:

(a) The process inventory can be estimated reasonably well using multiple NDA methods. Scanning of the process equipment with $\gamma$ ray detectors is effective in identifying localized deposits and verifying deposit homogeneity. $\gamma$ ray methods are effective in quantifying material in thin walled process equipment (diffusers and piping), and neutron methods are effective in quantifying material in thick walled process equipment (compressors and valves).

(b) Measurements of both total uranium and $^{235}\text{U}$ are required to determine the current operating conditions. Both the 185.7 keV and the 1001 keV $\gamma$ ray peaks need to be measured and evaluated to provide information on the equilibrium status of the inventory. (A minimum of three months of steady state operation is required for $^{238}\text{U}$ to reach equilibrium.)

(c) The best NDA method for determining the $^{235}\text{U}$ inventory of an individual thin walled item (e.g. a diffuser) is a measurement of the 185.7 keV $\gamma$ ray. The best measurement method for a thick walled item (e.g. a large compressor) is neutron techniques. Both measurement techniques can be used for determining the inventory of a group of items (e.g. a cell). The intensity of the 185.7 keV $\gamma$ ray peak changes proportionally with the magnitude of the inventory and does not require the process to be in equilibrium. Likewise, the neutron signal changes proportionally with the magnitude of the inventory.

(d) The NDA instrumentation used for the inventory measurements must be rugged enough to operate in a difficult, industrial environment. Ambient temperatures outside the cell housings can reach 70°C or more.

(e) For independent verification of the process inventory, the inspecting agency needs to have the capability to confirm the calibration process for converting the measurement data (counts per unit time) to the quantity of uranium or $^{235}\text{U}$. 
7. FUTURE ACTIVITIES

The results from these measurement evaluations are being used to develop an approach that can be used by the IAEA and ABACC to independently verify the process inventory of a gaseous diffusion cascade without compromising the security of proprietary commercial, industrial and technological information. The immediate priority is to complete the analysis and to evaluate the measurement data obtained during the measurement campaigns. In addition, development of NDA measurement methods to be used during inspections is continuing. Evaluation of the utility of lightly shielded $\gamma$ ray detectors is being performed. Additionally, a slab neutron detector is being fabricated which will provide enhanced efficiency for both total counts and coincidence counts, and will minimize the effects of neighbouring diffusers. Both of these measurement methods will be tested in the Pilcaniyeu cascade hall during 1994. Detailed technical briefings are planned for the IAEA and ABACC to present the measurement results and to discuss factors that must be considered when developing a safeguards approach (inspector cost and effort, intrusiveness and safeguards effectiveness). The optimum technique will be selected, and an approach will be developed for the implementation of this technique. A demonstration of the techniques will then be performed for the IAEA and ABACC.

REFERENCES


COMPREHENSIVE APPLICATION OF AN INSPECTION FIELD SUPPORT SYSTEM AT A LOW ENRICHED URANIUM FUEL FABRICATION FACILITY

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Abstract

COMPREHENSIVE APPLICATION OF AN INSPECTION FIELD SUPPORT SYSTEM AT A LOW ENRICHED URANIUM FUEL FABRICATION FACILITY.

The IAEA and the Japanese Government developed an inspection field support system (IFSS) utilizing laptop computers as a part of rationalization of inspection activities at nuclear facilities. In order to perform implementation tests of this system at a low enriched uranium (LEU) fuel fabrication facility, Japan Nuclear Fuel Co., Ltd. (JNF) was selected as the test facility. At the beginning, JNF developed software to provide the floppy disk (FD) media with various information required for the IFSS to verify the nuclear material general ledger at flow verification. JNF became the first facility of the IAEA Member States to introduce this system for the book audit at routine inspection in 1989. Regarding the application of the IFSS for physical inventory verification (PIV), various improvements on operator-provided FD media were done at each PIV since 1990. Consequently, a comprehensive application of this system was accomplished at the 1993 PIV. To date, the IFSS has been functioning well at inspection. As a result of past experience of practical use of the IFSS throughout frequent safeguards inspections, it was confirmed that the system is an effective measure for both the inspectorate and the operator, and the objective of the system is being accomplished as expected by the inspectorate.

1. INTRODUCTION

The IAEA and the Japanese Government have developed an inspection field support system (IFSS) utilizing laptop computers as a part of an effective and efficient measure for safeguards inspection at nuclear facilities [1, 2]. It is essential that information provided by the operator satisfies the requirements of IFSS if it is to be efficiently utilized. The operator’s computer software development is therefore very important for the application.

As a first step, the inspectorate requested the introduction of the IFSS for safeguards inspection at low enriched uranium (LEU) fuel fabrication facilities. In order to perform an implementation test of this system, Japan Nuclear Fuel (JNF) was selected as the test facility. JNF developed software to provide the floppy disk
(FD) media with various information required for the IFSS to verify the nuclear material general ledger at flow verification. JNF became the first facility in the IAEA Member States to introduce this system for the book audit at a routine inspection in early 1989. The paper describing these results was presented at an annual meeting of the Institute of Nuclear Materials Management (INMM) [3].

Trial use of the IFSS for physical inventory verification (PIV) began in 1989. Various improvements, mainly the upgrading of information quality on the PIV itemized list, were made at each PIV since 1990, on a step by step basis. Finally, a comprehensive application of the IFSS was accomplished at the 1993 PIV.

This paper reports the development process of the operators’ software, up to the comprehensive application of the IFSS for PIV, the experience gained and the system’s effectiveness.

2. OUTLINE OF THE MATERIAL CONTROL AND ACCOUNTING SYSTEM

There are four LEU fuel fabrication facilities producing fuel assemblies for commercial LWR nuclear power plants in Japan. JNF, one of these facilities, has

![Diagram of MBA/KMP and material flow]

**FIG. 1. MBA/KMP and material flow.**
750 t U licensed maximum production capacity and has supplied more than 40,000 fuel assemblies to 25 domestic BWR plants since the establishment of the company.

Currently, JNF, as an LEU bulk handling facility, receives five interim inspections and one PIV annually by both the IAEA and the Japanese Government. In 1989, the JNF material control and accounting (MC&A) system for safeguards purposes was changed to a single material balance area (MBA) for the entire process. The material balance for the MBA is maintained by a book-keeping technique comprising determination of inventory change and storage inventory by measuring nuclear materials at three flow key measurement points (KMPs) and at nine inventory KMPs respectively. The material balance control flow in JNF is shown in Fig. 1.

Data processing of the MC&A system is adapted to store the accounting information obtained through the maximum utilization of the facility operating information which is generated throughout the entire production process. This information relates to production control, quality control, safety control, etc. Then these data are processed by a dedicated computer system to generate various records and reports, the nuclear material general ledger, analysis of the facility measurement uncertainties, and is written to FDs for IFSS use.

3. IMPROVEMENT OF OPERATOR PROVIDED FD DATA

In the IFSS, inventory change data and itemized physical inventory data for each stratum are provided to the inspector on a 3½ inch FD for book audit and PIV purposes, respectively. These data are then processed rapidly on the inspector’s own computer. Therefore, the key to the success of an IFSS application depends upon the operator’s capability to provide, in a timely manner, effective data for the inspection activities.

JNF has developed and improved the data processing flow and quality of the FD provided for IFSS, as described below. As a result, file structures of FD data could be standardized and the information required for book audit and PIV could be provided by FD media.

3.1. Development of the data processing system

As a first step to the application of IFSS to inspection activities, adaptability of data transfer between the inspector’s computer system and the operator’s computer system was studied. Then, output data processing to the FD both for inventory change data for book audit and itemized physical inventory data by KMP for PIV were developed. The data processing flow is shown in Fig. 2.
3.2. Improvement of FD data

The most important aspect of the application of the IFSS at the facility is how to provide the operators' accountability data to the inspectorate by means of computer readable media on a routine basis. The key to the success of the IFSS depends on the file structure of the FD data provided by the operator. So JNF spent much time creating the proper file structure and information content, which had to satisfy the IFSS requirements.

The file structure of inventory change data for book audit was determined in 1989. The FD media have to date functioned adequately as a tool for book audit. On the other hand, the file structure of physical inventory itemized data for PIV has been improved, based on practical application during PIVs since 1990 and taking into account the inspectorate's request for file structure modification in the light of
TABLE I. FILE STRUCTURE OF PHYSICAL INVENTORY DATA FOR THE IFSS

<table>
<thead>
<tr>
<th>Column</th>
<th>Type</th>
<th>Length (bytes)</th>
<th>Field description</th>
</tr>
</thead>
<tbody>
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<td>001-004</td>
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<td>4</td>
<td>MBA code</td>
</tr>
<tr>
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</tr>
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<td>Inventory KMP code</td>
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<tr>
<td>017-020</td>
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<td>4</td>
<td>Material description code</td>
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<tr>
<td>022-025</td>
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<td>4</td>
<td>Number of items</td>
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<tr>
<td>027-038</td>
<td>N</td>
<td>12</td>
<td>Enriched uranium weight</td>
</tr>
<tr>
<td>040-051</td>
<td>N</td>
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<td>Enriched uranium isotopic weight</td>
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<td>Natural uranium weight</td>
</tr>
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<td>12</td>
<td>Depleted uranium element weight</td>
</tr>
<tr>
<td>081-085</td>
<td>C</td>
<td>5</td>
<td>Rod type</td>
</tr>
<tr>
<td>087-094</td>
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<tr>
<td>105-116</td>
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<td>12</td>
<td>Net weight</td>
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<tr>
<td>118-129</td>
<td>N</td>
<td>12</td>
<td>Gross weight</td>
</tr>
<tr>
<td>131-142</td>
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<tr>
<td>144-144</td>
<td>C</td>
<td>1</td>
<td>Additional information code</td>
</tr>
</tbody>
</table>

C = character, N = numeric.

inspection activities. The structure was finalized in 1993. The following arrangements and improvements of the data in FD media were made:

(a) Standardized FD file structure at each KMP
(b) Provision of all KMP data by FD media
(c) Addition of as much information as possible on storage location identification
(d) Addition of the store configuration code
(e) Addition of the material description code
(f) Unification of all KMP data including depleted and natural uranium in gram units
(g) Reduction of the total number of FDs to be provided by means of improved data input efficiency.
At this time JNF requested the IAEA and the Japanese Government to modify their IFSS data processing software to cater for the new file structure and the addition code, and to perform the proof test with operator provided test FDs in co-operation with the Nuclear Material Control Centre (NMCC).

As a result of the proof test, it was confirmed that all data on the FD are readable and processable with the inspectorate software. The file structure of the physical inventory data for the IFSS is shown in Table I.

4. EXPERIENCE OF IFSS APPLICATION TO INSPECTION ACTIVITIES

JNF was the first facility to introduce the IFSS for the book audit. However, at an early stage of IFSS application, various unexpected problems were encountered in the software. For example, the operator-provided nuclear material general ledger FD was not readable by the inspectorate’s computer because of insufficient data transfer between the host computer and the FD. Therefore, JNF re-examined its computer software and improved the preparation of the FD. Book audit time has been significantly reduced by the use of the IFSS at routine inspections since late 1989.

The IFSS was introduced for PIV, as an experimental test at the September 1989 PIV, and confirmed the inspector’s computer system ability to read the data from the operator-provided FD. However, the IFSS was not utilized at this PIV. In 1990, the IFSS was applied only for certain KMPs. Later, the amount of inventory data transmission from the FD to the computer increased owing to lack of data storage capacity in the computer, and delayed sampling plan preparation. Also, operator-provided information was insufficient. Hence the advantage of IFSS implementation was smaller than expected.

In 1992, the inspectorate improved its software to read operator-provided inventory information, and timely preparation of the sampling plan became possible. At this time, JNF considered improving the quality of various inventory information taking into account the IAEA requirements, and pursued overall improvement of the FD file structure and contents of the information in order to increase the efficiency of IFSS implementation. Consequently, a comprehensive application of this system was accomplished at the 1993 PIV, and it was confirmed that operator man-hours required for the inspection were reduced by providing the sampling plan in a timely manner.

5. ADVANTAGE OF IFSS IMPLEMENTATION

Currently, the IFSS at JNF contributes to the safeguards inspection activities by the generation of the verification sampling plan, random selection of the sample
<table>
<thead>
<tr>
<th>Item</th>
<th>Comparison of inspection activities</th>
<th>Operator’s advantages</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>BOOK</strong></td>
<td><strong>Operator-provided data</strong></td>
<td><strong>Comparison of inspection activities</strong></td>
</tr>
<tr>
<td></td>
<td>Before introduction of the IFSS</td>
<td>After introduction of the IFSS</td>
</tr>
<tr>
<td></td>
<td>• Hard copy of general ledger (GL)</td>
<td>• FD of GL</td>
</tr>
<tr>
<td></td>
<td>• Hard copy of IC summary (ICS) (IAEA: monthly, JNSB: MBP)</td>
<td>• Hard copy of ICS (IAEA and JNSB: MBP)</td>
</tr>
<tr>
<td></td>
<td></td>
<td><strong>Operator’s advantages</strong></td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Unification of inspection period</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Reduction of man-hours for data preparation</td>
</tr>
<tr>
<td></td>
<td><strong>Inspection procedure</strong></td>
<td></td>
</tr>
<tr>
<td></td>
<td>(1) Verify the material balance during the given period by summing up each IC data item manually</td>
<td>(1) Read the IC data of FD by computer</td>
</tr>
<tr>
<td></td>
<td>(2) Randomly select the IC data from GL manually</td>
<td>(2) Check the output ICS with operator-provided ICS</td>
</tr>
<tr>
<td></td>
<td>(3) Check the IC data with the source data</td>
<td>(3) Randomly selected the IC data on FD by computer</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(4) Check the IC data with source data</td>
</tr>
<tr>
<td></td>
<td><strong>Operator-provided data</strong></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Hard copy of itemized inventory list (IIL) and inventory summary (IS)</td>
<td>FD of IIL</td>
</tr>
<tr>
<td></td>
<td>FD of IIL</td>
<td>Hard copy of new IIL and IS</td>
</tr>
<tr>
<td></td>
<td><strong>Inspection procedure</strong></td>
<td></td>
</tr>
<tr>
<td></td>
<td>(1) Calculate No. of samples for verification manually</td>
<td>(1) Read the IIL data on FD</td>
</tr>
<tr>
<td></td>
<td>(2) Randomly select samples from IIL manually</td>
<td>(2) Output the sampling plan by computer</td>
</tr>
<tr>
<td></td>
<td>(3) Prepare sampling sheets manually</td>
<td>(3) Randomly select samples from FD by computer</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(4) Output sampling sheets by computer</td>
</tr>
</tbody>
</table>

**GL** = General ledger.
**IC** = Inventory change.
and output of various data sheets required for the verification activities. Also, the system contributes to verification of the nuclear material general ledger at the interim inspection and the PIV, including consistency checks and the closing inventory of the last accounting period and the opening inventory of the current accounting period, and to comparison of the ledger and the associated source document.

A comparison of safeguards inspection activities before and after introduction of the IFSS and the operators’ advantages from the implementation of the IFSS are shown in Table II.

6. CONCLUSION

Comprehensive application of the IFSS at the JNF LEU fuel fabrication facility was accomplished on achieving the practical implementation of the IFSS for PIV, following its practical use in book audit. Regarding the application of the IFSS for PIV, major improvements can be summarized as follows:

1. Standardization of the file structure of FD data for all KMPs.
2. Establishment of a special identification code to discriminate inventory information in order to meet inspection activities at a given KMP, leading to more practical and efficient utilization for the inspection.

The various data required for safeguards inspections can be processed rapidly by the IFSS, so we believe that the introduction of the IFSS and its practical use will become more important in achieving efficient inspection activities, in particular at large size bulk handling facilities in which there are normally large amounts of accounting data for verification. Experience of the comprehensive application of the IFSS at JNF shows that the system is an effective measure for both the inspectorate and the operator. The main advantages of IFSS application from the operator’s point of view are:

1. Reduction of operator man-hours required for inspection.
2. Standardization of the verification procedure by computerization.

Utilization of the IFSS should not be limited to book audit and PIV purposes. We believe that more use should be made of the IFSS for further streamlining inspection activities by adoption of a new safeguards approach, such as the mail box concept for short notice random inspection, the fuel cycle approach, process monitoring, etc. Also, the IFSS will become a more important inspection tool at fully automated paperless facilities to collect associated information for inspection activities since most of the accounting data will be stored directly in the computer database. We at JNF would like to continue co-operation with the safeguards authorities to improve the efficiency of safeguards inspection, which may provide significant advantages for both the inspectorate and plant operators.
ACKNOWLEDGEMENT

The authors wish to express their gratitude to H. Yosida of the Nuclear Material Control Centre, who gave them much support and aided co-operation between the IAEA and the Japanese government authorities, as well as helping to conduct various implementation tests.

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FIELD TEST OF SHORT NOTICE RANDOM INSPECTIONS FOR INVENTORY CHANGE VERIFICATION AT A LOW ENRICHED URANIUM FUEL FABRICATION PLANT

Preliminary summary*

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Abstract

FIELD TEST OF SHORT NOTICE RANDOM INSPECTIONS FOR INVENTORY CHANGE VERIFICATION AT A LOW ENRICHED URANIUM FUEL FABRICATION PLANT: PRELIMINARY SUMMARY.

An approach of short notice random inspections (SNRIs) for inventory change verification can enhance the effectiveness and efficiency of international safeguards at natural or low enriched uranium (LEU) fuel fabrication plants. According to this approach, the plant operator declares the contents of nuclear material items before knowing if an inspection will occur to verify them. Additionally, items about which declarations are newly made should remain available for verification for an agreed time. A six month field test of the feasibility of such SNRIs took place at the Westinghouse Electric Corporation Commercial Nuclear Fuel Division. Westinghouse personnel made daily declarations about both feed and product items, uranium hexafluoride cylinders and finished fuel assemblies, using a custom designed computer ‘mailbox’. Safeguards inspectors from the IAEA conducted eight SNRIs to verify these declarations. Items from both strata were verified during the SNRIs by means of non-destructive assay equipment. The field test demonstrated the feasibility and practicality of key elements of the SNRI approach for a large LEU fuel fabrication plant.

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1. INTRODUCTION

Short notice random inspections (SNRIs) for inventory change verification were first analysed theoretically by Gordon and Sanborn [1] following a suggestion by Brenner [2]. They and others were studying schemes for the verification of inventory changes at centrifuge enrichment plants [3].

For fuel fabrication plants dealing only with natural or low enriched uranium (LEU), the basic idea is that safeguards inspectors of the IAEA could verify the feed to and production of a plant by conducting a number of short notice or unannounced inspections scheduled by random selection throughout the material balance period. If certain conditions were met and the results of the SNRIs supported the declarations of the State and plant operator, then the declarations could be accepted for the entire material balance period — not just for the nuclear material present at inspections.

Such SNRIs offer a significant increase in effectiveness when compared to equal numbers of scheduled inspections that do not provide complete coverage of inventory changes during a material balance period. This is so even if the diversion detection probability, which depends on several factors, is small.

The Westinghouse Electric Corporation Commercial Nuclear Fuel Division Fabrication Facility, located in Columbia, SC, USA, was the site of a field test to determine the procedural feasibility of this approach [4]. Nuclear reactor fuel assemblies are made there from natural and LEU hexafluoride.

By agreement, the field test concentrated solely on verifications of uranium hexafluoride cylinders and finished fuel assemblies, by far the major part of the transfer verifications required by the safeguards criteria [5] for fuel fabrication plants.

This paper is a preliminary summary of a much longer report describing the field test and its underlying theory [6].

2. THEORY OF SHORT NOTICE RANDOM INSPECTIONS: INFEREN CE CONDITIONS

Three conditions must be met for the validity of a statistical inference about a population based on verification of a random sample [1, 5]:

(1) All items in the population must be available for selection for verification.

(2) The plant operator must declare to the IAEA values for the nuclear material content of items before knowing which items will be verified.

(3) The operator must not alter item identity or content after learning that an item is chosen for verification and before the verification actually occurs.

To fulfil condition (1) for a flow stratum encompassing inventory changes, an SNRI approach incorporates a set of possible inspection dates. These opportunities
should be frequent enough to allow verification of all items. Actual inspection dates are randomly chosen from these opportunities.

The 'mailbox' concept is used to fulfil condition (2) [1, 2, 7, 8]. Mailbox declarations are unalterable operator statements of accountancy values against which IAEA inspectors can compare the results of verification measurements.

Thus, condition (2) is satisfied by virtue of a mailbox to which the plant operator is regularly submitting inventory change data. Condition (1) is satisfied if some residence time for verification remains after the mailbox declaration for each item, and if this remaining residence time overlaps with an inspection opportunity. If, additionally, condition (3) can be met by adequate measures, then IAEA inspectors can select a random sample to verify during SNRIs and, on the basis of the verification results, make statistical inferences about the entire population [1, 5, 9–12].

These points are graphically explained by Fig. 1. After mailbox posting of their nuclear material content, only batches 11 to 13 of fuel assemblies were in residence for possible verification at the time of the beginning of the second inspection;
these batches would constitute the population for random sampling during the second inspection. Batches 14 and 15 could be verified at a third inspection.

Lu et al. [13] have pointed out another possibility for conducting SNRIs to satisfy condition (1).

3. FIELD TEST RESULTS

The mailbox proposal implemented for the field test consisted of a redundant system of an IAEA computer mailbox at the Westinghouse plant supplemented by telefax transmissions to Vienna. The computer was a Gateway 2000 486/33 desktop. It sat inside a specially fabricated anodized aluminium containment box that was sealed shut. One penetration permitted access to the ‘B’ floppy disk drive and another permitted access for the power cable and the communications cable between the computer and the monitor. The keyboard for the computer remained inside the containment.

For the duration of the field test, Westinghouse staff daily turned the mailbox computer on and responded to an automatically executing program to submit inventory change data by diskette. During the entire six month period, the mailbox recorded about 2700 transactions involving about 1000 assemblies. (Details of these results primarily concern assemblies.)

Four transaction events were specified as useful for the mailbox data: ‘births’, ‘changes’, ‘deaths’ and ‘shipments’. Dates, identifications and accounting information were submitted for each event, and the computer mailbox itself also dated each entry. Cylinder receipts and assembly production constituted births; connection of cylinders to the plant process and packing of assemblies constituted deaths; item accounting amendments constituted changes; and shipments applied to assemblies only.

One criterion for the success of SNRIs is the promptness of mailbox declarations. For assemblies produced after February 1993, about 80% of the entries were made one day after the event and about 15% three days after the event. The latter corresponds to the delay occurring because of weekends, when submissions were not made. Were the declarations delayed, items might be physically ‘dead’ before ‘birth’ declarations. This did not happen. Indeed, the data indicated ‘deaths’ and ‘birth’ declarations on the same day for only about ten assemblies — about 1% — during the testing period.

From March to August 1993, IAEA inspectors conducted eight SNRIs at the Westinghouse plant, arriving unannounced. Great efforts were made to avoid premature disclosure.

During SNRIs, IAEA inspectors unsealed the computer mailbox containment and obtained access to the keyboard. They thereupon extracted the mailbox data entered since the previous SNRI. Finally, they closed and sealed the containment.
With these data, the IAEA inspectors began operating special SNRI preprocessing software on a portable computer; it combined the mailbox data with the physical inventory determined at the previous SNRI to yield a new physical inventory for the SNRI in progress. The software highlighted those items whose residence time extended beyond the beginning of the SNRI in progress. From these, the inspectors randomly selected several for verification.

Armed with the mailbox inventory, plus the facility physical inventory lists (PILs) requested immediately after arrival, the IAEA personnel then went to the two relevant locations, the uranium hexafluoride storage pad and the fuel assembly storage area; Westinghouse personnel escorted them. They performed item counts and identifications and affixed temporary seals to the available items selected for measurement verifications. Most of the measurement verifications took place after the first day of the SNRIs.

From the time of arrival at the plant, the inspectors took about three hours on average (not including the SNRIs when this activity was deferred until the second day) to identify the fuel assemblies to be verified. Plant personnel supplied the PILs in about 1.25 hours on average.

The inspectors compared the mailbox data with the PILs and with the actual items on inventory. Discrepancies arose because of the dynamic nature of the plant at the time of the SNRIs, preprocessor software limitations, the inaccessibility of certain items and data errors. To understand the discrepancies, the IAEA personnel examined source documents and received additional information from the Westinghouse staff.

An important factor under study was the achieved residence time for verification, i.e. the number of days between assembly production and packing (Fig. 2). About 1.5% of the assemblies that were packed had a residence time of one day; about 20% had a residence time of four days or fewer; and about 40% had a residence time of seven days or fewer. If used on an a priori basis, these achieved values permit the calculation of detection probabilities as a function of future SNRI frequency [13]. Planning values for the residence times for verification were seven days for February to June and four days for July and August. These planning values were not met as minima because of plant operational needs.

On average about 160 fuel assemblies were on inventory at the SNRIs, according to mailbox data. For about 24 of these assemblies, the actual residence time had not exceeded the planning value selected for the SNRI. These were identified in the assembly storage area by tag check and, sometimes, serial number. A sample of them was further verified, about three by attributes non-destructive assay (NDA; gamma ray check) and about two by variables NDA (neutron collar [14]).
FIG. 2. Achieved fuel assembly residence times (days from production to packing).
4. CONCLUSIONS

Safeguards inspectors from the IAEA conducted eight test SNRIs at the Westinghouse plant from March to August 1993. The inspectors appeared completely unannounced and within a short time began inspection procedures that lasted two or three days.

Daily throughout the test period, Westinghouse staff members supplied to the IAEA inspectors data about inventory changes for feed uranium hexafluoride and product fuel assemblies. Data were both transmitted to the IAEA by facsimile and submitted to a custom designed computer 'mailbox' located at the Westinghouse plant. Both data routes functioned well.

The evaluation of the SNRI field test is not yet complete. Nevertheless, these accomplishments show that two main elements of the SNRI concept — the SNRIs themselves and the mailbox — are feasible and practical for a large LEU fuel fabrication plant.

The field test results also lead to many recommendations concerning the further development and possible implementation of SNRIs for inventory change verification. These recommendations encompass policy considerations, SNRI concepts and procedures, and mailbox hardware and preprocessor software.

ACKNOWLEDGEMENTS AND DEDICATION


This paper is dedicated to the memory of Leon Green, who led the International Safeguards Project Office at Brookhaven National Laboratory from 1978 to 1991.

REFERENCES


As a continuation of the measurements with uranium bulk samples that led to the development of the NDA method for uranium concentration measurements [1, 2], new measurements were made with the same uranium samples plus an additional $^{57}$Co excitation source.

The $^{57}$Co source was a ring shaped commercial source used for uranium concentration measurements in liquid uranium samples. The source was located on a planar germanium detector, and the tungsten walls of the source were constructed in such a way that they shielded the germanium detector, but the $^{57}$Co $\gamma$ rays could penetrate into the uranium samples positioned on the source. The excited uranium X rays were detected by the germanium detector through a central hole in the source.

The intensities of four uranium roentgen peaks were measured in uranium samples with different uranium concentrations (Table I).

It was found that, in contrast to the well known method of uranium concentration measurements in liquid uranium samples, based on measurements of uranium X rays excited by a similar external $\gamma$ source, the intensity of uranium X rays excited by $^{57}$Co $\gamma$ rays in uranium powder samples does not depend on the uranium concentration.

However, using the passive technique of uranium X ray measurements in the same bulk uranium powder samples, without any external source, a direct dependence of the uranium X ray intensities on the uranium concentration was found [1, 2]. An explanation of this effect can be given by a mathematical description of the uranium X ray intensities in different samples.

The intensities of the uranium X rays excited by an external $^{57}$Co $\gamma$ source can be written as:

$$I_X = S_0 \frac{C_U}{[(\mu_U^X + \mu_U^\gamma) - (\mu_m^X + \mu_m^\gamma)] C_U + (\mu_m^X + \mu_m^\gamma)}$$

where $C_U$ is the uranium concentration, $S_0$ is the intensity of the $^{57}$Co source, and
TABLE I. MEASUREMENTS OF SAMPLES OF DIFFERENT URANIUM CONCENTRATIONS WITH A $^{57}$Co SOURCE

<table>
<thead>
<tr>
<th>Uranium composition</th>
<th>Uranium concentration</th>
<th>Uranium X rays</th>
<th>94 keV</th>
<th>98 keV</th>
<th>111 keV</th>
<th>114 keV</th>
<th>94 keV/114 keV ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>U$_3$O$_8$</td>
<td>82.36%</td>
<td>404.42</td>
<td>653.08</td>
<td>230.65</td>
<td>75.19</td>
<td>5.38</td>
<td></td>
</tr>
<tr>
<td>UO$_3$</td>
<td>81.31%</td>
<td>406.49</td>
<td>659.40</td>
<td>230.54</td>
<td>75.21</td>
<td>5.41</td>
<td></td>
</tr>
<tr>
<td>Yellow cake</td>
<td>73.45%</td>
<td>412.26</td>
<td>661.37</td>
<td>233.74</td>
<td>76.76</td>
<td>5.37</td>
<td></td>
</tr>
<tr>
<td>Yellow cake</td>
<td>69.66%</td>
<td>415.39</td>
<td>665.45</td>
<td>236.68</td>
<td>76.51</td>
<td>5.43</td>
<td></td>
</tr>
<tr>
<td>UO$_2$ + Al$_2$O$_3$</td>
<td>60.38%</td>
<td>405.42</td>
<td>652.48</td>
<td>229.35</td>
<td>75.04</td>
<td>5.40</td>
<td></td>
</tr>
</tbody>
</table>

$\mu_X^U$, $\mu_\gamma^U$, $\mu_X^m$, $\mu_\gamma^m$ are the coefficients of X rays and $\gamma$ ray absorption in uranium and matrix material.

Usually, the uranium concentrations in liquid uranium samples are very small:

$$C_U \ll \frac{\mu_X^U + \mu_\gamma^U}{(\mu_X^U + \mu_\gamma^U) - (\mu_X^m + \mu_\gamma^m)}$$

Therefore, Eq. (1) can be transformed to

$$I_X = C_U \frac{S_0}{\mu_X^U + \mu_\gamma^U}$$

(2)

This is a linear dependence of the uranium X ray intensities on the uranium concentrations in the solutions.

The uranium concentrations in powders are much higher than those in liquid samples:

$$C_U \gg \frac{\mu_X^U + \mu_\gamma^U}{(\mu_X^U + \mu_\gamma^U) - (\mu_X^m + \mu_\gamma^m)}$$

Therefore, Eq. (1) can be transformed to

$$I_X = \frac{S_0}{\mu_X^U + \mu_\gamma^U}$$

(3)

This explains the non-dependence of the uranium X ray intensities on the uranium concentrations in powder samples.

A similar analysis could be done to explain the dependence of the passive X ray intensities on the uranium concentration in powder samples.
Passive X rays in uranium powder samples can be excited by \( \alpha \) and \( \beta \) particles and \( \gamma \) rays from the \( \alpha \) decay chains of \( ^{235}\text{U} \) and \( ^{238}\text{U} \):

\[
I_X = I^X_\alpha + I^X_\beta + I^X_\gamma + I^X_{234\text{U}}
\]

where \( I^X_\alpha, I^X_\beta, I^X_\gamma \) are the intensities of X rays excited by different types of radiation, and \( I^X_{234\text{U}} \) is the intensity of \( ^{234}\text{U} \) X rays from the decay of \( ^{234}\text{Pa} \).

For an infinitely thick sample, the X ray intensities are

\[
I^X_i = \rho U \int_0^{\infty} K_i S_i(X) \exp \left[ - \left( \rho_U \mu^X_U + \rho_m \mu^X_m \right) X \right] \, dx
\]

where \( I^X_i \) are the X ray intensities excited by \( \alpha, \beta \), and \( \gamma \) radiation in a layer \( dx \), located at the distance \( x \) from the germanium detector; \( \mu^X_U, \mu^X_m \) are the coefficients of the X ray absorption in uranium and in matrix materials; \( \rho_U, \rho_m \) are the densities of uranium and of matrix materials; \( K_i \) is a constant; and \( i \) is the type of the exciting radiation (\( \alpha, \beta, \gamma \)).

Then, for each type of the exciting radiation the X ray intensity is

\[
I^X_i = \frac{C_U}{(\mu_U - \mu^X_m) C_U + \mu^X_m} \left[ C^2_{235\text{U}} K_{i1} + K_{i2} \right] \frac{1}{1 + \frac{\rho_m \mu^X_m}{\rho_U \mu^X_U}}
\]

where \( C_U \) is the uranium concentration in the sample, \( C^2_{235\text{U}} \) is the \( ^{235}\text{U} \) enrichment, and \( K_{i1}, K_{i2} \) are constants.

The intensity of \( ^{234}\text{U} \) X rays is

\[
I^X_{234\text{U}} = (1 + C^{235\text{U}}) K \frac{1}{1 + \frac{\rho_m \mu^X_m}{\rho_U \mu^X_U}}
\]

and it does not depend on the uranium concentration in the sample.

The contribution of different exciting radiations to the total intensity of the passive X rays in uranium samples will depend on the absorption coefficients \( \mu_U \) and \( \mu^X_m \) of this type of radiation (\( \alpha, \beta, \gamma \)) in uranium and in matrix material. For example, for \( \beta \) particles the values of \( \mu_U \) and \( \mu^X_m \) are comparable, and there is a strong dependence of the intensity of the X rays excited by \( \beta \) particles on the uranium concentration.

However, for 185.7 keV \( \gamma \) rays of \( ^{235}\text{U} \), \( \mu_U \gg \mu^X_m \) for most uranium compositions. Therefore, only a weak dependence on the uranium concentration is observed. For the higher \( \gamma \) ray energies of \( ^{238}\text{U} \) (1001 keV) this dependence becomes stronger. Thus, for samples with different \( ^{235}\text{U} \) enrichment, the contribution of \( \alpha, \beta, \gamma \) excitation sources to \( I_X \) could be different.
Nevertheless, for most cases in safeguards practice, a reasonably simplified expression can be used to verify the uranium concentration and enrichment simultaneously in unknown samples.

Using the well known equation

$$I_{185.7\text{ keV}} = aC_{235\text{U}} + b$$  \hspace{1cm} (8)

Eq. (6) can be transformed to

$$C_{\text{U}} = \frac{K_1 I_{98.4\text{ keV}} + K_2 I_{185.7\text{ keV}} + K_3}{I_{185.7\text{ keV}} + K_4}$$  \hspace{1cm} (9)

where $I_{98.4\text{ keV}}$ and $I_{185.7\text{ keV}}$ are the measured intensities of 98.4 keV U X rays and 185.7 keV $^{235}\text{U} \gamma$ rays.

### TABLE II. URANIUM CONCENTRATION AND ENRICHMENT MEASUREMENTS

<table>
<thead>
<tr>
<th>$^{235}\text{U}$ enrichment (%)</th>
<th>U concentration (declared) (%)</th>
<th>Measured intensities</th>
<th>U concentration (calculated) (%)</th>
<th>U concentration ((decl.-calc.)/decl.) (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>4.46</td>
<td>84.66</td>
<td>46.19</td>
<td>36.22</td>
<td>85.80</td>
</tr>
<tr>
<td>2.95</td>
<td>84.52</td>
<td>36.43</td>
<td>23.94</td>
<td>85.23</td>
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<tr>
<td>1.94</td>
<td>84.56</td>
<td>29.53</td>
<td>15.64</td>
<td>83.70</td>
</tr>
<tr>
<td>0.71</td>
<td>84.53</td>
<td>21.68</td>
<td>5.73</td>
<td>82.79</td>
</tr>
<tr>
<td>19.82</td>
<td>84.55</td>
<td>137.43</td>
<td>157.29</td>
<td>82.31</td>
</tr>
<tr>
<td>3.11</td>
<td>84.78</td>
<td>37.67</td>
<td>24.95</td>
<td>86.63</td>
</tr>
<tr>
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<td>85.03</td>
<td>21.69</td>
<td>5.74</td>
<td>82.80</td>
</tr>
<tr>
<td>0.23</td>
<td>84.70</td>
<td>19.14</td>
<td>1.84</td>
<td>84.75</td>
</tr>
<tr>
<td>19.88</td>
<td>87.30</td>
<td>143.55</td>
<td>160.45</td>
<td>86.33</td>
</tr>
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<td>35.00</td>
<td>87.59</td>
<td>242.56</td>
<td>284.39</td>
<td>87.75</td>
</tr>
<tr>
<td>45.55</td>
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<td>366.00</td>
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</tr>
<tr>
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<td>596.68</td>
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<td>92.42</td>
<td>75.60</td>
<td>538.84</td>
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<td>75.22</td>
</tr>
<tr>
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<td>648.11</td>
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<td>68.64</td>
</tr>
<tr>
<td>0.71</td>
<td>60.38</td>
<td>15.87</td>
<td>5.27</td>
<td>60.25</td>
</tr>
</tbody>
</table>
After calibration measurements with different uranium samples, the uranium concentration and the $^{235}$U enrichment can be determined. As an example, the fitting based on Eq. (9) was made using the results of the previous uranium sample measurements [1] (Table II).

For verification of the samples in the containers, the wall attenuation factors should be estimated for 98.4 keV X rays and 185.7 keV $\gamma$ rays. Sometimes, the 144 and 185.7 keV $\gamma$ rays of $^{235}$U can be used for this, but they should be corrected for any unknown uranium concentration, which could be very different in unknown samples.

The non-dependence of uranium X rays on the uranium concentration in powder sample measurements with an external $\gamma$ source can be used to make a correction for the X ray absorption in the walls of different containers. It then becomes possible to measure the uranium concentration and the $^{235}$U enrichment in uranium bulk samples through the wall of a container without knowledge of the wall thickness or the type of material used. A combination of the two types of uranium X ray measurements (passive measurements and measurements with an external $^{57}$Co excitation source) can be used for this measurement.

If the intensity of uranium X rays excited by a $^{57}$Co source does not depend on the chemical composition of the sample, then the difference in their intensities, compared with the standard sample measurement, can be used to calculate the attenuation effect in the container wall for uranium X rays and 185.7 keV $\gamma$ rays.
A number of measurements with different types of absorbers (Al, Fe, Cu, Cd) and with the same uranium sample were made at the Seibersdorf Analytical Laboratory to demonstrate this technique. Figure 1 shows the results of measurements of the 98.4 keV/185.7 keV ratio made without an external source, and the results of measurements of the 94.2 keV/144 keV ratio made with the $^{57}$Co source.

REFERENCES


NEW EXPERIENCE AND INITIATIVES

(Session 5)

Chairman

D. PERRICOS
IAEA
IAEA ACTIVITIES AND EXPERIENCE IN IRAQ UNDER THE RELEVANT RESOLUTIONS OF THE UNITED NATIONS SECURITY COUNCIL

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Abstract

IAEA ACTIVITIES AND EXPERIENCE IN IRAQ UNDER THE RELEVANT RESOLUTIONS OF THE UNITED NATIONS SECURITY COUNCIL.

United Nations Security Council Resolution 687 (1991) mandated, inter alia, the destruction of all weapons of mass destruction — chemical, biological, ballistic and nuclear — existing in Iraq, including equipment, facilities and materials used for their production. Resolution 715 (1991) adopted an open-ended plan for ongoing monitoring and verification aimed at preventing a reconstitution of Iraq's capabilities in the production of weapons of mass destruction. Under these resolutions the IAEA was given responsibility to implement the Security Council mandate in the nuclear area, with the assistance and co-operation of the United Nations Special Commission. The paper provides an overview of the IAEA's activities in Iraq under United Nations Security Council resolutions and offers some comments on the lessons to be learned.

1. INTRODUCTION

On 6 April 1991 the United Nations Security Council adopted Resolution 687, the so-called cease-fire resolution which, inter alia, mandated the destruction of all weapons of mass destruction — chemical, biological, ballistic and nuclear — existing in Iraq. The IAEA was given sole responsibility under this resolution to destroy, remove or render harmless not only nuclear weapons but also any existing capability to acquire them, including prohibited precursor materials such as enriched uranium, plutonium and all facilities, equipment and materials used for their production.

On 15 April 1991 the Director General created an Action Team to implement Resolution 687: specifically to pursue the identification and elimination of activities in Iraq aimed at the acquisition of nuclear weapon capability and the ongoing monitoring and verification of Iraq to ensure no resumption of a nuclear weapons programme.
2. THE SPECIAL NATURE OF IAEA POST-GULF-WAR ACTIVITIES IN IRAQ

It must be noted that the cease-fire resolution, which Iraq was obliged to accept, provided the IAEA with the legal basis to implement a countrywide search of a highly intrusive nature (the any-place-any-time approach), with inspection teams being granted rights similar to those of an occupying army. Any attempt made by the Iraqi side to limit or ignore these rights has met with an immediate, firm reaction by the UN Security Council, and, in a couple of cases, with military retaliation. The exercise of these rights has played an important role in the discovery and dismantling of the clandestine programme. While it is not conceivable that similar rights might be granted under ordinary circumstances, there are other ingredients which contributed to success in Iraq, the inclusion of which in the search for undeclared activities and material in other States may be less objectionable. As examples, one may mention the access to information provided by third parties and the extensive use of modern analytical techniques in environmental sampling.

3. RESULTS TO DATE: AN OVERVIEW

Twenty-three IAEA nuclear inspection teams have so far visited Iraq to inspect facilities, interview key personnel, inventory nuclear materials, identify prohibited items and carry out destruction and removal operations.

The Iraq nuclear weapons programme was carried out at nine dedicated sites. On these sites there were dozens of major processing buildings that represented an investment equivalent to several billion US dollars. Most of these buildings were either destroyed during the war or demolished under inspection teams’ supervision. What now remains at these sites is offices, warehouses and light buildings with no unique capabilities.

The programme involved millions of dollars’ worth of specialized equipment. Much of this equipment was destroyed either by the war or by inspection teams. The IAEA teams have destroyed, removed or rendered harmless over 1900 individual items as well as 600 tonnes of sensitive alloys useful in a nuclear weapons programme or in uranium enrichment activities.

The clandestine Iraqi nuclear programme consisted of a diversified and well financed approach to developing multiple techniques for the production of highly enriched uranium (HEU). The Gulf war, and subsequent inspections, stopped this production effort well before any significant amount of such material was produced. All identified facilities, and equipment related to fissile material production, were destroyed or rendered harmless either during the war or under the supervision of IAEA inspection teams.

At the time of the Gulf war a significant amount of HEU in the form of fresh (i.e. unirradiated) fuel was in Iraq under IAEA safeguards. There existed as well
HEU exceeding a significant quantity (SQ) in the form of irradiated fuel. There was concern that this material could have been diverted to produce a nuclear explosive. Pre-war estimates that Iraq could produce a nuclear explosive within a few months explicitly assumed that this material could be diverted. It was not. All this material has been accounted for by the IAEA. The fresh fuel was removed from Iraq in November 1991 and the removal of the irradiated fuel — a much more complex operation — was completed in February 1994.

Of continuing concern is the possible existence of a clandestine stockpile of illicitly obtained HEU or plutonium. Although no such stockpile has been discovered it was a major consideration in deciding to destroy facilities and equipment that could fabricate HEU or plutonium into a nuclear explosive.

Assuming that the programme had continued along the same lines, keeping its original goals, it is estimated that it would have taken several more years to complete a nuclear weapon. Much of the programme was just being put together at the time of the Iraqi invasion of Kuwait and many pieces were missing. It does not appear that these pieces were coming together in a well-organized way and the programme had a long way to go both in terms of organization and technical progress.

Actually, the weapons programme relevant facilities and plants were suggestive of a grandiose and over-designed programme. If a political decision were taken today to produce a crude weapon (source of fissile material unspecified), it could be done without many of the specialized facilities that had been built in Iraq for weaponization. Good equipment would be needed, however, and much of what Iraq did acquire has been destroyed.

The theoretical aspect of the weapons development studies is a worry despite the current sanctions and monitoring regime. The low visibility of this work prevents the IAEA from assessing that it is not continuing. Indeed, Iraq's progress in the theoretical aspects could lead to a more efficient experimental programme in the future if, for any reason, Iraq were to resume weapons development. While the more visible activities, such as fissile materials production, fabrication studies and testing, are impeded and deterred by the inspections, calculations, simulations and design efforts could continue with little risk of detection.

Another remaining key element in this context is the technical experience that has been gained to date. If this expertise is held together, small scale research activity, for instance in centrifuge design, may continue with a low probability of being discovered. These are low signature activities not likely to be revealed to inspectors without extraordinary luck or the defection of knowledgeable Iraqi personnel.

A principal purpose of the ongoing monitoring and verification regime, now being phased in, is to prevent any reconstitution of indigenous programmes for uranium enrichment or plutonium production. Such activities are high profile in terms of cost, visibility and foreign procurements. All require continuing observation.
3.1. The identification phase

The identification of the various elements of the clandestine Iraqi programme was largely completed by the end of September 1991, i.e. six months after the adoption by the Security Council of the cease-fire resolution. Charting the map of this programme has entailed a number of difficulties, including — on some occasions — dramatic confrontations between Iraqi authorities and IAEA inspection teams. During this phase the Iraqi Government employed a strategy of obstruction and delay in its efforts to conceal the real nature of its nuclear projects, while, on the other hand, demonstrating a level of co-operation in some less sensitive areas.

As to the completeness of the picture obtained, it is the considered opinion of the IAEA, based on the results of 23 inspection missions, the analyses of thousands of samples, the evaluation of several hundred documents confiscated in Iraq, the assessment of procurement and other information obtained from Member States of the IAEA, that the essential components of the clandestine programme have been identified. Even if the picture may lack some detail, it is consistent, with no obvious gaps.

Success in the rapid identification of the secret Iraqi programme is due, in no small measure, to the support extended to the IAEA by its Member States through the provision of intelligence information and of experts who expanded the competence of the IAEA inspection teams in particular areas. The combination of intelligence information and experts with rapid and intrusive field inspections to verify and follow up intelligence leads has proved to be a very powerful tool in achieving success. This is an important lesson learned from the Iraqi experience.

3.2. Destruction and removal of prohibited items

The second main task laid upon the IAEA by Resolution 687 concerns the destruction, removal or rendering harmless of the essential ingredients of the Iraqi nuclear weapons development programme, including the nuclear-weapons-usable material known to have been in Iraq in the form of safeguarded reactor fuel.

Extensive destruction of Iraqi nuclear installations occurred during the Gulf war as a result of the air raids by Coalition forces. Additional destruction of equipment and material was carried out by the Iraqi army at the end of the war and prior to the start of IAEA inspections, in an attempt by Iraq to remove evidence of its secret programme. Also, important equipment, machine tools, instrumentation, spare parts, stock materials and components were salvaged and hidden by Iraqi personnel. An important exception was Al Atheer, the site where weaponization activities had been planned. Facilities at this site were still under construction at the time

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1 See also Appendix 1.
2 See also Appendix 2.
of the war, which is probably one of the reasons it went practically unscathed through the conflict. The association of this site, as well as several others, with the clandestine Iraqi programme was not known at the time of the Gulf war to either national or international authorities. The role of these facilities was only uncovered through the inspection process.

Since September 1991, i.e. when the scope of the clandestine Iraqi programme came into focus, the IAEA has been supervising the systematic destruction of facilities, technical buildings, equipment and other items proscribed under UN Security Council Resolution 687 which had escaped destruction or which had been only slightly damaged. This process is practically completed, although the possibility of finding more prohibited items cannot be entirely ruled out and surprises are always possible. As to the quantities of weapons-usable nuclear material (HEU in the form of reactor fuel elements) known to have been in Iraq under IAEA safeguards, these were found untouched and have been fully accounted for. The removal operation of this special nuclear material as well as of a small amount (6 g) of separated plutonium has been completed and has entailed, inter alia, a complex technical effort to clear part of the enriched fuel elements from the rubble of a bombarded research reactor.

3.3. Ongoing monitoring and verification activities

On 27 November 1993 Iraq formally accepted Security Council Resolution 715, adopted in November 1991, which established the basis of an open-ended monitoring and verification plan. This plan is aimed at preventing the reconstitution by Iraq of its nuclear weapons programme. Iraq's acceptance of the resolution three years after its adoption represents a major step towards Iraq's fulfilment of all obligations imposed by the Security Council resolutions.

From the time this resolution was adopted in November 1991 the IAEA started implementing actions directly relevant to the ongoing monitoring and verification, such as material accountancy and containment measures. In 1992, a periodic radiometric survey of the main water bodies in Iraq was started. This was sensitive enough to detect radioactive and other signatures that would reveal the existence of undeclared nuclear activities. The first series of results show that no unknown nuclear facility has been operating in Iraq in the last three years. On the basis of verification activities and available information, it is reasonable to conclude that Iraq is not operating clandestine nuclear facilities, in particular a reactor or a reprocessing plant.

The IAEA will now gradually phase in all the elements of the ongoing monitoring and verification plan, which consists of a package of activities and techniques such as:

- on-site inspections, both routine and unannounced;
- extended and continuous presence of inspectors in Iraq;
— close control of the natural uranium inventory remaining in Iraq;
— use of seals, tags and video surveillance;
— monitoring dual-use items (equipment and materials);
— periodic environmental sampling of air, water, soil, biota and vegetation;
— use of advanced sensors and systems to detect signatures of prohibited activities (enrichment, reprocessing, weaponization) on mobile land-based and aerial platforms;
— satellite and low-altitude imagery;
— import/export monitoring;
— power line monitoring;
— interviews with key personnel;
— assessment and follow-up of information received from Member States.

4. LESSONS LEARNT

The events in Iraq have not only highlighted the need to strengthen the IAEA safeguards system — and, in fact, the non-proliferation regime as a whole — but have also heightened the readiness of Governments to contribute to these improvements.

During 1992 and 1993, the IAEA Board of Governors supported proposals for strengthening safeguards and increasing the ability of the safeguards system to detect the existence of, and gain access to, undeclared nuclear activities in States with comprehensive Safeguards Agreements. The proposals relate to:

(a) Access to carry out special inspections at any location which the IAEA has reason to believe it needs to visit to obtain additional information relevant to safeguards.

(b) The early provision of design information about new facilities or modifications to existing facilities as soon as the decision is taken to construct or modify the facility. The IAEA's authority to verify design information is a continuing right that extends throughout the facility life cycle.

(c) The reporting of exports, imports and production of nuclear material, as well as exports and imports of certain equipment and non-nuclear material which could be relevant to a weapons programme.

The use of environmental sampling as a tool to help assess the completeness of a State's declaration regarding its nuclear activities has now been implemented in a number of cases (in addition to Iraq) and its applicability in the broader safeguards context is being considered. Additional changes involving, for example, increased intensity of safeguards in countries with more than a significant quantity of HEU and/or plutonium distributed among small facilities are being implemented.
These measures are intended to improve and broaden the scope of the existing safeguards system. This system has worked well in verifying the non-diversion of declared nuclear material at declared nuclear installations. However, the system was not geared to providing assurance that no undeclared nuclear installations existed. Although the safeguards system, as originally designed, provided the legal authority to do this, the Secretariat lacked the information needed to implement this authority. Therefore, it was not timidity but the lack of information about undeclared sites meriting inspection that prevented the discovery of Iraq’s clandestine programme. This situation is being corrected through actions by the Secretariat with the support of Member States.

The discoveries in Iraq highlighted the importance, for effective safeguards, of three types of access: to information, to sites and to the Security Council of the United Nations.

In using inspections as a tool for verification, the first basic requirement is for information regarding locations which might have undeclared nuclear-related items or facilities, requiring inspection. In this context the IAEA gathers much information of its own from its general verification activities, from States themselves, through in-depth analysis of information about nuclear activities obtained from the media and other open literature and, now, through more detailed reporting by States on export of nuclear material and equipment. Additionally, the IAEA now receives information obtained by its Member States through national intelligence means. The IAEA is of the view that no information relevant to safeguards, whatever its provenance, may be ignored, but all information must be critically analysed to determine its credibility.

The second basic requirement is for unlimited right of access for inspectors to locations which the IAEA considers to be relevant to safeguards, even at short notice. Where IAEA access to information and to sites is not forthcoming, access to the Security Council becomes of particular importance. The Relationship Agreement of 14 November 1957 between the United Nations and the IAEA contains provisions allowing for prompt interaction between the United Nations, including the Security Council, and the IAEA. If a State fails to comply with its Safeguards Agreement, the IAEA is obliged to refer the matter to the Security Council, which may decide to take enforcement action to induce a State to accept inspection.

In its statement of 31 January 1992, the Security Council emphasized not only the integral role of fully effective IAEA safeguards in implementing the NPT but also its readiness to take "appropriate measures in the case of any violations notified to them by the IAEA". All this attests to the fact that the Security Council is conscious of the risks inherent in proliferation. It is also sensitive to two specific requirements: that IAEA safeguards must be sufficiently effective to detect any breach or concealment with a high degree of probability; and that the international community needs to be able to continue to trust in the credibility of the safeguards system.
ASSESSMENT OF THE STATE OF CRITICAL TECHNOLOGY AREAS

Nuclear materials: All highly enriched uranium in Iraq under pre-war safeguards has been accounted for. About one third, in the form of fresh reactor fuel, was removed from Iraq in November 1991. The balance, contained in irradiated fuel, was removed in two shipments (December 1993 and February 1994).

About 500 tonnes of natural uranium have been identified and tracked through the Iraqi uranium processing system. Teams are currently reviewing whether the Iraqi declaration of uranium stocks is complete and credible. The IAEA removed six grams of plutonium produced in Iraq (roughly two thirds of which had been produced illegally).

Assessment: Several hundreds of tonnes of natural uranium are under IAEA seal. Minor accountability problems need now to be solved in order to be satisfied that all nuclear materials in Iraq have been accounted for. Ongoing monitoring and verification activities will have to ensure that none of this uranium is used for prohibited activities.

Uranium ore concentration: The ore concentration plant at Al Qaim was completely destroyed during the war.

Assessment: No capability to process uranium ore indigenously now exists in Iraq. The Iraqis have taken no steps to rebuild this plant.

Nuclear material conversion: The nuclear materials feed plant at Al Jazirah was completely destroyed during the war.

Assessment: This key capability is completely destroyed at the production plant level. No backup capability is known or suspected.

Electromagnetic isotope separation (EMIS): This clandestine programme was discovered early in the inspection process. Most of the now known EMIS equipment was damaged in the war. The IAEA inventoried EMIS items, destroyed the remaining pieces, and verified quantities through suppliers. Several facilities that had not been completely destroyed during the war were destroyed by Iraq under IAEA supervision. Dual-use equipment utilized in the manufacture of EMIS components remains under seal. A cruise missile attack on the Al Rabiyah facility in January 1993 did additional damage to equipment which could have been used in the future to reconstitute EMIS. Reconstitution of this programme seems unlikely insofar as it was a large programme that had relied on a blind spot in western intelligence to get as far as it did.
Assessment: The EMIS programme is completely destroyed. It was an indigenous approach to isotope separation that escaped detection. The programme was facing serious difficulties in startup and implementation because of a lack of technical depth among Iraqi technicians. It would have been several years before it could have produced enough uranium for military purposes.

Centrifuge programme: Iraq declared its facilities and much of the centrifuge equipment in July 1991. Two centrifuge prototypes had been tested with some success in test bed experiments. All known centrifuge components and specialized tooling were destroyed in 1991. Other specialized, but dual use, equipment is now under IAEA seal. Suppliers of sensitive equipment and materials as well as sources of technical advice have been identified. Iraq eventually admitted that the Rashdiya facility had a design role in the centrifuge programme. This disclosure came after over a year of pressure from the IAEA.

Assessment: The Iraqi centrifuge programme was in an early stage, using clandestinely obtained European designs and illicitly obtained components to assemble a few prototype research machines. When the Gulf war stopped this programme, Iraqi engineers were still struggling to overcome manufacturing problems they were facing for the industrial scale production of centrifuges. The procurement of hundreds of tonnes of special metals and components, enough to build thousands of machines, was discovered. These materials have been seized and destroyed.

Nuclear reactors and a plutonium programme: The two nuclear reactors at Tuwaitha were totally destroyed by bombing during the war. Suspicions of the existence of an underground reactor have existed since before the war. All plausible information regarding suspected sites has been followed up and has proved to be incorrect.

Assessment: Although suspicions of an underground reactor are vague and seem to be based on rumour, the IAEA is continuing to search for any evidence of an underground reactor and the requisite peripherals such as irradiated fuel reprocessing and nuclear waste handling. No information of any verifiable quality currently exists to support evidence of the existence of a clandestine reactor.

Nuclear weapons design: A programme to assess requirements for a nuclear weapons design existed in Iraq before the war. It consisted of a plan to investigate all the practical elements of designing and building a prototype nuclear weapon. A number of specialized facilities, including buildings for high explosives testing and radioactive materials handling, had been built at Al Atheer to support this programme. These facilities have all been destroyed by Iraq under IAEA supervision. The Tuwaitha nuclear research site was also largely destroyed during the war.

Assessment: The Iraqi nuclear weapons design effort was not very advanced and consisted of a broadly based study of all aspects of producing a uranium core implosion
weapon. Sophisticated concepts for a practical design had not been achieved, as a number of problems remained to be overcome. The hardware and facilities to support this programme have been destroyed, but the concepts remain.

Programme documentation and personnel: One of the early IAEA inspection teams seized over 50,000 pages of documents from the Iraqi Atomic Energy Commission. A substantial fraction of this material consisted of technical progress reports. These were translated and analysed to provide first-hand information of the status of Iraqi activities and achievements in their gaseous diffusion, chemical enrichment, EMIS and weaponization programmes. Correspondence found in this material indicated that other documents had been taken away and hidden by the Iraqi security services just before the team arrived. The Iraqis claim that all programme documentation had been destroyed much earlier. Virtually all of the scientists associated with the nuclear programme remain in Iraq.

Assessment: Iraq's experience could enable it quickly to reconstitute a nuclear weapons programme. The capable scientists remain. How they are currently employed is difficult to ascertain because they have been dispersed. It seems highly probable that a set of documents about the programme remains hidden. The important physical facilities are all destroyed, however, and would have to be rebuilt at great cost, in order to reconstitute the programme.

Appendix 2

STATUS OF FACILITIES AND EQUIPMENT USED IN THE IRAQI NUCLEAR PROGRAMME

Tuwaitha Nuclear Research Centre: Technical headquarters of the programme and site of many R&D functions. This site was devastated and much of the equipment on this site was destroyed during the war. Little additional destruction was necessary.

Al Qaim uranium concentration plant (from Iraqi ores): Destroyed during the war.

Al Jezirah feed materials plant: This plant converted yellow cake to nuclear grade UO₂ and UC₄ for the EMIS programme; it was under consideration also for future production of UF₆. The buildings were destroyed during the war. Most equipment has been accounted for and has been destroyed or is in very bad condition.

Tarmiya EMIS plant: This site was largely destroyed during the war. The electromagnetic separators were damaged by the Iraqis in attempts to conceal them and
their destruction was completed under the supervision of the IAEA. The inspection
teams supervised the demolition of the remaining process building (the beta calutron
building), the utility system and a few other structures so that the site can no longer
be used for its original purpose.

Ash Sharqat EMIS plant: This was a twin of Tarmiya that had not yet been com­
pleted when it was heavily bombed during the war. The IAEA teams requested and
supervised additional demolition along the lines of that carried out at Tarmiya. No
process equipment had been installed at that time at Ash Sharqat.

Al Atheer Materials Centre: This was a partially completed nuclear weapons
development and production site. It was virtually untouched by the war. The eight
specialized process buildings comprising some 350 000 square feet (about
32 500 m²) of laboratory space were blown up by Iraqi demolition teams under
IAEA supervision. A large amount of high quality equipment which had been
installed or stored there was also destroyed.

Al Rabiyah manufacturing plant: This was a plant with large mechanical work­
shops designed and built for the manufacture of large metal components for the Iraqi
EMIS programme. The main function had been to support the EMIS programme.
The plant had high quality, though not unique, machine tool capabilities. The IAEA
inventoried the plant and has been inspecting it regularly. Several pieces of equip­
ment in the plant are monitored under the terms of the ongoing monitoring and verifi­
cation plan. The plant was severely damaged by a cruise missile attack in January
1993 but was totally rebuilt by June 1993.

Dijila electronics plant: This plant supported electronics fabrication activities for
the Iraqi Atomic Energy Commission (IAEC). It has almost no unique pieces of
equipment. The plant was not damaged, and will continue to be monitored by the
IAEA.

A ninth site close to Al Walid was in an early construction phase to house the
centrifuge manufacturing facility of the Al Furat project. This was also the site where
Iraq had planned to establish its first pilot cascade of 100 centrifuges, scheduled to
start operation in mid-1993. This site is now being converted to a conventional
research centre for the Iraqi army and is included in the ongoing monitoring and
verification activities.

A number of other manufacturing workshops were contracted by the IAEC for
the production of components relevant to the weapons programme. All these work­
shops have been identified and are now subject to IAEA monitoring, including the
Engineering Design Centre at Rashdiya, where some of the R&D activities for cen­
trifuges were carried out.
EXPERIENCE OF AN EX (DE FACTO) NUCLEAR WEAPON STATE WITH THE APPLICATION OF POST-IRAQ SAFEGUARDS

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Abstract

EXPERIENCE OF AN EX (DE FACTO) NUCLEAR WEAPON STATE WITH THE APPLICATION OF POST-IRAQ SAFEGUARDS.

Following the discovery of uranium as a by-product of the gold mining industry during the Second World War, South Africa became interested in the peaceful uses of nuclear energy in the late 1960s — amongst these being peaceful nuclear explosions. The character of this part of the nuclear R&D programme changed to that of developing a nuclear deterrent capability over the ensuing two decades as a result of military considerations and resulted in the production of six complete gun assembled devices. Towards the end of the 1980s the national and international political scene changed dramatically for the better, leading to a decision to abandon this programme, dismantle the devices and destroy the non-nuclear parts, accede to the Non-Proliferation Treaty and conclude a comprehensive Safeguards Agreement with the IAEA. All the highly enriched uranium recovered from the devices was declared and placed under safeguards. The conclusion of the Agreement occurred at the height of the Iraq crisis and, given the speculation about South Africa’s nuclear capability, it was not surprising that the IAEA was asked to “verify the completeness of South Africa’s declaration of nuclear material and facilities”. A special team was appointed for this unprecedented task. Owing to South Africa’s new commitment to non-proliferation and its declared policy of transparency the team was allowed to co-opt nuclear weapons experts from outside the IAEA, and access was provided to all records, key personnel and facilities, some of which were non-nuclear on private or military sites. The South African team involved in the verification exercise over a period of more than two years made some observations which the paper presents in the interest of promoting non-proliferation — especially in those States which may wish to follow South Africa’s example of ‘de-proliferation’.

1. INTRODUCTION

This paper seeks to convey the impressions of those on the South African side who were intimately involved with senior IAEA staff for a period of more than two years in establishing the completeness of South Africa’s declaration of nuclear material and facilities after accession to the NPT on 10 July 1991 and the signing
of the Safeguards Agreement on 16 September of that year. Seeing that Resolution 687 on Iraq was adopted in April 1991, that by 16 September the fifth inspection of the Iraqi facilities was already in full swing, and against the background of long standing speculations about South Africa’s nuclear capabilities, the signing of the Safeguards Agreement occurred at a bad time for both the IAEA and South Africa. The IAEA was under immense pressure due to public perceptions of its past performance in Iraq. South Africa, again, was expected to prove its bona fides completely outside the formal terms of the Agreement and in a totally changed safeguards environment. The way in which the IAEA handled this unprecedented situation and the reaction of the international community created some perceptions on the South African side. It is conceded that perceptions may be regarded as a myopic view of history, i.e. without the benefit of perspective brought about, hopefully, by the passage of time. However, the observations may be of use to the IAEA and to those States who may also be contemplating the dismantling of their nuclear capabilities, and are offered in the spirit of promoting non-proliferation.

It is not our intention to comment on the wisdom, or otherwise, of political decisions made by the South African Government of the time. However, in evaluating the reasons for the buildup and eventually the dismantling of South Africa’s nuclear capability, the then prevailing national and international situation must be considered.

2. SOME HISTORICAL EVENTS

South Africa entered the nuclear age in 1944, when the British Government requested the South African Prime Minister to investigate the possible occurrence of uranium deposits in South Africa. It turned out that uranium could be obtained as a natural by-product of the well-established gold mining industry in South Africa. Since then, South Africa has been, and still is, an important producer of uranium ore concentrates. It was then only natural that South Africa would embark on a course to investigate how this natural resource could be exploited to its best advantage.

Nuclear research and development work started in earnest in the late 1950s and early 1960s, the main thrust being on developing an indigenous power reactor and a uranium enrichment process. For financial reasons a choice had to be made in 1967 and this fell on the enrichment project. In the light of the potential of adding value to South Africa’s uranium exports, this was a natural choice. At this time the peaceful uses of nuclear explosives were widely investigated, notably the Ploughshare project in the USA. Seeing that South Africa is an important mining country it was decided to combine the expertise gained in the terminated reactor project with uranium enrichment and to investigate the development and uses of peaceful nuclear explosions. It is significant that it was the Minister of Mining and not the Minister
of Defence who gave approval for the first research and development work on a gun assembled device in 1971.

Following the successful development of an enrichment process the Pilot Enrichment (Y) Plant was designed and built and delivered its first highly enriched uranium (HEU) in January 1978. The non-nuclear part of the first gun assembled device had been completed in 1977 and was fitted with HEU when it first became available.

At this time South Africa became involved in a war in Angola fought by surrogate forces with sophisticated weaponry. As a result of this and the arms embargo the Government felt it prudent to change the character of the peaceful nuclear explosive programme to that of a nuclear deterrent capability along certain strategic guidelines which were adopted in 1978. These guidelines had a very important effect on the number and sophistication of the nuclear devices produced over the years right up to the termination of the programme. The guidelines made provision for three phases:

Phase 1: Strategic uncertainty. The capability is not acknowledged or denied.
Phase 2: Covert acknowledgement to certain international powers.
Phase 3: Overt deterrence by public announcement or an underground demonstration of the capability.

No offensive tactical application was foreseen or intended and in practice the strategy never advanced beyond Phase 1.

The Government made another important decision in September 1985, i.e. to limit the number of gun assembled devices to seven and to allow only limited theoretical and developmental work on more advanced devices. The practical result was that when President F.W. de Klerk assumed office towards the end of 1989 and shortly afterwards took a decision immediately to terminate the programme in the light of momentous national and international changes, only six gun assembled devices had been completed and the nuclear material for a seventh was available. No nuclear tests were ever conducted. A test facility was, however, available had it been necessary to demonstrate the capability.

On instruction by the President, work on the Pilot Enrichment Plant was stopped, the devices dismantled, the hardware destroyed, the HEU recast and removed, and the production facility decontaminated and converted to commercial use before South Africa signed the Safeguards Agreement in September 1991. All the recovered HEU was consequently included in the Initial Report submitted on 30 October 1991. This report also included a list of all the facilities in South Africa which contained nuclear material as of 30 September 1991.

The Safari research reactor (20 MW) was acquired from the USA in the early 1960s and first became critical in March 1965. During its early years it operated under a bilateral agreement which was later replaced by a Trilateral Safeguards Agreement, INFCIRC/98, in September 1967, later modified to INFCIRC/98/Mod.1
in September 1977. The reactor is designed to operate on HEU fuel which was supplied by the USA under contract. In 1976 this contract was unilaterally cancelled. The power level was consequently dropped to 5 MW to conserve the available fuel, and HEU obtained from the Pilot Enrichment Plant was used to manufacture 45% enriched domestic fuel in order to allow the reactor to continue producing essential medical and industrial isotopes. This was publically announced in 1981.

A contract was furthermore concluded with France towards the end of 1976 for the erection of the Koeberg nuclear power station, consisting of two units of 922 MW each. A Trilateral Safeguards Agreement, INFCIRC/244, was signed and entered into force on 5 January 1977, even before construction started on the site.

In spite of these plants being under safeguards it soon became evident that fuel for the Koeberg power plant would also become a political issue — as was the case for Safari — and it was decided to build a second enrichment plant, the Semi-Commercial Enrichment (Z) Plant, to supply the low enriched uranium (LEU) that would be required and also to build a fuel fabrication plant. It was deemed to be more economical to build and operate these plants than to suffer the loss of income by Koeberg and to pay interest on a very expensive, idle plant. Furthermore, the enforced shutdown of Koeberg and the resulting loss of overseas trained personnel could have resulted in the permanent abandonment of the project. The Semi-Commercial Enrichment Plant was commissioned in phases from the middle 1980s and started producing LEU in August 1988. To smooth the partly intermittent early supply of domestic LEU, a certain quantity of LEU had already been imported and has since been used for domestic fuel for Koeberg. Discussions were held on several occasions with the IAEA on the requirements for placing this enrichment plant under safeguards. Agreement could, however, not be reached.

3. **COMPLETENESS**

Shortly after South Africa's accession to the NPT and the entry into force of the Safeguards Agreement the Director General of the IAEA was requested by the General Conference of the IAEA and the General Assembly of the United Nations to "verify the completeness of the inventory of South Africa's nuclear installations and material".

This task was as new to the special Technical Team appointed by the Director General as it was to South Africa and seems to have now created a precedent for countries with extensive nuclear programmes newly acceding to the NPT.

To understand what was meant by 'completeness' recourse may be made to the IAEA Safeguards Glossary:
Reconciliation of accounting with operating records. An IAEA inspection activity with the purpose of identifying and clarifying any inconsistency between the accounting and operating data in order to determine the completeness, correctness and consistency of the data.

South Africa had previously offered to submit the historical operating records of the Pilot Enrichment Plant to the IAEA for verification. However, it soon became evident to both sides that the Technical Team could only succeed in its task if South Africa provided all available historical operating and accounting records relating to both the HEU and LEU cycles.

Although this was completely outside the legal requirements of the Safeguards Agreement, South Africa decided to submit these records on a completely voluntary basis in the interests of its new commitment to transparency and non-proliferation. This was done in two phases:

— Before the announcement by the State President on 24 March 1993 of South Africa’s terminated nuclear deterrent programme, all records relating to the HEU and LEU cycles were submitted and all historical imports and exports declared. The Technical Team compared this with the inventory declared in the Initial Report, which included all the HEU recovered from the dismantled nuclear devices. The result of the Team’s investigations was a report of the Director General to the General Conference in September 1992.

— After the announcement of the already dismantled nuclear deterrent programme, information on this programme was provided in great detail to the Team. This included details of mainly theoretical work done over the years on more advanced concepts and the status of these projects at the termination of the programme. The Team reported its findings to the Director General and the General Conference in September 1993.

To assist the Team further in its unprecedented task, a policy of visits ‘any time, any place — within reason’ was followed with the IAEA. The team was therefore provided with full and free access to key persons previously involved with the terminated programme as well as to facilities previously used for the programme. Some of these facilities had been abandoned for more than a decade and included nuclear and non-nuclear facilities in the nuclear and armament industries, as well as military and private sites. Facilities were identified by South Africa — even facilities on the very fringes of the programme — and access to others was requested by the Technical Team on the basis of information supplied by a Member State (or States). In line with its policy of transparency South Africa furthermore allowed the Team to co-opt nuclear weapons experts from outside the IAEA to provide the required expertise.
4. LESSONS LEARNED

4.1. A shot in the foot?

The safeguards system is an international political solution to the non-proliferation problem based on the premise that the misuse of nuclear materials can be technically monitored by an independent, competent organization. However, the same international political community showed that formal supply agreements concerning the peaceful application of nuclear energy and materials conducted under international safeguards could be unilaterally abrogated — thereby implying that the system it had created could not be trusted.

4.2. Embargoes and boycotts are counterproductive

If a State considers its national security at risk — rightly or wrongly — and if this is further compounded by the suspension of all nuclear co-operation and an embargo on the supply of conventional armaments, whether it is morally defensible or not, the practical effect will be that a State with the necessary resolve and means will create or obtain whatever it deems necessary to fill the void. In South Africa’s case this led to the creation of a nuclear deterrent capability and a sophisticated conventional weapons industry which is today one of the most important export industries in the country.

4.3. Quantitative and qualitative conclusions

Initially there had been much speculation about the ‘apparent discrepancy’ found by the IAEA in South Africa. This did not refer to missing nuclear material of which records existed. It was a theoretical conclusion based on the $^{235}\text{U}/\text{U}$ ratio of the uranium declared in the Initial Report.

The records made available to the IAEA reflected the operation over many years of two plants with unique operating characteristics — operated for strategic reasons and not under strict safeguards discipline, with large uncertainties in the $^{235}\text{U}$ content of depleted material and in tens of thousands of waste drums. In spite of this, the team persisted in trying to verify its theoretical assumptions.

Rather than concentrating on theoretical quantitative matters only, the IAEA should be bold enough to also make value judgements of qualitative aspects such as openness, co-operation, the provision of access and information, i.e. of the transparency of the State. In South Africa’s case the abrupt termination of the programme was very obvious and the Technical Team had seen many examples of this. This obviously signified a dramatic change of direction which brought an entirely different dimension to the fore, i.e. one of trust. The IAEA’s approach can be compared to a boarding party on that famous abandoned ship, the Marie Celeste, comparing
the number of table place settings with the passenger list and concluding that there is a discrepancy — totally overlooking the fact that something drastic has happened to the ship.

A situation of mutual trust should and can be built up in a spirit of complete openness and co-operation by both sides. It is very difficult for outsiders to evaluate the process and assess the level of trust achieved. If it is then expected of the IAEA to make value judgements of non-measurable quantities such as trust, co-operation, openness, the international community should equally be expected to display its faith in the organization it created and sustains as a technically competent, specialized group of people who are in a better position to judge than the many ill-informed critics on the sidelines. The international community should visibly support an impartial and independent IAEA — South Africa will surely do its part.

4.4. Two-way transparency

We in South Africa have discovered that you cannot walk the path we have chosen without complete transparency. But transparency should not be a one-way mirror, transparent in one direction only. The actions of the IAEA should be equally transparent to the State. In this regard, and accepting that is was necessary to strengthen safeguards after Iraq, we feel that the IAEA should be especially careful in the way it uses information supplied by Member States, as there is a great danger of manipulation by those supplying the information, probably according to their own agendas. Any suspicion of such manipulation will forever deter those States who may contemplate following the path chosen by South Africa — to the detriment of non-proliferation.

4.5. Seeing and not believing?

States contemplating the dismantling of their nuclear capabilities will equally be deterred by the blunt non-acceptance of their bona fides by the IAEA and the international community. Safeguards Agreements require that plant records be kept for five years only. However, South Africa was in the fortunate position of being able to supply the IAEA with historical records going back more than 15 years. The reaction of the international community was that these records were probably fakes, which were specially falsified. They came to this conclusion without the benefit the IAEA Team had in examining the consistency of various types of original records, interviewing plant personnel, visiting a variety of facilities, etc. However, in spite of this advantage the IAEA succumbed to pressure and requested South Africa to submit some of these records for forensic age testing by a Member State. A genuine lack of records would, on the other hand, most probably have led to an accusation of the deliberate destruction of these!
4.6. Confidentiality

Whereas South Africa has no reason to question the integrity of the Technical Team appointed by the Director General, the question of the confidentiality of information as it diffuses further through the IAEA should remain an item of the highest priority. It is interesting to note that the confidentiality of information plays a much more prominent role in the Chemical Weapons Convention than foreseen by the NPT or Safeguards Agreements. Confidentiality is a very important issue because there will be many attempts of parallel verification by States and the media, who without the access to records, facilities and personnel that the IAEA has, can come to wrong conclusions, resulting in counterproductive pressures.

4.7. Access anywhere, any time?

To enable an IAEA Team involved in a completeness verification exercise to gain sufficient confidence in the bona fides of the State, it is inevitable that it be provided with full and free access during the verification process. The obvious question arises; how to maintain the level of confidence achieved without compromising commercial and industrial confidentiality once the verification process has been completed, especially since Safeguards Agreements do not provide for access to non-nuclear plants. A balance could probably be obtained by the State voluntarily and formally allowing additional access, but under controlled conditions such as the concept of managed access foreseen in the Chemical Weapons Convention.

4.8. Advice

A final word of advice to States with extensive nuclear programmes who may contemplate accession to the NPT: This is not a step to be taken half-heartedly or for dubious political reasons. It requires full commitment to the ideals of the NPT and a good measure of patience and determination to surmount the many obstacles that will be encountered on the way of trying to ‘prove’ completeness.
VERIFICATION OF COMPLETENESS AND CORRECTNESS OF INVENTORY

Experience gained in the verification of the completeness of the inventory of South Africa’s nuclear installations and material

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Abstract

VERIFICATION OF COMPLETENESS AND CORRECTNESS OF INVENTORY: EXPERIENCE GAINED IN THE VERIFICATION OF THE COMPLETENESS OF THE INVENTORY OF SOUTH AFRICA’S NUCLEAR INSTALLATIONS AND MATERIAL.

South Africa’s accession to the NPT on 10 July 1991 was promptly followed by the signing of a comprehensive Safeguards Agreement, INFCIRC/394, with the IAEA on 16 September 1991. Four days later, the General Conference of the IAEA adopted a resolution, GC(XXXV)/RES/567, requesting the Director General to ensure early implementation of the Safeguards Agreement and to verify the completeness of the inventory of South Africa’s nuclear installations and material. The activities carried out to verify the correctness of the inventory of nuclear material, included in the initial report, extended over several months and involved long established measures such as the examination of contemporary operating and accounting records, and destructive and non-destructive analysis of the nature and quantity of individual items and batches. The assessment of the completeness of the inventory of South Africa’s nuclear installations and material was carried out as a separate exercise by a team of senior members of the IAEA Department of Safeguards specifically appointed for the purpose by the Director General. South Africa’s extensive nuclear fuel cycle made the task of the assessment of completeness complex, requiring considerable inspection resources and extensive co-operation from the State authorities regarding the provision of access to defunct facilities and historical operating records. The task was further complicated when, on 24 March 1993, State President de Klerk announced; in a broadcast speech to the Parliament, that South Africa had developed and subsequently dismantled a “limited nuclear deterrent capability” involving the design and manufacture of seven gun-assembled (HEU) devices. An augmented IAEA team, composed of the personnel assigned to carry out the assessment of completeness, and, among other specialists, nuclear weapons experts were assigned to assess the status of the former nuclear weapons programme and to ascertain that all of the nuclear material involved in the programme had been recovered and had been placed under safeguards.
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In accordance with Article 62 of the Safeguards Agreement, South Africa provided an initial report, dated 22 October 1991, on all nuclear materials subject to safeguards as of 30 September 1991. The first inspections carried out by the IAEA in accordance with the comprehensive Safeguards Agreement commenced in November 1991.

2. VERIFICATION OF CORRECTNESS OF INVENTORY

The activities carried out to verify the correctness of the inventory of nuclear material, included in the initial report, extended over several months. The initial report is a comprehensive document and includes quantitative data on all types of nuclear material, on a facility by facility basis, expanded by attachments which provide detail on the location and the number of items of nuclear material contained in each respective facility. It was therefore possible, on the basis of the data contained in the initial report and subsequent inventory changes, to establish an itemized list of the nuclear material inventory of each facility. The verification of such itemized lists was carried out during the first few months of the implementation of the comprehensive Safeguards Agreement, in accordance with the requirements for physical inventory verification (PIV) specified in the IAEA 1991–1995 Safeguards Criteria, using established accountancy verification measures.

Unlike other Member States which had entered into comprehensive safeguards agreements, in recent years South Africa had a number of nuclear facilities, of indigenous origin, which had not been subject to safeguards transfer agreements (INFCIRC/66) and were thus relatively unknown to the IAEA (see Table I).

This situation made it necessary to devote considerable effort to the understanding of the facility processes in order to establish workable safeguards approaches for interim implementation during the period in which the facility attachment would be negotiated. This process was facilitated through a joint seminar, which provided the opportunity for the IAEA to explain the accountancy procedures appertaining to an INFCIRC/153 type agreement and for the South African State System of Accounting and Control and facility operators to provide an insight into the facilities and their operating procedures.
**TABLE I. SOUTH AFRICA'S NUCLEAR INSTALLATIONS**

<table>
<thead>
<tr>
<th>Installations formerly inspected in accordance with Safeguards Transfer Agreements</th>
</tr>
</thead>
<tbody>
<tr>
<td>SAFARI-1 research reactor, Atomic Energy Corporation, Pelindaba</td>
</tr>
<tr>
<td>Hot cell complex, Atomic Energy Corporation, Pelindaba</td>
</tr>
<tr>
<td>Koebberg power reactor units 1 and 2, Electricity Supply Commission</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Installations inspected in accordance with the comprehensive Safeguards Agreement</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uranium conversion, UF₆ production plant</td>
</tr>
<tr>
<td>Pilot HEU enrichment plant (Y-plant), now defunct</td>
</tr>
<tr>
<td>HEU storage facility</td>
</tr>
<tr>
<td>HEU-UF₆ and metal/alloy production plant</td>
</tr>
<tr>
<td>HEU fuel fabrication plant</td>
</tr>
<tr>
<td>Semi-commercial LEU enrichment plant (Z-plant)</td>
</tr>
<tr>
<td>MLIS laser enrichment R&amp;D facility</td>
</tr>
<tr>
<td>LEU fuel fabrication plant</td>
</tr>
<tr>
<td>Natural uranium/depleted uranium metal plants</td>
</tr>
<tr>
<td>Decontamination plants</td>
</tr>
<tr>
<td>Waste storage compound</td>
</tr>
<tr>
<td>Locations outside facilities</td>
</tr>
</tbody>
</table>

From the time of the 'initial inspections' in late 1991, verification activities carried out in South Africa have been based on the IAEA 1991–1995 Safeguards Criteria. In October 1992 a near-simultaneous PIV, involving all South African facilities, was successfully carried out and all quantity goals were attained during that first material balance period. A similar exercise was carried out in August 1993.

As might be expected, it proved necessary to make a number of corrections to the data included in the initial report, resulting from the continuing efforts of the SSAC to ensure the accuracy of the data, errors identified during the inspection process and corrections to estimations resulting from measurements made by the facility operators after the issue of the initial report. This latter aspect was particularly relevant in the case of material recovered as a result of decontamination of plant components.

At present, facility attachments are in force for six facilities and it is intended to complete negotiations for the remaining facilities during 1994.
3. ASSESSMENT OF COMPLETENESS OF INVENTORY OF NUCLEAR INSTALLATIONS AND MATERIAL

The assessment of the completeness of the inventory of South Africa’s nuclear installations and material was carried out as a separate exercise by a team of senior members of the Department of Safeguards specifically appointed for the purpose by the Director General.

South Africa’s extensive nuclear fuel cycle made the task of the assessment of completeness complex, requiring considerable inspection resources and extensive co-operation from the State authorities regarding the provision of access to defunct facilities and historical operating records.

A practical basis was established through which the mutual consistency of the inventory of nuclear installations and material and hence their completeness could be determined. The declared inventory was first evaluated with respect to production, imports and usage, and then the isotopic balance of the inventory was calculated and compared with its natural uranium origin.

Through this process, the declared inventory was found to be consistent with the declared production and usage data, but the calculated isotopic balance indicated ‘apparent discrepancies’ with respect to the HEU produced by the defunct pilot enrichment (Y) plant and with respect to the LEU produced by the semi-commercial enrichment (Z) plant. The direction of these apparent discrepancies could be interpreted to indicate that an amount of uranium-235 was unaccounted for.

Having regard to the period of time involved (for the Y-plant in particular) and the absence of accurate accountancy of the depleted uranium waste stream, such ‘apparent discrepancies’ were not unexpected. It was, however, considered necessary to continue efforts to further clarify them, with priority being given to the Y-plant data. Further examination of records, particularly those covering the recovery of HEU material following the shutdown of the plant, resulted in a significant reduction of the magnitude of the apparent discrepancy.

To supplement these activities, an exhaustive examination of the performance of the Y-plant over its entire operating history was undertaken. This examination involved the analysis of data from several thousand operating records, which detailed the plant status, on a daily basis, in terms of availability of separating modules, the feed rate and the assay of feed, product and tail streams. Technical documents describing phenomena which had affected the plant performance were also studied.

The results of this examination showed that the amount of HEU declared to have been produced by the Y-plant was consistent with the plant’s production capacity.

On the basis of these studies the IAEA determined that it was reasonable to conclude that the uranium-235 balance of the HEU, LEU and depleted uranium produced by the Y-plant was consistent with the natural uranium feed and that the
amounts of HEU that could have been produced by the plant were consistent with
the amounts declared in the initial report.

Work continues on the assessment of the apparent discrepancy associated
with the LEU produced by the semi-commercial enrichment plant.

4. SOUTH AFRICA'S NUCLEAR WEAPONS PROGRAMME

The inventory of HEU declared by South Africa in its initial report was sub­
stantial and the IAEA recognized that this material could have been taken to indicate
that a significant component of the HEU inventory had been recovered from an aban­
doned nuclear weapons programme or, less likely, had been accumulated to supply
a planned nuclear weapons programme which had been abandoned prior to its
implementation.

South Africa had no obligation to declare what had been the past purpose of
this material. Equally, the primary task of the IAEA was to ascertain that all nuclear
material had been declared and placed under safeguards; priority was given to this

In addition to the established accountancy verification measures, the IAEA,
acting on information received from Member States, carried out inspections, includ­
ing the taking of environmental samples, at a location that was later declared to be
an unused nuclear weapons test facility in the Kalahari desert and at a number of
abandoned buildings (including a general purpose critical facility) located just out­
side the Pelindaba security fence. The South African authorities were very co­
operative in facilitating access to these locations but claimed lack of detailed
knowledge of their past use.

It is now a matter of record that, on 24 March 1993, President de Klerk, in
a broadcast speech to the South African Parliament, made a declaration that "at one
stage, South Africa did, indeed, develop a limited nuclear deterrent capability".

President de Klerk’s statement included a description of the scope and objec­
tive of the ‘capability’ and the rationale for its abandonment and South Africa’s
accession to the NPT.

It is significant to note that on the day of the declaration, two members of the
IAEA team were present at the Atomic Energy Corporation (AEC), Pelindaba,
carrying out follow-up actions directed towards the clarification of the apparent
discrepancy in the isotopic balance of the HEU material produced by the Y-plant.
On the day after the declaration, these two team members made a preliminary
visit to a number of key facilities concerned with the abandoned nuclear weapons
programme.

In the following five month period, the team, augmented by nuclear weapons
experts, carried out inspections at a number of facilities and locations that had been
<table>
<thead>
<tr>
<th>Year</th>
<th>Event</th>
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<tbody>
<tr>
<td>1970</td>
<td>Uranium enrichment project announced</td>
</tr>
<tr>
<td>1971</td>
<td>Approval for R&amp;D based on gun-assembled device relating to nuclear explosions for peaceful purposes</td>
</tr>
<tr>
<td>1973</td>
<td>Investigation into separation of lithium isotopes</td>
</tr>
<tr>
<td>1974</td>
<td>Prime Minister approves limited programme for development of nuclear weapons as deterrent. First stage of pilot enrichment plant commissioned. Approval for test site development in the Kalahari desert.</td>
</tr>
<tr>
<td>1975</td>
<td>Work on the Kalahari test shafts commenced</td>
</tr>
<tr>
<td>1976</td>
<td>Export from the USA of fuel for the SAFARI-1 research reactor stopped</td>
</tr>
<tr>
<td>1977</td>
<td>Kalahari test site abandoned. Full cascaded operation of the pilot enrichment plant</td>
</tr>
<tr>
<td>1978</td>
<td>First HEU product withdrawn from the pilot enrichment plant</td>
</tr>
<tr>
<td>1979</td>
<td>First nuclear device completed by the AEC. Decision that ARMS COR should take over programme and produce all further devices</td>
</tr>
<tr>
<td>1980</td>
<td>Construction of tritium handling laboratory completed</td>
</tr>
<tr>
<td>1981</td>
<td>ARMS COR/Circle facilities completed. Approval of the Gouriqua programme for commercial PWR technology development, as well as possible future tritium and plutonium production</td>
</tr>
<tr>
<td>1982</td>
<td>Second device completed</td>
</tr>
<tr>
<td>1985</td>
<td>Government decision to limit number and type of devices to seven gun-assembled devices, to further develop implosion technology and to study more advanced concepts. Lithium-6 Avlis programme redirected towards lithium-7 production for water chemistry control in commercial power reactors</td>
</tr>
<tr>
<td>1987</td>
<td>Commercial programme for tritium radioluminescent light sources started</td>
</tr>
<tr>
<td>1987–1989</td>
<td>Completion of four additional devices</td>
</tr>
<tr>
<td>1989–1991</td>
<td>Construction of facilities at ARMS COR/Advena central laboratories</td>
</tr>
<tr>
<td>1989</td>
<td>Decision to terminate nuclear weapons programme (November). Gouriqua programme stopped</td>
</tr>
<tr>
<td>Year</td>
<td>Event Description</td>
</tr>
<tr>
<td>------</td>
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</tr>
</tbody>
</table>
| 1990 | Pilot enrichment plant ceased operation (February)  
Order by State President for destruction of the six completed nuclear devices and the incomplete seventh device (26 February) |
| 1991 | Accession to the NPT (10 July)  
All HEU returned from ARMSCOR/Circle to the AEC (14 March to 6 September)  
Signature and entry into force of the Safeguards Agreement (16 September)  
Initial report submitted (30 October)  
Start of ad hoc inspections (November) |
| 1993 | Destruction of documentation relating to nuclear weapons programme  
ordered by State President on 17 March; destruction completed on 23 March  
State President’s announcement in Parliament of the existence and subsequent abandonment of the former nuclear weapons programme (24 March)  
Preliminary visit by IAEA team members to the ARMSCOR/Circle facilities (25 March)  
Visits of the IAEA team to assess the status of the former nuclear weapons programme (22 April to 4 May, 3-11 June and 9-13 August) |

declared to have been involved in the former nuclear weapons programme, with the objectives as summarized below:

- To gain assurance that all nuclear material used in the nuclear weapons programme had been returned to peaceful usage and had been placed under IAEA safeguards.
- To assess that all non-nuclear weapons specific components of the devices had been destroyed; that all laboratory and engineering facilities involved in the programme had been fully decommissioned and abandoned or converted to commercial non-nuclear usage or peaceful nuclear usage; that all weapons specific equipment had been destroyed and that all other equipment had been converted to commercial non-nuclear usage or peaceful nuclear usage.
- To obtain information regarding the dismantling programme, the destruction of design and manufacturing information, including drawings, and the philosophy followed in the destruction of the nuclear weapons.
- To assess the completeness and correctness of the information provided by South Africa with respect to the timing and scope of the nuclear weapons programme, and the development, manufacture and subsequent dismantling of the nuclear weapons.
— To consult on the arrangements for, and ultimately to witness, the rendering useless of the Kalahari test shafts.
— To visit facilities previously involved in or associated with the nuclear weapons programme and to confirm that they are no longer being used for such purposes.
— To consult on future strategies for maintaining assurance that the nuclear weapons capability would not be regenerated.

These objectives were based on the IAEA's rights and obligations under the Safeguards Agreement and on the stated policy of the South African Government for full transparency with respect to the country's former nuclear weapons programme. The IAEA team had extensive discussions with the South African authorities and technical staff at the AEC and at the State owned armaments corporation (ARMSCOR) who had been responsible for the production phase of the nuclear weapons programme. Detailed briefings were provided on the various phases of the programme and on the associated development and production facilities. Information on the future development of the programme, which had been envisaged before the order to dismantle the nuclear weapons programme intervened, was also provided.

Information from Member States was used to confirm that all relevant facilities/locations had been inspected.

On the basis of official documents, programme records and information obtained through interviews with principal personnel at the various facilities and locations involved in the programme, the IAEA team was able to document the timing and scope of the nuclear weapons programme; a chronology of the main events is given in Table II.

In the ten year period up to 1979, all research and development work on nuclear explosive devices was done by the South African Atomic Energy Board and its successor, the AEC. This work resulted in the production of a 'non-deliverable demonstration device', which was designed in such a way that, if the need arose, it could be rapidly deployed for an underground test to demonstrate South Africa's nuclear weapons capability. Its purpose remained that of a demonstration device throughout the programme; it was never converted to a deliverable device.

It was in 1979 that the responsibility for the nuclear weapons programme was transferred to ARMSCOR, while the AEC was made responsible for the production and supply of HEU and for theoretical studies and development work in nuclear weapons technology. The headquarters of ARMSCOR's nuclear weapons activities were the so-called Circle facilities, located some 15 km away from the AEC's establishment at Pelindaba. The Circle facilities were constructed during 1980 on the basis of designs provided by the AEC and were commissioned in May 1981. The nuclear weapons programme thus established involved:

— The development and production of a number of deliverable gun-assembled devices;
— Lithium-6 separation for the production of tritium for possible future use in boosted devices;
— Studies of implosion and thermonuclear technology;
— Research and development for the production and recovery of plutonium and tritium.

In September 1985 the South African Government decided to limit the scope of the programme to the production of seven gun-assembled devices, to stop all work related to possible plutonium devices and to limit the production of lithium-6, but to allow further development work on implosion technology and theoretical work on more advanced devices.

The first prototype deliverable device had been completed in December 1982, but it was not until August 1987 that the first qualified production model was completed. The delay was largely due to the implementation of a rigorous engineering qualification programme directed towards safety and security under a range of postulated storage, delivery and accident scenarios. When, in November 1989, the decision was taken by the Government to stop the production of nuclear weapons, four further qualified deliverable gun-assembled devices had been completed and the HEU core and some non-nuclear components for a seventh device had been fabricated. On 26 February 1990 the State President issued a written instruction that, inter alia, all existing nuclear devices were to be dismantled and the nuclear materials were to be melted down and returned to the AEC in preparation for South Africa’s accession to the NPT.

By the time of the IAEA team’s visit in April 1993, the dismantling and destruction of weapons components and the destruction of the technical documentation had been nearly completed. Dismantling records concerning the HEU components of the weapons were available and provided sufficient detail to enable the ARMSCOR data to be correlated with the corresponding data in the nuclear material accountancy records maintained by the AEC.

The dismantling of the non-nuclear components of the weapons had been carried out in accordance with procedures approved by the South African authorities. A number of destroyed or partially destroyed components had been retained and were shown to some members of the team in April 1993. Remaining records, in the form of ‘build history’ logbooks for the completed weapons and the experimental devices, were examined and compared with the dismantling listings. Identification numbers of remaining components were compared and found to be consistent with those shown in the records.

The team carried out an audit of the records of the transfer of weapons-grade HEU between the AEC and ARMSCOR/Circle.

As a result of this audit, the team concluded that the HEU originally supplied to ARMSCOR/Circle had been returned to the AEC and was subject to IAEA safeguards at the time of entry into force of the Safeguards Agreement.
The team visited all facilities identified as having connection with the former nuclear weapons programme. It is appropriate to record the active co-operation of the South African authorities in arranging for access to all facilities that the team requested to visit — both those facilities which had been provisionally listed by the South African authorities as having direct connection with the former nuclear weapons programme, or with peripheral activities, and additional facilities identified by the team. The IAEA is not in possession of any information suggesting the existence of any undeclared facilities.

Actions were taken by ARMSCOR to render useless the test shafts at the Vas-trap (Kalahari) site, in accordance with a plan incorporating specific suggestions made by the IAEA team. Although the implementation of this plan met with some initial practical difficulties, the measures to render the test shafts useless were successfully completed in July 1993 and were witnessed by IAEA safeguards inspectors.

The equipment used for uranium metallurgy at ARMSCOR/Circle had been returned to the AEC at the end of the programme. The whole uranium metallurgy process area at ARMSCOR/Circle had been dismantled and decontaminated. The machine tools used for manufacturing the HEU and high explosives components had been decontaminated and are now available for commercial non-nuclear applications. The South African authorities stated that specialized equipment supporting the weapons systems in the form of computerized testing equipment has been rendered useless through the destruction of the specific software.

5. GENERAL CONCLUSIONS

It was determined that the magnitude of the apparent discrepancy in the uranium-235 balance associated with the Y-plant was such that, having regard to the normal uncertainties expected to be involved in the plant historical operating and accounting records, it was reasonable to conclude that the uranium-235 balance of the HEU, LEU and depleted uranium produced by the pilot enrichment plant was consistent with the uranium feed. Assessment of the production capacity of the pilot enrichment plant, on the basis of operating records and supporting technical data provided to the IAEA team by the AEC, indicated that it was reasonable to conclude that the amounts of HEU which could have been produced by the plant were consistent with the amounts declared in the initial report.

The IAEA team’s audit of the associated records indicated that all of the HEU provided by the AEC to the nuclear weapons programme had been returned to the AEC and was subject to IAEA safeguards at the time of entry into force of the Safeguards Agreement.

The findings from the team’s examination of records, facilities and remaining non-nuclear components of the dismantled/destroyed nuclear weapons and from the
team's evaluation of the amount of HEU produced by the pilot enrichment plant showed consistency with the declared scope of the nuclear weapons programme.

The team found no indication to suggest that there remained any sensitive components of the nuclear weapons programme which had not been either rendered useless or converted to commercial non-nuclear applications or peaceful nuclear usage.

These general conclusions had strong technical bases and were significantly supported by the transparency and openness of the South African authorities with respect to access to information and locations, in particular the stated and demonstrated willingness of the authorities to facilitate access to any location that the IAEA may identify.

The IAEA's assessment of the completeness of South Africa's inventory of nuclear facilities and materials and its assessment of the status of the former nuclear weapons programme is not free from uncertainty. As in all cases where a large nuclear programme comes under safeguards, there can be no absolute assurance that all the nuclear material, subject to safeguards under the Agreement, has been declared to the IAEA and that all nuclear facilities have similarly been declared.

In the case of South Africa, the results of extensive inspection and assessment, and the transparency and openness shown leave relatively little room for speculation to the contrary. However, in the future, and without prejudice to the IAEA's rights under the Safeguards Agreement, the IAEA plans to take up the standing invitation of the South African Government — under its reiterated policy of transparency — to provide the IAEA with full access to any location or facility associated with the former nuclear weapons programme and to grant access, on a case by case basis, to other locations or facilities that the IAEA may specifically wish to visit.

**BIBLIOGRAPHY**


Reporting Scheme Endorsed by the IAEA Board of Governors

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Abstract

Reporting Scheme Endorsed by the IAEA Board of Governors.

As part of a series of proposals aimed at strengthening safeguards, the proposals of the IAEA Secretariat for the Universal Reporting System were considered at the meeting of the Board of Governors in June 1992. Pending further consideration by the Board of the terms of such a system, it was understood that all States willing to provide, on a voluntary basis, the required information would do so as they deemed appropriate. The Secretariat circulated to all Member States forms that could be used by them to provide information about exports, imports and inventories of nuclear material, and about exports and imports of specified equipment and non-nuclear material, in addition to the information required under existing Safeguards Agreements. Twenty-six Member States and the Commission of the European Communities (CEC) responded to the Secretariat’s invitation, indicating their willingness to provide either all or part of the information requested, in some cases subject to certain conditions. A revised Secretariat paper on reporting on exports, imports and production of nuclear material, and on exports and imports of specified equipment and non-nuclear material, was considered by the Board at its meeting in February 1993. The Board endorsed a reporting scheme, noting that participation by Member States would be voluntary in nature and hoping that subscription to it would attain universality. The Secretariat notified all Member States about the Board’s decision and circulated forms on which they could report the relevant information. To date, 27 States and the CEC have submitted reports, pursuant either to the invitation in 1992 or to the scheme endorsed by the Board in 1993. In addition, 13 States replied, but have not yet sent reports.

1. Background

Provision of comprehensive information to the IAEA on the worldwide production and transfer of all nuclear material used in peaceful activities would contribute to increased confidence that such material was being used only for peaceful purposes. At present, the IAEA correlates the information received on exports and imports of nuclear material with the measures taken by it to verify non-diversion of nuclear material. This correlation is not comprehensive because the data on imports and exports received by the IAEA do not include all nuclear material (e.g. uranium...
ore concentrates are not always subject to such reporting) or all exports (many exports for non-nuclear purposes are excluded) (see Fig. 1). Regarding production, the IAEA receives only limited information on nuclear material that is not yet of a composition and purity suitable for fuel fabrication or isotopic enrichment (i.e. material that is still in the form of uranium ore concentrates).
A more comprehensive reporting scheme on exports and imports of certain equipment and non-nuclear material commonly used in the nuclear industry would also provide a measure of transparency regarding nuclear activities in all States and would contribute to confidence in the peaceful use of such equipment and material. Currently, the IAEA receives only information on the transfer of such equipment and material that are subject to INFCIRC/66/Rev.2 type Safeguards Agreements in the importing State (see Fig. 2).
2. HISTORY OF DISCUSSION ON VOLUNTARY REPORTING

2.1. February 1992 Board meeting

As part of a series of proposals aimed at strengthening safeguards, the Board of Governors considered two proposals by the Secretariat related to the reporting and verification of the export, import and production of nuclear material, and of the export, import and production of certain equipment and non-nuclear material for peaceful purposes. Many useful comments were made at the February 1992 Board meeting and it was agreed that the Secretariat would have further discussions with groups of Member States and would then revise these proposals for resubmission to the Board in June 1992.

2.2. June 1992 Board meeting

Following consultation with Member States the Secretariat revised the papers presented at the February meeting, and submitted two papers to the June Board. Many Board members at this meeting were in favour of a voluntary system for reporting exports and imports of nuclear material, and for reporting exports of relevant equipment and non-nuclear material. Different views were expressed with regard to reporting on inventories of nuclear material and on imports of such equipment and non-nuclear material, and with regard to any additional right of inspection. The Secretariat was requested to examine those issues further, in consultation with Member States, and to report to the Board, as appropriate. It was also agreed that all States willing to do so would in the meantime provide the IAEA, on a voluntary basis, with information referred to in the two papers, as they felt appropriate.

On 9 July 1992, the Director General circulated a letter to all Member States concerning universal reporting of nuclear material and of certain equipment and non-nuclear material. Twenty-six Member States (including the twelve Euratom States) and the CEC responded to the Secretariat's invitation, indicating their willingness to provide either all or part of the information requested, in some cases subject to certain conditions.

2.3. February 1993 Board meeting

After consultations with Member States, the Secretariat prepared a specific proposal and submitted it to the Board for consideration in February 1993. Upon completion of its deliberations, the Board, noting that participation by Member States in the reporting scheme would be voluntary in nature and hoping that subscription to it would attain universality, decided:
To endorse a proposal for the establishment of a reporting scheme for nuclear material and specified equipment and non-nuclear material as a means of strengthening the IAEA safeguards system;

(b) To encourage Member States to participate in the scheme, providing the IAEA with the relevant information relating to their exports and imports of nuclear material and the exports of specified equipment and non-nuclear material;

(c) To invite States that were willing to do so, to provide information on their production of nuclear material and on their imports of specified equipment and non-nuclear material;

(d) To request the Secretariat to enter into appropriate arrangements through an exchange of letters with States willing to participate in the reporting scheme, and to report to the Board periodically on developments in that regard.

3. BENEFIT OF REPORTING INFORMATION

The Secretariat has identified certain categories of information which would help to strengthen IAEA safeguards. These are discussed below.

3.1. Information on exports of nuclear material

The provision of such information would enable the IAEA to compare the details given by exporting States with the proposed usage declared by importing States. Most supplier States already have a mechanism in place for regulating exports of nuclear material that is not subject to the reporting requirements under IAEA Safeguards Agreements.

3.2. Information on imports of nuclear material

This information would provide a more complete picture of nuclear material in States with nuclear programmes. The provision of such information by importing States would also enable transfers of nuclear material to be cross-checked with information given by exporting States. However, while most States with nuclear programmes have mechanisms in place to regulate the import of nuclear material, States with very small programmes and/or States needing nuclear material for non-nuclear purposes may not currently have such mechanisms and would therefore find it difficult to report on imports. This difficulty could be alleviated to some extent if the exporting State were to notify simultaneously the IAEA and the Government of the importing State of the export of nuclear material.
3.3. **Information on production of nuclear material**

This information would give the IAEA a more complete picture of the nuclear material in the State concerned. Certain elements of this kind of information are already available to other organizations but are not transmitted to the IAEA.

3.4. **Information on exports of specified equipment and non-nuclear material specially designed or prepared for nuclear uses**

Most suppliers have accepted commitments to regulate the export of such items. Notifying the IAEA of exports of these items would add an additional measure of transparency in relation to nuclear activities in all States. As a practical measure and for reasons of convenience only, it was decided to utilize the list presented in document INFCIRC/254/Rev.1/Part 1 when reporting on exports (and imports, if applicable) of specified equipment.

3.5. **Information on imports of specified equipment and non-nuclear material specially designed or prepared for nuclear uses**

This information would enable cross-checks to be made against notifications of exports, thereby focusing the attention of the IAEA on relatively few remaining cases. In cases where States do not have any regulatory mechanisms for controlling imports of such items, it would be helpful if the relevant authorities of the exporting State were to notify the authorities of the importing State of the transfer(s) in question.

4. **FOLLOW-UP ACTIONS BY THE SECRETARIAT**

4.1. **Reporting arrangement**

Following the endorsement by the Board of Governors of the scheme for reporting, the Director General sent a letter to all States informing them of the Board's decision on 24 May 1993. It was expected that the States which had responded favourably to the Director General's earlier letter of July 1992, inviting them to provide additional information on exports and imports, on a voluntary basis, would also respond favourably to the endorsed reporting scheme. The guidance on the modalities of reporting had also been sent to all Member States by the DDG-SG on 4 June 1993. Figures 3-7 show forms for different types of reports.

To date, 27 States and the CEC (for 12 States) have submitted reports pursuant either to the invitation in 1992 or to the scheme endorsed by the Board in 1993. An additional 13 States have replied, but have not yet sent any reports.
FIG. 3. Form for reports of export of nuclear material.

FIG. 4. Form for reports of import of nuclear material.
Country

Report of Production of Nuclear Material for the Period

<table>
<thead>
<tr>
<th>From</th>
<th>To</th>
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<tbody>
<tr>
<td>Uranium</td>
<td>Thorium</td>
</tr>
</tbody>
</table>

FIG. 5. Form for report of production of nuclear material.

4.2. Internal follow-up action

Data processing and routine communications with States, similar to those established for information received by the IAEA under Safeguards Agreements, have already started. However, implementation is proceeding slowly because of the small amount of information received so far. In order to encourage participation in the scheme and to initiate reporting, the Secretariat has also had contact with States which have not yet responded or submitted any reports.

In addition to the establishment of a database on reports of equipment and non-nuclear material from States, it will be necessary to establish and operate a subsidiary information system, such as that for open-source information, which will be required to utilize effectively the information concerning exports (and imports when available) of equipment and non-nuclear material. Work on the establishment of such an information system has already started.

Systematic analysis of information from safeguards inspections, the reporting scheme on exports/imports of nuclear material and on exports of specified equipment and non-nuclear material, open-source and other data such as IAEA non-safeguards data is very important for identifying at an early stage any instance in which the available information on the nuclear activities of a State appears to be inconsistent with its declaration to the IAEA Secretariat. A tentative outline of the functional
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<tbody>
<tr>
<td>4. Date of Export License:</td>
<td>5. Export License Number:</td>
<td></td>
</tr>
<tr>
<td>6. Issuing Authority:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>7. Exporting Organization:</td>
<td></td>
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<tr>
<td>8. Importing Country:</td>
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<tr>
<td>9. Importing Organization:</td>
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<tr>
<td>10. Location:</td>
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<tr>
<td>11. Government Agency:</td>
<td></td>
<td></td>
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<tr>
<td>12. Date of Export:</td>
<td>13. Expected Date of Arrival:</td>
<td></td>
</tr>
<tr>
<td>17. Description of Item (use additional form if necessary and complete items 1 and 2):</td>
<td></td>
<td></td>
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</tbody>
</table>

**FIG. 6.** Form for report of export of specified equipment or non-nuclear material.

<table>
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<tbody>
<tr>
<td>4. Importing Organization:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>5. Location:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>6. Exporting Country:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>7. Exporting Organization:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>12. Description of Item (use additional form if necessary and complete items 1 and 2):</td>
<td></td>
<td></td>
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</tbody>
</table>

**FIG. 7.** Form for report of import of specified equipment or non-nuclear material.
The following points, especially in connection with specified equipment and non-nuclear material, should be considered in more detail in the early stages of implementation of the reporting scheme:

- Standardization of reporting;
- Establishment of communication channels;
- Information on granting of licences for import and export;
- Information on denial of licences for import and export;
- Follow-up activities and consultations with recipients of export information;
- Reporting of imports of specified equipment and non-nuclear material;
- Integration of the reporting scheme and open sources and other databases;
- Analysis of the information.
5. CONCLUSION

Through the new reporting scheme an increased level of openness and transparency of nuclear activities will be achieved, and the IAEA will be able to provide better assurance on its safeguards system.

Although some progress has been made, much remains to be done. The IAEA will try hard to realize the proposed reporting scheme in co-operation with its Member States, and the persons concerned will certainly assist the IAEA in its endeavours. It is expected that the good co-operation in this field will continue.
IAEA DEVELOPMENT PROGRAMME FOR A STRENGTHENED AND MORE COST EFFECTIVE SAFEGUARDS SYSTEM

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Abstract

IAEA DEVELOPMENT PROGRAMME FOR A STRENGTHENED AND MORE COST EFFECTIVE SAFEGUARDS SYSTEM.

The IAEA is evaluating a number of measures and recommendations for improving the cost effectiveness of its safeguards system. The measures under evaluation, through a two-year development programme known as 'Programme 93 + 2', were recommended by the Standing Advisory Group on Safeguards Implementation (SAGSI). SAGSI is comprised of experts appointed in consultation with governments to advise the Director General on effectiveness and efficiency in the safeguards system. The IAEA programme is designed to further assess, develop and test SAGSI's recommendations and related measures, including evaluations of their technical, legal and financial implications. Six specific areas are being evaluated: cost analysis of present safeguards implementation; assessment of potential cost saving measures; environmental monitoring techniques for safeguards application; increased co-operation with State Systems of Accounting and Control (SSACs) of nuclear material and other measures for improving the cost effectiveness of safeguards; improved analysis of information on a State's nuclear activities; and enhanced safeguards training. The results from these evaluations will be integrated into proposals for more effective and efficient safeguards that are expected to be made to the IAEA Board of Governors in early 1995. An integral part of the programme is the active participation of a number of countries in technical areas, including environmental monitoring field trials. Countries that have offered assistance include Argentina, Finland, France, Germany, Hungary, Indonesia, Russia, South Africa and Sweden. Overall, the programme builds upon actions already taken to strengthen the IAEA's safeguards system. These include Board decisions in late 1991 and early 1992 regarding the early provision and use of design information, and a voluntary reporting scheme on imports and exports of nuclear material, as well as exports of specified equipment and non-nuclear material.

1. INTRODUCTION

In April 1993, the Standing Advisory Group on Safeguards Implementation (SAGSI), acting on a request by the Director General and with the support of the Safeguards Department, reported its recommendations for improving the cost effectiveness of safeguards. The Director General reported to the Board on SAGSI's
recommendations in June 1993 (GOV/2657). The Board requested the Director General to submit to the Board in December 1993 concrete proposals for the assessment, development and testing of the measures proposed by SAGSI.

At the meetings of the Board in December 1993, the Deputy Director General for Safeguards introduced document GOV/2698 describing the Secretariat's development programme for a strengthened and more cost effective safeguards system. Further progress in the programme (called 'Programme 93+2') was reported to the Board in February 1994 (GOV/INF 729). The aim of Programme 93+2 is to develop proposals for a strengthened and cost effective safeguards system, together with an accompanying evaluation of the technical, legal and financial implications, for presentation to the Board of Governors at the March 1995 meeting. In its reporting to the Board, the Secretariat has emphasized that any strengthening measures that go beyond the scope of Safeguards Agreements can only be implemented with the agreement of the States concerned.

Programme 93+2 covers all the areas considered by SAGSI at its April 1993 and subsequent meetings for improving the cost effectiveness of safeguards and deals with both declared and undeclared nuclear activities. It addresses possible new measures for strengthening safeguards, in particular, increasing the capabilities of the IAEA to detect undeclared nuclear activities; cost reductions and enhancing the efficiency of current safeguards activities; alternative procedures and techniques that may be more efficient or effective in carrying out safeguards, including greater co-operation with State Systems of Accounting and Control (SSACs) of nuclear material; and possibilities for replacing some current safeguards procedures by alternative procedures and approaches which maintain the effectiveness of safeguards but which can be implemented with less effort and lower cost.

Work under the Programme is organized into six task areas, with a seventh task for overall management and integration of the results. The core Programme 93+2 team is comprised of a programme manager, six task officers and representatives from the Legal and External Relations Divisions. The work is being accomplished through a series of internal working groups, drawing from expertise throughout the Secretariat, and extensive support from Member States.

Past reports to the Board have emphasized the importance of Member State participation in testing measures being investigated under the Programme. To date, 17 Member States have offered their assistance in the conduct of environmental monitoring field trials and related technical areas and in field trials of other strengthening and more cost effective measures involving alternative safeguards approaches and increased co-operation with SSACs. Additional offers, particularly relating to the application of environmental monitoring techniques in the vicinity of reprocessing operations and to the broader field trials, would be welcomed.

Progress summaries for each of the tasks are given below, including an assessment of the progress expected by the end of 1994. The timing and content of the field trials are decisive for developing the specificity of the proposal for the March 1995
Board. The field trial activities are resource intensive for both the Department of Safeguards and the Member States participating in the trials. Support Division staff are being utilized to the extent possible; however, some field trial activities are much more efficiently and effectively carried out by experienced inspectors knowledgeable about the facilities involved. The scheduling of field trials that makes effective use of experienced staff is complicated by the requirement that Programme 93+2 development activities not interfere with routine implementation of safeguards.

2. PROGRESS REPORT ON PROGRAMME 93+2 TASKS

2.1. Task 1: Cost analysis of present safeguards implementation

The primary objective of Task 1 is the assessment of the costs of implementing safeguards as a function of the magnitude of the various technical safeguards parameters (timeliness, significant quantities and probabilities of detection). The implementation costs associated with current values of these parameters and the cost sensitivity to changes in the values are being evaluated. A reasonable range in the value of each parameter has been defined for the cost assessment. The SAGSI report suggests that if some of the strengthening measures prove effective, then trade-offs could ultimately be possible between those measures and certain elements of the current safeguards system. An important factor in evaluating the merits of any possible trade-offs is relative costs. Task 1 will be completed by the end of 1994.

Experience based estimates of the cost to carry out a particular kind of inspection (e.g. physical inventory verification (PIV) or interim inspection) at a specific facility or the cost to implement safeguards for a year at a specific facility are available. However, assumptions are needed for dealing with the distribution of indirect costs, the amortization of equipment costs, etc. When a change in the magnitude of a technical implementation parameter (e.g. timeliness) results in more or fewer interim inspections, the estimation of the increase or decrease in associated costs is relatively straightforward. When changes in the technical implementation parameters (e.g. significant quantity or probability of detection) result in changes in the level of specific activities performed during an inspection, the associated changes in cost are difficult to estimate because the costs of unit activities (e.g. a non-destructive analysis (NDA) measurement) are facility specific. For example, they depend on the time allocated by a facility operator for the physical inventory taking and verification at a bulk handling facility. To the extent possible, these factors will be accounted for in the cost sensitivity analyses.

The cost data being developed for Task 1 include facility specific and average costs for travel and subsistence; destructive analyses (DA) — (transport and analysis); NDA including equipment procurement and maintenance costs; inspection
effort for verifying inventories (interim and PIV), including establishing the population of items, selecting samples and performing the verification measurements; surveillance, including procurement, installation, servicing and review; seal checking, replacement and verification; records audit and review; and design information verification.

These cost figures relate primarily to activities carried out in the field, and the data collection work is nearly complete. These are the activities most directly affected by any changes in the assigned values of the technical implementation parameters. However, these figures do not include the costs of the myriad of other activities that directly or indirectly support the implementation of safeguards, such as follow-up on discrepant results, training, and development of improved equipment. The sensitivity analysis identifies and quantifies those costs that are affected by a change in the values of the technical implementation parameters.

Other work included in Task 1 is well under way. Papers have been prepared on termination, exemption and substitution of safeguards on nuclear material. The policy paper for the termination of safeguards on nuclear material in waste is nearing completion. The costs of any activities deriving from these papers will be assessed. An evaluation of a change in the starting of safeguards to an earlier point in the uranium verification process has been carried out. Procedures are being prepared for reviewing and verifying design information throughout the facility life cycle, and criteria for accepting final decommissioning are also being developed. The costs of introducing these measures will be assessed.

2.2. Task 2: Assessment of potential cost saving measures

Task 2 has as its objective the identification and evaluation of a number of technical and administrative measures that have the potential to reduce costs associated with the current implementation of safeguards. The development work associated with these measures is being performed either within the Secretariat or through Member State Support Programmes. This task was added to Programme 93+2 following the December 1993 Board of Governors Meeting (see GOV/INF/729). Task 2 is scheduled to be complete by the end of 1994.

Major cost sectors associated with the implementation of safeguards, and thus the areas targeted for potential cost savings, are staff costs (60%), equipment (13%) and travel (13%). As the number of facilities and the quantities of nuclear material under IAEA safeguards continue to increase, so will the need for more efficient use of trained staff. However, staff utilization efficiencies may be improved and travel costs reduced through use of modern technology, through economies in safeguards operational modes, by enlarging existing or establishing new field offices and through efficient use of office automation equipment. Cost savings in the equipment sector may be achieved through greater standardization and the sharing of equipment and analytical services costs with the operator.
The measures so far identified for their potential to improve utilization of staff and reduction in travel and equipment costs are being assessed in the following sub-tasks. The analysis under each sub-task includes identifying the situations where current safeguards activities could be modified or replaced; estimating the resource requirements and costs of the current activities; identifying and assessing the specific changes or replacement activities to be considered; estimating their resource requirements and costs; and determining the cumulative net costs (savings or increase).

Sub-task 2.1. Use of NDA equipment capable of functioning in an unattended mode as a means for reducing inspection frequency and effort. Existing cases include bundle counters at reactors and NDA measurements at mixed oxide (MOX) fuel fabrication facilities. Other cases being identified include the verification of inter-bay transfers of spent fuel in on-load reactors (OLRs) and shipments of finished fuel from fuel fabrication plants to reactors.

Sub-task 2.2. Remote interrogation of NDA and control/surveillance (C/S) equipment to reduce the frequency and duration of inspections for servicing equipment and collecting recorded data. With the development of enhanced technologies, remote interrogation or transmission of C/S data to headquarters or a field office for review and analysis offers savings in inspection effort. Other examples include the bundle counters and core discharge monitors at CANDU stations and NDA (with C/S) measures operating in an unattended mode.

Sub-task 2.3. Use of commercially available (standard) equipment to reduce the use of unique or specialized equipment (more than 100 types of instruments are currently maintained) and the associated extensive and costly support services. Use of standard off-the-shelf equipment in modular form may permit rapid assembly of the equipment needed to meet the ever changing demands and at a lower cost (with assurance of sufficient support for maintenance/upgrading).

Sub-task 2.4. Sharing equipment and installation costs with the operator in instances where the SSAC and the nuclear plant operator have an interest in Secretariat owned and installed safeguards equipment (e.g. for resolving facility accountancy problems or for plant operation reasons). The sub-task involves a cost–benefit analysis of the specific instances (existing and planned) where the Secretariat and operator/State have agreed on the joint use of safeguards equipment whose procurement, installation and maintenance costs are shared.

Sub-task 2.5. Sharing a State's chemical analytical services to limit the need for the Secretariat to increase its capability to carry out highly accurate chemical analysis (DA) at the Safeguards Analytical Laboratory in Seibersdorf. The possible savings in the analytical costs and the costs for the shipment of the samples, if operator
owned and run analytical services can be authenticated and thereby used, will be evaluated.

Sub-task 2.6. Expanded application of certain current safeguarding schemes, such as the zone approach and dual C/S, to increase the effectiveness of safeguards, optimize use of Secretariat resources and reduce inspection effort, while fulfilling the requirements of the safeguards criteria. The cost saving potential of a broader application of these schemes, given that certain technical problems are solved, is being evaluated.

Sub-task 2.7. Mail-in of surveillance video tapes by the SSAC to reduce inspection frequency. This was successfully field tested in 1992–1993. The broad application and associated cost implications are being evaluated.

Sub-task 2.8. Enlargement of the Regional Offices and establishment of new offices as possible means to reduce the heavy travel costs associated with the large inspection effort requirements in States/regions far away from Vienna, such as in South America, the Commonwealth of Independent States (CIS) and the Far East. Cost–benefit analyses regarding the expansion of existing Regional Offices in Tokyo and Toronto and the establishment of new field/regional offices will be performed.

A number of other administrative measures will be reviewed where changes have the potential to reduce the costs of safeguards implementation. Examples include waiving a State's radiation protection requirements for IAEA inspectors where the State accepts the IAEA's radiation protection and monitoring system, adoption of Chemical Weapons Convention provisions such as multiple entry visas and universal designations, and the expanded use of electronic data processing.

2.3. Task 3: Environmental monitoring techniques for safeguards application

Task 3 evaluates the use of environmental monitoring techniques to enhance the Secretariat's ability to detect undeclared nuclear activities. The further development, assessment and use of environmental monitoring as a strengthening measure was one of SAGSI's principal recommendations. The task, which will involve a number of field trials, is focused on:

(a) Evaluating the practicality, effectiveness and cost of the use of environmental monitoring under a range of conditions;
(b) Establishing and documenting environmental signatures associated with a variety of nuclear activities (with an emphasis on uranium enrichment, reactor and reprocessing operations) at both long and short range;
(c) Establishing and documenting sample collection and analytical procedures and quality control requirements; and

(d) Establishing a 'clean room' sample handling and screening capability at Seibersdorf, extending the existing network of analytical laboratories to include the capabilities for the analysis of environmental samples and establishing certification requirements for laboratories added to the network.

Substantive progress has been made in all areas. The results from the currently scheduled field trials should provide a basis for at least an initial evaluation of the applicability of environmental monitoring to safeguards. The field trials have been scheduled such that the evaluation, together with the identification of any open questions, will be complete by the end of 1994. Cost information necessary to support a cost–benefit analysis will also be available by then. The clean laboratory is expected to be in operation before the end of 1995.

The processing of nuclear material, as in any production or manufacturing process, loses small amounts of process materials to the immediate environment, even though great care is taken to limit losses. The extent of the losses depends on the nature of the process, the material, the control measures to limit losses and the migration of losses beyond the immediate environment. Nuclear materials have specific physical properties (e.g. radioactivity) that make it possible to detect and characterize losses that may be present in the environment in only very small quantities. This capability, together with the possibility that specific signatures can be unambiguously correlated with specific nuclear processes, is why environmental monitoring is seen as having promise with respect to the detection of undeclared activities. The goal of the environmental monitoring field trials is to assess the utility of these methods for safeguards application.

A number of Member States have responded to the Secretariat's request for help with offers to host field trials and provide other related technical assistance. The scheduled field trials involve determining the environmental monitoring signatures of reactors (Sweden and Hungary in September and October 1993); enrichment (the United States of America, South Africa and Argentina in March, April and May 1994); reprocessing (Japan and the Russian Federation in April and June 1994); and research centres (Indonesia and the Republic of Korea in May 1994).

Additional environmental samples will be collected in connection with the field trials being planned under Task 4. The number of samples collected in each field trial varies considerably, depending on the sample medium being examined and the type of facility. The overall sample volume is high. For example, in the South African field trial nearly 600 samples (smears, vegetation, soil and water) were collected at 57 sampling points.

The emphasis in the scheduled field trials is on short range monitoring in that all planned sample collections are in the vicinity of nuclear facilities. The trials will provide information on how far from facilities the various signatures can be detected.
This will allow some inferences regarding the long range detection problem; however, currently planned field trials do not include the evaluation of long range monitoring through the collection of high volume air samples or the sampling for gaseous effluents.

Results from the field trial in Sweden are remarkable. Water, sediment and biota samples were collected at 30 locations in coastal waters in the vicinity of five nuclear facilities. Results show that nuclear operations in coastal areas can be detected in water and sediment samples up to 20 km from the facility, depending on local transport and mixing conditions. A minute quantity of plutonium ($\sim 10^{-15}$ g/L) from a water sample taken near a research facility showed high burnup isotopes consistent with spent fuel characterization studies being conducted there and was clearly distinct from samples from other locations which showed only fallout plutonium. Preliminary results from the Hungarian field trial show the presence of Cs isotope signatures in sediments taken at and downstream (40 km) of the reactor site. No such signature was observed upstream of the site.

Specialized laboratories in several Member States are being used for the analysis of environmental samples. A sample distribution and reporting protocol that protects the identity of the samples is in place. Laboratories in these and other Member States will be considered for inclusion in the Secretariat's expanded network. An IAEA procedure has been developed for auditing the quality assurance programmes of candidate laboratories before accepting their participation in the expanded network.

2.4. **Task 4:** Increased co-operation with SSACs and other measures for improving the cost effectiveness of safeguards

Task 4 has three objectives:

(a) To assess measures other than environmental monitoring to strengthen safeguards by providing an increased assurance of the absence of undeclared nuclear activities in a State;

(b) To assess how, and under what conditions, increased co-operation with SSACs could be achieved and what savings could result; and

(c) To assess the possibilities for cost savings in traditional safeguards activities resulting from strengthening the safeguards system.

These three objectives reflect the merger of the original Tasks 2 and 4 as reported in GOV/INF/729. Progress is reported below for each of the three objectives.

By the end of 1994, all of the elements being assessed under this task will have been tested under field trial conditions. There should be sufficient data to give an initial indication at the March 1995 Board of the costs and effectiveness of strengthening measures and of the cost savings, and any impact on effectiveness, achievable
through increased co-operation with SSACs or through new approaches. It is likely that further trials continuing well into 1995 will be required to refine the approaches tested.

2.4.1. Strengthening measures

The strengthening measures being assessed in Task 4 are based primarily on increased transparency, which involves two complementary features — increased physical access (openness) and increased access to information (broader declaration). Current requirements regarding a State’s declaration to the Secretariat are limited to nuclear material (from the starting point of safeguards), associated processes (to the extent that the process related information is needed to safeguard the nuclear material), and nuclear facilities and design information (for facilities containing or expected to contain declared nuclear material) within a State’s territory or under its control. Board of Governors’ decisions regarding the early provision of design information and the voluntary reporting scheme have strengthened the declaration process. The broader declaration being considered in this task, in combination with certain verification activities, is intended to make a State’s nuclear fuel cycle and associated activities as transparent as possible. Current thinking regarding the contents of this broader or expanded declaration is that it should include, in addition to all nuclear material, a description and the location of all nuclear related processes, production, research and development and training. A model expanded declaration has been developed as a programme working paper and is being used by the Secretariat in preparatory consultations with Member States hosting field trials.

Inspector access has been a key issue since the beginning of safeguards. Access for routine inspections, under a comprehensive Safeguards Agreement, is limited to specific points (called ‘strategic points’) deemed necessary for the Secretariat to meet its safeguards obligations. The wider access to be tested in the field trials involves access at any time, and without advance notice and ‘managed’ access for protecting sensitive information. It will include access based on specific information or the need to implement a technical measure (e.g. environmental monitoring). Access elements have been identified which provide the Secretariat with a consistent position in the preparatory consultations with Member States. The access arrangements are intended to permit the IAEA to be able to gain the necessary access and carry out the necessary activities while recognizing the State’s right to protect non-relevant sensitive information. Field trials in Australia, Canada, Finland and Sweden are being defined in this spirit and are under way in three of the four States. The role of the SSACs in facilitating the strengthening measures and the costs of implementing them are also being assessed.

2.4.2. Increased co-operation with SSACs

Co-operation between an SSAC and the Secretariat is a necessary condition for achieving effectiveness of safeguards implementation. In most cases the SSAC’s role
in such co-operation has involved providing information required under the Safeguards Agreement, securing access to facilities and to nuclear material and establishing an accountancy system at facility and State levels. The approach under Task 4 is to evaluate the possible increased degree of co-operation commensurate with SSAC resources and capabilities and with the Secretariat’s need to maintain effectiveness and draw its own independent conclusions. The experience gained in developing the New Partnership Approach with Euratom has been used in developing the approach under Task 4. Some technical elements from the New Partnership Approach are being tested in the field trials in Sweden.

The first step was to devise a questionnaire, to be completed by SSACs, to establish the technical and manpower resources, operational capability, legal powers, information holdings and administrative structure of each SSAC. The questionnaire has been completed and sent out to a few SSACs on a trial basis and some revisions have been made on the basis of comments received.

The next step was to devise a model pattern of increased co-operation by listing all of the candidate activities, largely, but not entirely, related to inspections, which an SSAC could perform, either by itself or jointly with the IAEA in order to increase the efficiency of IAEA verification activities and hence reduce the IAEA’s costs, or to reduce the extent of IAEA activities. The critical requirement against which all of the candidate SSAC activities are tested is that safeguards effectiveness and the IAEA’s ability to draw independent conclusions are maintained. The three following levels of co-operation, as identified by SAGSI, are being investigated, but no decision has been taken on them:

— *enabling activities*, which involve a greater role of the SSAC in pre-inspection arrangements and other preparatory activities to increase the efficiency of IAEA inspections;
— *joint or shared activities*, which could incorporate such things as joint research and development projects and training programmes, shared laboratory and other measurement equipment, and joint efforts to solve problems; and
— *SSAC inspection activities*, whose results would be taken into account by the Secretariat, under specified conditions, with the intent of reducing the extent of Secretariat inspections.

On the basis of the revised questionnaire, the structure, capabilities and resources of the SSAC for all non-nuclear-weapon States having a comprehensive Safeguards Agreement and a significant nuclear fuel cycle will be evaluated. Each SSAC will be fitted into the model described above, according to its capabilities, and an estimate made of the cost savings that would arise if the increased co-operation were implemented. Finally, the issue of regional systems (RSACs) is being addressed.
2.4.3. Cost saving measures in traditional safeguards activities

An increased assurance of the absence of undeclared activities, advances in technology and new approaches could lead to the possibility of reducing the present costs of safeguards aimed at detecting the diversion of declared nuclear material. Task 4 will assess the effectiveness of the strengthening measures in increasing assurance of the absence of undeclared activities. In the context of this increased assurance, elements of the present safeguards system, e.g. timeliness inspection activities for irradiated fuel, will be assessed to see if they can be done differently (possibly with the aid, also, of advanced technology), or less often, or not at all. The cost savings and impact on the effectiveness of such approaches can then be assessed. It is planned to test approaches at light water reactors, OLRs, fuel fabrication plants, irradiated fuel storage facilities and research reactors.

2.5. Task 5: Improved analysis of information on States' nuclear activities

Task 5 focuses on the analysis of information available to the IAEA about a State's nuclear activities. The objective of the task is to ensure the development and establishment of a coherent and comprehensive approach to the acquisition, management and analysis of information from open sources, safeguards inspection data (including results from environmental monitoring), the reporting scheme on imports and exports of nuclear material and exports of specified equipment and non-nuclear material, design information and the expanded declarations referred to in Task 4. A highly disciplined and phased approach is being taken to ensure that the resulting information system will use Secretariat resources effectively to identify at an early stage any instance in which the available information about a State's nuclear activities appears to be inconsistent with its declaration to the Secretariat.

Work under way in Task 5 is focused on three areas:

(a) Development of a diversion critical path structure;
(b) Identification and evaluation of potential information sources; and
(c) Identification and development of computer hardware and software for information management and analysis.

The work on the diversion critical path and the procedures for evaluation of open source information are expected to be complete by the end of 1994. Organizational elements and work on computerization, together with estimates of the resources necessary to implement the system fully, will continue well past that time. The work is being heavily supported by Member State Support Programmes.
2.5.1. Diversion critical path

With expert assistance from Member States, a diversion critical path is being developed as a means to structure both the information and the analysis requirements. This path will include all known pathways, beginning with source material through the various stages of the fuel cycle to the production of weapons-usable material and subsequent weaponization. For example, the conversion block would include all known processes for the conversion of ore concentrates to the chemical forms required for feed to the various enrichment processes, the fabrication of reactor fuel and the production of metal, including all intermediate conversion steps. The possibilities for shortening paths by external procurement and assistance will be included.

Each step and process in the pathways is identified as a node, and any combination of nodes that could result in weapons-usable material is a viable path. Each node is characterized by the process description, any special equipment, infrastructure and non-nuclear material requirements and its potential environmental signatures. The weaponization block will include equipment and material signatures and process identification and descriptors such as the production of tritium, enriched lithium and alpha emitting radionuclides.

The diversion critical path will provide a structure for the analysis of all information available to the IAEA and a basis for the evolution of the model expanded declaration (Task 4). The activities declared by a State as constituting its nuclear related research and development programme will be placed in this same critical path structure. The path structure is being developed in a textual format and as an expert system. The output of this analysis process will be a comprehensive set of profiles reflecting the specific nuclear activities of States.

2.5.2. Potential information sources

A computerized system for storage and retrieval of safeguards relevant information derived from open sources (e.g. public media and scientific publications) and from existing Secretariat databases on power and research reactors and nuclear fuel cycle facilities has been established and periodically updated. It also contains information on States' nuclear regulations, energy requirements, production and resources, nuclear and nuclear related programmes, international co-operation, and companies, firms and organizations working in the nuclear field. The open sources include some information on nuclear exports and imports, including dual use items, which can be compared with the information provided by the State officially (e.g. the reporting scheme and the expanded declaration). The analysis of this information is also being used for further assessment of the expanded declarations.

A number of other open information sources have been identified and are being evaluated. Examples include the International Nuclear Information System (INIS),
a Secretariat database containing approximately 1.5 million documents covering a broad spectrum of nuclear related subjects, and the information published as part of the Emerging Nuclear Suppliers Project (ENSP). Information from exporting States on actions on export licence applications (e.g. denials) for dual use commodities has been identified as a potentially valuable addition to the information provided under the reporting scheme.

2.5.3. Identification and development of computer hardware/software capabilities

The work in this area has identified commercially available software necessary to meet the information management and analysis needs being identified in Tasks 3, 4 and 5. One example is a geo-referenced data management system for the environmental monitoring data and parts of the expanded declaration. The intent is to provide a highly flexible modular information system which can acquire, store and process diverse types of information from the many sources, analyse the information in the context of the diversion critical paths relevant to a particular State, and identify any inconsistencies in the information and other aspects of the information that warrant more specialized evaluation or investigation.

2.6. Task 6: Enhanced safeguards training

Task 6 will identify, develop and implement training programmes to provide the staff of the Secretariat with the necessary skills to carry out the new measures to strengthen and improve the cost effectiveness of safeguards. Completion of the task will establish a training base should decisions be made in the future to proceed with routine implementation of some or all of the measures being tested. Training for design information verification, increased co-operation with SSACs, and environmental monitoring have already been introduced. The work of Tasks 4 and 5 will be sufficiently advanced by the end of 1994 for the training elements necessary to implement the other measures being assessed to be identified. The costs associated with training support for the proposed measures will be assessed.

Training on design information verification throughout the lifetime of facilities has been prepared and incorporated in training courses. Special exercises and workshops on design information verification are being held at Member States' facilities. Belgium has identified a permanently closed down facility that might be available for IAEA training on design information verification, and a technical visit is planned for June 1994.

A new training course, 'IAEA safeguards for SSAC personnel', is being developed which will emphasize different aspects of IAEA activities in the field of international safeguards directly related to co-operation with State systems. The new course will be given for the first time in October 1994 at the IAEA Headquarters. The part of the completed questionnaires on SSACs prepared within Task 4; dealing
with training activities or training needs of State systems is being analysed to identify other opportunities for co-operative or joint training programmes. A Euratom–IAEA joint training programme based on the New Partnership Approach is being developed and will be implemented annually.

The environmental monitoring field trial at a US enrichment facility was a joint sample collection and training exercise. Eleven Secretariat staff were trained in sample collection planning, sample media and sample collection and handling techniques. Emphasis was placed on methods to avoid cross-contamination. Procedures that incorporate the elements of this training exercise have been developed and are being utilized in the ongoing field trials. The staffing of the field trials is being organized in a way to expand the base of trained personnel. The Secretariat, with the help of two Member State Support Programmes, is developing a more formal and extensive training course to support environmental monitoring.

Other training in support of Programme 93+2 must necessarily await developments elsewhere in the Programme. A training exercise that generally supports the strengthening measures of Task 4 through a broadening of inspector observational skills is being developed through a Member State Support Programme. Training on a multimedia geo-referenced information storage and retrieval system is under way, and training on other commercially available software has been requested. The training necessary to implement the systematic analysis of information on a State’s nuclear activities has yet to be identified.

2.7. Task 7: Proposal for strengthening and improving the efficiency of the safeguards system

The integration of the results of Tasks 1 to 6 into proposals for more effective and efficient safeguards will be the final part of the Programme and will be dealt with in Task 7. The proposals will be assessed for effectiveness, cost and the possible trade-offs among the strengthening measures and certain elements of the current system. The relative merits of the proposals will be fully explored and presented. Task 7 will also incorporate a description of any legal implications of the proposals. Furthermore, new administrative and legal measures will be addressed aimed at facilitating safeguards implementation regarding such issues as designation of inspectors and visa requirements. From this analysis the Secretariat will be in a position to make a detailed proposal to the Board on a strengthened and more cost effective system which will cover both the safeguarding of declared material and facilities and the detection of undeclared activities.

An element of each of the tasks is the assessment of the legal issues associated with the measures under consideration. These analyses:

(a) Address the IAEA’s existing authority under current Safeguards Agreements to carry out the measures being assessed;
(b) Identify the extent to which additional authority is necessary to permit the IAEA to implement such measures; and

(c) Describe, where necessary, legal arrangements or instruments for securing the IAEA’s right to do so.

It is possible even at this stage to identify a number of basic issues related to the IAEA’s right of access to various categories of locations, the purposes or activities for which such access may be requested and the role of the Board of Governors and of individual Member States party to Safeguards Agreements in effectuating the various proposals. Other corollary legal issues which might arise, for example, in the context of contractual matters, administration (e.g. travel arrangements, regional offices), health and safety, and liability, will also be addressed.
INCREASED CO-OPERATION BETWEEN THE IAEA AND EURATOM

The New Partnership Approach

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Abstract

INCREASED CO-OPERATION BETWEEN THE IAEA AND EURATOM: THE NEW PARTNERSHIP APPROACH.

A New Partnership Approach (NPA) between the IAEA and Euratom has been founded to replace the 'observation' and 'Joint Team' regimes, which were found inefficient. The paper highlights the circumstances surrounding the birth of the NPA and the status of its implementation. Under the NPA the IAEA can be cost effective without delegating the inspection activities and the responsibilities essential to the fulfilment of the objectives of safeguards implementation. (The IAEA would perform all the required activities to meet the safeguards criteria and draw independent conclusions.) The elements of the NPA are divided into inspection arrangements for various facility types and support activities, and the technical effectiveness of Euratom has enabled the translation of these elements into practical arrangements. Savings of more than 1000 person-days of inspection (PDIs) have been achieved. When fully implemented, the NPA will result in a much reduced inspection effort of about 1000 PDIs, compared to approximately 3100 PDIs which were expended in 1990–1991, in non-nuclear-weapon States of Euratom. The savings accrued are being used in other areas.

1. THE NEED FOR INCREASED CO-OPERATION

The arrangements ('observation' and 'Joint Team' regimes) devised some years ago by the IAEA and Euratom required a high inspection effort and resulted in unnecessary duplication. A review of the inspections at fuel fabrication plants within Euratom, which accounted for 60% of Euratom/IAEA inspections effort under INFCIRC/193, illustrates this point. At two mixed oxide (MOX) fuel fabrication plants, under the Joint Team approach, the IAEA required 650 and 400 person-days of inspection (PDIs), and at a uranium fuel fabrication plant, under the 'observation' regime, the IAEA required 450 PDIs. Although these fuel fabrication plants are extreme cases, even for other facilities the inspection effort of the IAEA was much higher than that required for safeguarding such types of facilities.
Another example is the number of samples taken, transported and analysed in the separate laboratories of the two organizations. In 1990, the IAEA took more than 300 samples for analysis during inspections within Euratom, and it may be assumed that the number taken by Euratom was at least equal to that taken by the IAEA. Therefore, it is reasonable to assume that more than 600 samples were taken for analysis by the IAEA and Euratom, whereas only about half this total number of samples was necessary.

The duplication of resources is not limited to these examples. It also applies to areas such as R&D and training. In most cases, the IAEA and Euratom are working separately in the R&D field. For example, both were developing different video surveillance systems.

It was important to reverse this trend, so as to give effect to two of the basic tenets of the agreement, namely, that the IAEA and Euratom should co-operate in implementing safeguards and should avoid unnecessary duplication of effort.

2. SOLUTION OF THE PROBLEM

2.1. Birth of the New Partnership Approach

Under Article 25 of INFCIRC/193 the two organizations have established a Liaison Committee which meets as a High Level Liaison Committee and a Lower Level Committee. A Working Group was established by the High Level Liaison Committee in September 1991. The task of the Working Group was to examine ways and means by which co-operation and co-ordination between Euratom and the IAEA in the implementation of INFCIRC/193 could be enhanced. The Working Group prepared two reports which were submitted to the High Level Liaison Committee in April 1992. It recommended discontinuation of the existing observation and Joint Team arrangements and the initiation of a partnership approach, which should allow both the IAEA and Euratom to meet their responsibilities under the NPT Safeguards Agreement in the most effective and efficient manner. Furthermore, the Working Group recommended that immediate discussion take place between the two organizations on implementation of the recommended approach.

On 28 April 1992 Director General Hans Blix and Commissioner Cardoso e Cunha met in Brussels and endorsed the recommendations of the Working Group. To this effect an agreement was signed by the Director General and the Commissioner. The agreement provided the necessary components of a New Partnership Approach (NPA) which would lead to improvements in the working arrangements for the application of safeguards within the European Community.

A Technical Group (Euratom and IAEA) was established to work out practical arrangements and this work has proceeded since July 1992.
2.2. Euratom/IAEA Liaison Committee (INFCIRC/193)

The initiative of 28 April 1992 by Director General Hans Blix and Commissioner Cardoso e Cunha also required that re-evaluation of the role of the Liaison Committee and its relationship to its subsidiary bodies be undertaken.

The procedures and working arrangements of the Liaison Committee established under Article 25 of the Protocol to INFCIRC/193 have now been revised to ensure efficient and effective implementation of safeguards in the non-nuclear-weapon States (NNWS) of Euratom. The arrangements were agreed upon on 26 November 1993 and the details are contained in Annex 1 of the Summary of Decisions of the 22nd High Level Liaison Committee (HLLC) meeting held in Brussels on 25–26 November 1993.

2.2. Elements of the New Partnership Approach

Under the NPA the IAEA can be cost effective without delegating to this particular form of a State System of Accounting and Control (SSAC) of nuclear material the inspection activities and the responsibilities essential to the fulfilment of the objectives of safeguards implementation (the IAEA would perform all the required activities to meet the safeguards criteria and draw independent conclusions). This is consistent with the Director General’s statement to the June 1992 Board meeting:

"We assume that arrangements which would be expressive of a genuine partnership would be acceptable to our membership, while arrangements which would be tantamount to a delegation of our safeguards tasks to our partners would not be acceptable. For the Agency, the principal requirement is that an equal partnership must guarantee the Agency’s access to all necessary information and enable it to draw independent conclusions and obtain the necessary degree of assurance and thus meet its own safeguards goals."

The new approach is based inter alia on optimization of the necessary practical arrangements and the use of commonly agreed safeguards approaches and inspection planning, procedures, activities, instruments, methods and techniques.

Other elements of the NPA are:

(a) Increasing common use of technologies to replace, to the extent possible, the physical presence of inspectors by appropriate equipment.

(b) Performance of inspection activities on the basis of the principle ‘one-job — one-person’ supplemented by quality control measures to enable both organizations to satisfy their respective obligations to reach their own independent conclusions and required assurances.
(c) Use of commonly shared analytical capabilities in order to reduce the number of samples to be taken, transported and analysed.

(d) Co-operation in research and development and in the training of inspectors with the aim of achieving a reduction of resources spent on both sides and of leading to commonly agreed products and procedures.

2.4. Summary description of the examples of practical arrangements under the NPA for various facility types

Table I lists the various facility types covered under the NPA.

<table>
<thead>
<tr>
<th>Facility type</th>
<th>No. of facilities</th>
</tr>
</thead>
<tbody>
<tr>
<td>LWRs without MOX</td>
<td>40</td>
</tr>
<tr>
<td>LEU fuel fabrication plants</td>
<td>4</td>
</tr>
<tr>
<td>MOX fuel fabrication plants</td>
<td>3</td>
</tr>
<tr>
<td>Stores with unirradiated plutonium</td>
<td>4</td>
</tr>
<tr>
<td>LWRs with MOX</td>
<td>6</td>
</tr>
<tr>
<td>Wet stores comprising irradiated fuel</td>
<td>8</td>
</tr>
<tr>
<td>Enrichment plants</td>
<td>2</td>
</tr>
<tr>
<td>Dry stores comprising irradiated fuel</td>
<td>4</td>
</tr>
<tr>
<td>Other stores (e.g. UF₆ open air stores)</td>
<td>12</td>
</tr>
<tr>
<td>Research reactors and critical assemblies</td>
<td>46</td>
</tr>
<tr>
<td>Location outside facilities</td>
<td>128</td>
</tr>
</tbody>
</table>

(a) LWRs without mixed oxide

A scheme for a Partnership Approach for LWRs without MOX has been agreed which allows both the IAEA and Euratom to meet their responsibilities under the Agreement (INFCIRC/193).

The arrangements for LWRs involve one physical inventory verification (PIV), three intermittent inspections (IRIs) at quarterly intervals for timeliness purposes and those inspections necessary for verification of shipments of spent fuel. The quarterly
IRIs could be arranged so that they could be performed in a technical and competent manner by one inspector from either organization or could be equally shared by the organizations.

Surveillance and containment measures with tamper indicating capabilities are used to assist the IAEA to reach its independent conclusions. Euratom install and remove the sealed surveillance units at interim inspections. Location indicating devices are fitted to the surveillance units to provide authentication of the location where the units are installed and removed. Configurations of the devices are shown in Figs 1(a), (b) and (c).

Work is under way to develop a tamper indicating device for the Digiquad/Uniplex system (Fig. 1(d)).

Review of the results of surveillance will continue to be performed by the IAEA and Euratom together in Luxembourg. Both Euratom and the IAEA are making arrangements to acquire the necessary implementation experience on selected LWRs without MOX. In parallel, preparations for full implementation of the proposed arrangements are also under way.

(b) Low enriched uranium fuel fabrication plants

In LEU fuel fabrication plants the partners agreed to perform one physical inventory per year and a certain plant specific number of interim inspections. Provided that the inspections and inspection activities are planned and structured in such a way that the requirements of the IAEA are fulfilled, this number would not exceed five routine interim inspections per year.

The development of an unattended mode of measurements for LEU fuel assemblies is under discussion between Euratom and the IAEA. This would enable 100% coverage of flow verification of fuel assemblies.

(c) Mixed oxide fuel fabrication plants

In one MOX fuel fabrication facility the continuous presence of inspectors will be replaced by a presence of four to five days a month while meeting all the requirements of the timely detection and flow verifications. This will be made possible through the use of technologies to replace the physical presence of inspectors by appropriate equipment. Such arrangements for shipment and receipt verifications are depicted in Figs 2(a) and 2(b). A reduction of PDIs from about 410 (1990/1991) to 100 per year is foreseen when all the components are installed.

2.5. Savings under the NPA

To date, savings of more than 1000 person-days of inspection have been achieved, as shown in Fig. 3. This achievement has been mainly due to
FIG. 1. (a) Twin Minolta configuration with tamper-indicating capability; the whole system is sealed. (b) Modular integrated video system (MIVS) configuration with tamper-indicating capability; the whole system is sealed.
FIG. 1. (c) Twin Minolta configuration with tamper-indicating capability; front view. (d) Tamper-indicating devices for Digiquad/Uniplex system, under development.
Discontinuation, for practical purposes, of the observation and Joint Team regimes. Efforts have been directed to ensure that inspection activities are planned to cover only the requirements of the safeguards criteria. A comparison of PDIs between arrangements under observation/Joint Team and the NPA regimes for selected facilities (IAEA effort) is shown in Fig. 4.

Examples of how savings have been achieved are as follows:

- The frequency of inspections of small facilities is limited to the requirements of the criteria (in most cases once per year and once every two years).
- The number of inspections in LEU fuel fabrication plants is limited mainly to those required to cover the requirements of the IAEA safeguards criteria and one inspector is used for interim inspections. In one LEU fuel fabrication plant in Germany PDIs have been reduced from 450 to 65 per year (Fig. 4).
Notes:
1. Neutron measurement performed on PuO₂ container.
2. Positions of PuO₂ containers are sealed with VACOSS seals.
3. Neutron measurement is interfaced with control/surveillance measures (VACOSS and surveillance).
Such arrangement may be performed in the absence of inspectors.
4. Serial number identification of PuO₂ containers by surveillance and VACOSS seals.

Notes:
1. Serial number identification of the PuO₂ containers leaving the store under surveillance and VACOSS seals.
2. Such arrangements may be performed in the absence of inspectors.
3. Containment/surveillance review could be done in Luxembourg

FIG. 2. Proposed arrangements for MOX fuel fabrication plants. (b) Verification activities on receipt of PuO₂ containers.

— The principle of one-job — one-person is effectively utilized (supplemented with quality control measures) at PIV inspections.

— The follow-up and balancing of mixes (FBOM) scheme at one MOX fabrication plant has been abandoned (the plant at present is not fully operating). The FBOM scheme required high inspection effort and was manpower intensive. As a result, the inspection effort requirements in PDIs have been reduced from about 650 to 330 per year.

— Normally one IAEA inspector is sent to interim inspections at one MOX fuel fabrication plant (optimization of resources). The principle of one-job — one-person is effectively utilized (supplemented with quality control measures). The inspection effort in PDIs has been reduced from about 410 (1990/1991) to 290 (1993). Further reduction is expected.
Effort required for LEU plants and WALW and small facilities
(Suspension of LEU plants/small facilities)

\[ \begin{array}{c|c|c|c|c|c|c} \text{Year} & 1990 & 1991 & 1992 & 1993 & 1994 \\ \hline \text{Actual inspection effort expended} & 3292 & 3017 & 1338 & 1679 & 1671 \\ \text{Reduction due to NPA} & 275 & 275 & 275 & 275 & 275 \\ \text{Licence delays} & 250 & 1096 & 250 & 1548 & \\ \text{Net reduction (shutdown decreases + new facilities increases)} & & & & & \\ \hline \end{array} \]

FIG. 3. Distribution of inspection effort (PDI) and reduction due to the NPA (IAEA effort).
Altogether, when fully implemented, the NPA could result in total savings of about 2000 PDIs. The savings accrued from the NPA are being used in other areas (in particular, newly independent States).

2.6. Practical arrangements for NPA support activities

Papers have been agreed covering the common use of inspection instruments, methods and techniques; co-operation in training; and co-operation in R&D. A working group has been established for implementation. Co-operation in the use of commonly shared analysis capabilities and co-operation in the use of new technology are under discussion.

3. WHY IT HAS BEEN POSSIBLE TO ESTABLISH THE NPA WITH EURATOM

3.1. Technical effectiveness of Euratom

The technical effectiveness of Euratom's system and organization has enabled the translation of the elements of the NPA into practical arrangements. The IAEA
intends to continue to make use of Euratom's capabilities to develop and establish optimal practical arrangements, thereby reducing inspection effort while performing activities required by the safeguards criteria and drawing its independent conclusions.

The technical effectiveness of Euratom's system can be illustrated by identifying some of its key features.

(a) Euratom has a fully established system and organization based on more than 30 years experience.

(b) Euratom carries out its functions through the continuous or intermittent presence of its inspectors in facilities.

(c) The range of activities performed by Euratom includes:
   - inspections to cover physical verification activities, flow verifications, verifications at strategic points and audit activities
   - destructive and non-destructive assay
   - establishing historical measurement data
   - stratification and sampling plan preparation
   - material balance evaluation
   - application of containment and surveillance systems.
   - transmission to the IAEA of reports (physical inventory listing, material balance report, inventory change report)
   - design information verification and re-examination
   - transmission of Euratom's findings to the IAEA under Article 21 of the Protocol
   - follow-up activities on anomalies and discrepancies discovered during inspections.

(d) Other capabilities available in Euratom:
   - surveillance review station
   - seal verification
   - calibration of instruments
   - destructive analysis laboratories
   - computer services
   - research and development
   - training.

3.2. Impact of the NPA on the operator of inspected facilities

The NPA brings a series of benefits to the operator of facilities in the non-nuclear-weapon States:

- there is less intrusion on the operator;
- reduced time and effort is spent by the operator on safeguards activities and inspection;
— the operator is dealing with common inspection procedures and arrangements, thus minimizing conflicting demands by the two inspectorates;
— thanks to the advanced transmittal of precise information on programme activities (production, campaigns, shipments, receipts, etc.) by the operator, the inspectorates can plan effective and efficient safeguards activities and inspection scheduling;
— increased co-operation with the operator could reduce the presence of inspectors at the facility, e.g. with the application, by the operator, of VACOSS seals with surveillance.

4. CONCLUSIONS

The above described elements of the NPA for non-nuclear-weapon States of Euratom could be extended to other areas as long as there exist technical capabilities which would enable the IAEA to make use of and maintain independent conclusions. The present activities being undertaken by the IAEA for improving the effectiveness and efficiency of the safeguards system through strengthening measures and increased co-operation with SSACs would provide an opportunity for such an assessment (93 + 2 Programme).

Furthermore, the activities foreseen under the cut-off arrangements could also benefit from extended application of the NPA elements.
1. INTRODUCTION

The United States Department of Energy (DOE) is committed to developing technologies to meet escalating requirements for the IAEA Non-Proliferation Treaty (NPT) monitoring and associated inspections. This commitment involves the customization and transfer of existing remote monitoring/information management technologies for use by the IAEA. We describe here an information management system called international nuclear safeguards inspection support tool (INSIST), which was developed by the Pacific Northwest Laboratory (PNL)\(^1\) to support the IAEA Action Team in its role of monitoring and verifying compliance under United Nations Special Commission (UNSC) Resolutions 687, 707 and 715.

Initial emphasis was placed on developing and deploying technology and databases customized to support the IAEA Action Team. Throughout the design and customization of INSIST, emphasis was placed on information storage and retrieval capabilities for data gathered by the Action Team. In addition, PNL provided the Action Team with maps and satellite images and other relevant Iraqi databases to further facilitate the following activities:

- Monitoring nuclear activities, facility operations and nuclear material inventories;
- Assisting in inspection planning and training;
- Providing post-inspection analysis;
- Providing on-site inspection support;
- Reporting on inspection findings.

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\(^1\) Pacific Northwest Laboratory is operated by the Battelle Memorial Institute for the United States Department of Energy under Contract DE-AC06-76RLO 1830.
Because this effort transferred the existing DOE capabilities in information management and demanded rapid development and deployment of a workstation at the IAEA in Vienna, the existing capabilities were customized to provide the IAEA Action Team with a simple but effective tool for storing and retrieving inspection information. At all phases of this effort, interaction with the end-user was crucial to maximize user involvement in defining, prioritizing and co-developing capabilities. The goal was to encourage the user to become an integral part of the development of future, more advanced capabilities. To achieve this goal, PNL staff have been working with the Action Team in Vienna to better evaluate technologies to support future needs.

2. DESCRIPTION OF THE WORKSTATION

The INSIST workstation provides multimedia data management to accommodate multisource inspection data. Information in INSIST is geographically referenced, providing an easy and effective tool for storing and retrieving worldwide maps, satellite and aerial imagery, conventional photography, site and facility information, inspection videos, and text. To do this, INSIST integrates analogue (video)

FIG. 1. The IAEA Action Team's INSIST workstation — a storage and retrieval tool for inspection data. INSIST has been in operation at the IAEA since August 1993.
and digital processing techniques. Video uniquely provides rapid access and nearly unlimited storage capacity. In addition, because video has been developed for and deployed in the commercial market, it is an inexpensive, familiar and readily available technology.

INSIST was developed on a UNIX based computer, linked to several video processing and input/output peripheral devices (Fig. 1). When possible, equipment was selected to accommodate NTSC (US) and PAL (European) video standards, and a switchable 220/110 V power supply. A read-only videodisk player is used to retrieve videodisk maps and charts, and a WORM laserdisk recorder stores image based inspection data (satellite imagery, photography, etc.). A specialized video card provides video display and processing on the workstation. Videotaped images are displayed on the computer using a serial-controlled VCR. Text and graphics are input using a scanner and can be output with a colour PostScript laser printer. For briefing purposes, the video signal can be routed to a large-screen TV monitor.

To make INSIST intuitive and comfortable for users, a major effort was undertaken to design a simple user interface. In keeping with the industry wide migration to common graphical user interface (GUI) standards, PNL ported existing and customized features into an all-inclusive Motif interface. Users accustomed to other Motif or Windows programs will immediately be able to engage basic features and will be encouraged to explore the more advanced features of INSIST because of the ease of use and the familiar feel of the INSIST interface. A great deal of attention was given to make navigation through the different data sets and features of INSIST as natural as possible, with a minimum of requirements for learned commands or keyboard procedures. The majority of INSIST user interaction is performed with the mouse, which will allow both novice and experienced computer users to develop a high level of expertise in interacting with INSIST's data sets. The importance of this strategy lies in the fact that many nuclear safeguards experts have limited experience using advanced computer systems.

3. INSIST USER INTERACTION

The INSIST interface is shown in Fig. 2. User information stored on a videodisk recorder is displayed in the video window shown at the top right of Fig. 2. Examples of information include videodisk maps and charts developed by the US Government, satellite imagery (Landsat, SPOT, and Russian 2-metre), aerial photography, conventional photography (primarily from on-site inspections), and live video.

The control panel is shown at the top left of Fig. 2. This panel allows the user to configure and interact with INSIST and to control window displays. All controls are implemented using simple pull-down menus and pop-up action boxes. Command options allow the user to select country, area, or site of interest (FILE), to display
FIG. 2. The INSIST user interface. At the top left is the control panel, providing access to INSIST's multimedia information; at the bottom left is a digital reference map of Baghdad; and at the top right is a window displaying a Russian 2 metre satellite image of Baghdad.
image or map sets (SELECT), and to run commercial or customized application programs such as text retrieval (TOOLS). In addition, the control panel reports the active reference map and video image, the co-ordinates of the cursor, and the land-cover category for the area under the cursor.

The reference window is shown at the bottom left of Fig. 2. This window contains a digital reference image which is used to retrieve and display information in the video window. Reference images are maps and diagrams ranging in scale from a world map to building diagrams. The reference maps provide the ability to zoom from the world scale to a specific area of interest. The reference maps used by the IAEA Action Team include official government maps, Iraq tourist type country and city maps, and very specialized thematic maps (i.e. land use and population density). In addition, specialized applications have been developed to retrieve inspection specific information. An example is an equipment retrieval application in which a design graphic is used as the reference image. This application accesses equipment photographs and detailed design graphics. Another inspection specific application is facility retrieval, in which a site or facility diagram is used as the reference image. Then, by selecting individual buildings from the site diagram, the user can view all information associated with selected buildings. In this way, an inspector who is unfamiliar with a site can become knowledgeable about individual buildings and rooms, their functions and the reported impressions of visiting inspectors.

A commercial text retrieval system has been incorporated into INSIST to provide intelligent text processing and management for the Action Team's extensive collection of inspection reports, declarations, documentation and descriptions. The text retrieval system not only performs basic keyword searches but also allows the user to construct sophisticated queries. The information retrieved from these structured queries is ranked according to relevance to the topic queried. Future enhancements will focus on developing associations between INSIST's textual and geographically based information.

4. CURRENT STATUS AND FUTURE OPPORTUNITIES

Since the deployment of the INSIST workstation for the IAEA Action Team, efforts have been concentrated on enhancing the database, demonstrating system functionality, training users, and gaining insights into future developments to meet the needs of the Action Team. The success of the initial INSIST workstation was instrumental in implementing a follow-up project to develop and deploy a second INSIST workstation to support the Programme 93 + 2 at the IAEA. This workstation, which was deployed in January 1994, will support the field trials under the Programme 93 + 2. Because of the extensive environmental monitoring requirements associated with the field trials, software development will involve incorporating environmental data analysis and management functionality. More recently, the DOE
has initiated a programme to transfer INSIST-like capabilities to the UNSCEAR in New York to support its role in Iraqi inspections. Delivery of this workstation is scheduled for June 1994.

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TRIALS IN AUSTRALIA RELATED TO THE ASSURANCE OF THE ABSENCE OF SIGNIFICANT UNDECLARED NUCLEAR ACTIVITIES

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1. INTRODUCTION

The nuclear weapons development programme in Iraq used facilities and nuclear material that were almost entirely separate from its safeguarded facilities and material. To detect such a strategy, the IAEA inspectors need the freedom of access that hitherto has been available only through a special inspection under paragraph 73 of INFCIRC/153. Also, they must be free to base a decision to carry out an inspection upon any information available to the IAEA, not just information derived in the ordinary course of implementing safeguards. (Article VIII A of the IAEA’s Statute obliges Member States to make available to the IAEA any information which would help it.)
2. **EXTENDED ACCESS INSPECTIONS**

As noted above, the IAEA needs the capability to investigate indications suggesting undeclared nuclear activities. To achieve this, there is a case for carrying out inspections in NPT States at locations other than strategic points, without the strong and location specific supporting evidence needed for a true special inspection, as endorsed by the IAEA Board of Governors in February 1992.

Before a right of this kind can be established as a widely accepted principle of comprehensive safeguards agreements, the ground rules under which it would operate need to be formulated. The Chemical Weapons Convention (CWC) provides useful guidance for that.

### 2.1. Environmental monitoring

The results of environmental monitoring based on the measurement of fission products or nuclear material in samples such as water, soil or air may have an important role to play in complementing the extended access inspection concept. The capabilities of these techniques in 'long range' and 'short range' detection activities need to be determined. However, it would seem reasonable to expect that 'short range' monitoring could be used near an installation to confirm that no undeclared nuclear activities are taking place in this installation.

### 2.2. Proposed Australian–IAEA trials

Together with the IAEA, Australia is currently developing trials of an extended access regime (including environmental monitoring). Such trials are a useful opportunity both to develop the modalities for extended access arrangements and to test the practicalities of the concept.

Discussions with the IAEA cover two broad test cases:

(a) **Access to declared nuclear installations:**

At, and in the vicinity of, declared nuclear installations, Australia has offered to give IAEA inspectors:

- A right of access to areas additional to the strategic points specified in Facility Attachments; and
- A right of access both to strategic points and to other places around declared installations, at any time, i.e. on an unannounced or short-notice basis.

(b) **Access to locations other than declared installations** (anywhere in Australia).

Australia has proposed trials of inspections around declared nuclear installations (case (a)) without any principal restriction. However, it is necessary to elaborate detailed modalities primarily for access to locations other than declared installations, i.e. case (b).
There are several aspects of the proposed extended access regime in case (b) which require detailed elaboration in the course of the trials, including:

— The 'trigger' for an inspection of this kind;
— The degree of consultation required between the IAEA and the State concerned;
— Protection of confidentiality at sensitive sites (managed access);
— Designation of inspectors.

3. ASSESSMENT

The trials of an extended access regime also address the way in which the effectiveness of such a regime might be assessed either at the design level (i.e. for planning purposes, before the regime is implemented) or at the evaluation level (for reporting on performance after the event, e.g. in the Safeguards Implementation Report).

One approach is to examine the redundancies and complementarities between the proposed new safeguards measures and the conventional safeguards measures, using an ‘acquisition route’ approach, such as that described in Ref. [1]. Further development of the approach could include the following steps:

(a) Development of descriptors derived from actual experience for the detection probabilities and false alarm rates of the individual components of the overall safeguards approach.

(b) Development of rules for combining the descriptive probabilities and the false alarm rates for cases where a given acquisition route is covered by more than one component of the approach. This would include cases where components are applied sequentially, in the sense that an extended access inspection might follow other indications from analysis of the enhanced information flow.

(c) Use of these descriptors and rules to arrive at a descriptive overall detection probability and false alarm rate for each acquisition route.

4. CONCLUSIONS

It is possible to identify trade-offs for accepting strengthened safeguards regimes incorporating extended access inspections. Such trade-offs can reduce the frequency of IAEA inspections at declared facilities and in some cases may lead to an increase in the cost effectiveness of the inspections. Trade-offs can be recognized most clearly in cases where the IAEA is able to derive assurance that no unsafeguarded reprocessing facility exists. However, more work is needed regarding the value of trade-offs in all cases, especially where the IAEA is able to derive assurance that no unsafeguarded enrichment facility exists.
REFERENCE

SAFEGUARDS STATISTICS
AND DATA PROCESSING

(Session 6)

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STATISTICAL ANALYSIS EMPLOYED IN IAEA SAFEGUARDS

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Abstract

STATISTICAL ANALYSIS EMPLOYED IN IAEA SAFEGUARDS.

The paper discusses the various areas in which statistical methods apply to nuclear materials accountancy safeguards. These methods relate in one way or another to the most basic accountancy data created by the inspector — the measured amount of nuclear material in a specially selected item. This value is then compared with the operator's declared value and inferences are drawn on the state of control based on the individual difference and on the accumulated operator–inspector difference. Major aspects of the problem that are considered include the inspection sampling plans, material balance evaluations, near real time accountancy, and estimation of measurement error parameters. The focus is on improvements made in statistical methodology within the past two years, and on the current situation with respect to evaluating such improvements and incorporating them into the statistical package of techniques in use at the IAEA.

1. INTRODUCTION

At the Fourth International Conference on the Facility Operations–Safeguards Interface, held in Albuquerque in 1991, a number of papers were delivered by IAEA staff members in the session on Statistical Methods for Nuclear Safeguards [1]. The underlying theme of these papers was the development and application of improved statistical methods by the IAEA to enhance nuclear materials accountancy safeguards. This improved methodology was made possible by improved verification measurement capabilities, by advances in standardized data reporting and processing, and by increasingly more powerful computer capabilities that permit the use of statistical tools heretofore infeasible to apply.

The key development underlying all of these advances has been the evolution of a measurement error database. Without this database, the body of statistical techniques would have been primarily of academic interest; with this database, inspection planning and the drawing of conclusions based on inspection verification data has become more objective and quantitative, owing to the combination of the improved database and improved statistical techniques.
This paper discusses the various areas in which statistical methods apply to nuclear materials accountancy safeguards. The focus is on the development and application of statistical methods relevant to IAEA inspection planning and data analysis. The methods have broader application possibilities.

2. OPERATOR–INSPECTOR DIFFERENCES

Before considering specific statistical methods, some general statements are made about the IAEA’s approach to verifying nuclear materials accounts. The problem is to implement a strategy for verifying materials accounts when the resources necessary to measure every item are lacking. Thus, the problem becomes one of making inferences about the state of control of all items by ascertaining the state of control of some suitably selected subset of items, i.e. it reduces to a problem in statistical sampling and inference.

We defer for a moment the question of how a sample of items is selected for verification measurement. The two pieces of information associated with each item sampled are the operator’s declared amount of material and the inspector’s measured amount. The difference between the two amounts, expressed either on an absolute basis or on a relative basis, is called the operator–inspector difference. The resulting sample of differences is regarded as a sample of differences from the total set of differences that would be observed if all items were measured. A historical data file is maintained of operator–inspector differences identified by facility, stratum and method of measurement. This historical data file is labelled the operator–inspector difference database (OIDDB) and serves a wide variety of needs. Before considering the various ways in which the database is exploited, some remarks are made about the status of the database.

3. MEASUREMENT ERROR DATABASE

The OIDDB used for the storage of verification data has been in existence for destructive analysis (DA) data since 1981. A few years ago, with the improvements of non-destructive analysis (NDA) techniques, the database was extended to include NDA data as well. Evaluations of the data provide current estimates of the measurement error parameters classified by facility, stratum and measurement method.

Up to the end of 1993, around 14 000 DA results and 50 000 NDA results had been stored in the central database and used for a variety of purposes. From a descriptive statistics viewpoint, there are two basic computer programs that process these data and yield parameter estimates that are used for a variety of purposes, to be discussed later.
— ERREST yields estimates of random and systematic error parameters for both the operator's declared data and the inspector's measured results. The estimates are stored in another central data file, the measurement error database (MEDB).

— TREND displays the evaluation results in a transparent format providing, in graphic form, control charts which show time trends in average differences and error estimate components. These estimates reflect the observed verification measurements performance and include combined errors associated with both the facility's declaration and the inspector's measurement.

Both ERREST and TREND are versatile and permit selection of subsets of data from the OIDDB on the basis of specific time intervals, stratum, measurement methods, etc.

In addition to the IAEA inspection related purposes of these data, they are useful for other purposes as well. As one important application, they provide a valuable feedback to the developers of instruments and measurement techniques in assessing to what extent the in-field performance approaches the envisioned capability of a given measurement method. Further, they have provided valuable input to the international working groups on DA and NDA that are involved in defining target values [2].

Some specific inspection related uses of the MEDB are now discussed, with emphasis on the statistical techniques.

4. INSPECTION SAMPLING PLANS

Error parameter estimates for both NDA and DA verification measurements are required for the calculation of sample sizes in the IAEA's inspection planning. The inspection sampling plan in use by the IAEA since the beginning of 1991 consists of two versions: an exact iterative version and an approximate plan, the approximate plan being the standard plan in common usage.

Both versions of the plan require knowledge of the error parameter values for the (up to) three verification measurement methods. The plans are basically designed from an attributes inspection viewpoint. Specifically, within a given stratum of material, sample sizes for the (up to) three methods are optimally determined, such that if the operator's declared total for a given stratum differs from the actual total by some goal amount (e.g. a significant quantity), then this discrepancy will be detected with at least \((1 - \beta)\) probability, \(\beta\) being an input parameter determined by the IAEA Safeguards Criteria. 'Detection' means that one or more 'defects' will be identified in the sample of items, a 'defect' consisting of an operator–inspector difference exceeding three measurement error standard deviations associated with that difference. Hence, a knowledge of the error parameters is needed both for calculating the sample sizes and for defining a defect.
Cases can be cited in which it is cost effective to deviate from the sample sizes called for by the standard approximate sampling plan. For example, if the plan calls for measuring, say, four items by DA and twenty items by quantitative NDA, the relative costs of making these two kinds of measurements may be such as to make it attractive to measure many more items by NDA if one DA measurement can be eliminated. One version of the sampling plan algorithm permits the user to input a set of sample sizes and to observe the detection probability as a function of the number and size distribution of the defects. As long as the minimum detection probability exceeds the required \((1 - \beta)\) probability, the plan should be an acceptable alternative to that determined by the standard approach. To facilitate calculation of the non-detection probabilities, an improved approximation to the hypergeometric distribution function was developed as part of the statistical development work routinely under way at the IAEA.

The sampling plans have also been extended recently in another direction. For some facilities (a reprocessing facility being a prime example), all or most of the batches, or items, may be measured by a relatively imprecise method, with a subset of the batches measured by a more precise method and yet another subset measured by a third measurement method. This situation departs considerably from the situation originally envisioned when the sampling plans were developed, the departure being in two directions: the batch or item sizes are large relative to a goal amount, and some items are measured by more than one method. Revised statistical methods for calculating non-detection probabilities were developed to address this situation and, in fact, the aforementioned improved binomial approximation to the hypergeometric distribution function was necessitated by this situation because the parameter values are such that the standard binomial approximation is no longer valid.

Prior to the development and utilization of the above inspection sampling plans, another criterion was invoked, namely that a sufficient number of precise verification measurements must result in a variance of the difference statistic \(D\) being sufficiently small to detect a discrepancy between declared and actual amounts that is accumulated by so-called bias defects, i.e. defects that are so small that they probably escape detection in the attributes inspection. Experience suggested that sample sizes calculated on an attributes basis were rather consistently sufficiently large to result in the required small variance of \(\hat{D}\), and so this criterion was formally deleted in the inspection planning phase. Nevertheless, the verification data play an important role in detecting the aforementioned accumulation of small (bias) defects. This role is discussed in the next section.

5. MATERIAL BALANCE EVALUATIONS

Of elemental importance in reaching a conclusion about a facility’s state of control is the well known and much discussed statistic component MUF (material
unaccounted for). Methods for calculating the standard deviation of a declared MUF, and hence those making a statistical statement about it, are widely documented. The MUF and its associated standard deviation are routinely calculated by the IAEA as an important part of its material balance evaluation.

The MUF is based on the operator’s declared data. Hence, it is of little value unless these data are verified via inspection measurements. The measurements performed according to the inspection sampling plan discussed above play the role of verifying the MUF. The attributes inspection provides some assurance that a few large defects will not invalidate the calculated MUF. With respect to the so-called bias defects, $\hat{D}$ has been a part of the material balance evaluation for many years. As with MUF, the statistical methodology associated with calculating $\hat{D}$ and its standard deviation is well documented and widely known, as is the methodology associated with the so-called inspector’s MUF, namely $MUF-D$, which is simply the operator’s declared MUF adjusted for the estimated bias of $D$.

In the recent past, the evaluation of operator-inspector difference data has been modified and extended, to adapt to changing conditions on the one hand, and to provide a closer correspondence between the mathematical model and the real situation on the other hand. A brief discussion of some of these developments follows.

First and foremost, there is the dramatic improvement in the quality of quantitative NDA inspection data and the attractiveness of including such data as a part of the $\hat{D}$ statistic. Thus, procedures now exist for computing $\hat{D}$ as a weighted mean of the DA and the NDA data. This action results in achieving a smaller standard deviation of $\hat{D}$ at no additional cost, since it utilizes the NDA data, as well as the DA data, in both an attributes mode and a variables mode.

Whether NDA data are included or excluded does not affect an important problem associated with the above analysis. The calculation of the standard deviation of $\hat{D}$ assumes a detailed knowledge of the source of the operator’s declared values such that the error structure, or the model of the declared values, can realistically be assumed to be known. In some cases, this information may be difficult to acquire. This situation has led to the so-called generalized difference statistic $\hat{G}$ — a statistic in which nothing is assumed about the structure of the operator’s declared values.

Internal IAEA memoranda provide guidelines as to when to apply the so-called traditional difference statistic $\hat{D}$ and when to apply the generalized difference statistic $\hat{G}$. Further, these memoranda give procedures that can help to decide when to calculate absolute differences and when to calculate relative differences. In the latter case, the statistic is biased, but a bias correction factor is calculated to remove the bias.

If the generalized difference statistic, whether calculated for absolute or relative differences, is generically denoted by $\hat{G}$, then it is of interest to note that the variance of $MUF-\hat{G}$ is the same as the variance of $\hat{G}$, since, under the generalized model, MUF has no error structure. Although this may seem strange, it is also comforting to know that the variance of $MUF-\hat{D}$ and the variance of $MUF-\hat{G}$ are essen-
tially the same on an expectation basis if, in fact, the operator’s values are affected only by errors of measurement.

There is yet another option for obtaining an estimate of the actual amount of material in a given stratum. This option is feasible in those situations in which the declared values may be of very poor quality, e.g. in a scrap stratum. In this event, a smaller uncertainty in the estimated total amount may result if the declared values are ignored and if the extrapolation to the total is based only on the inspector’s measurements. Again, internal IAEA memoranda give guidelines for deciding when it is feasible to use $MUF\hat{G}$ rather than $MUF\hat{D}$.

Up to this point in the discussion, it has been assumed that there are no more than two measurement methods for the inspector in a given stratum. There are cases, e.g. in a reprocessing facility, where the three measurement methods are of a sufficiently high quality to warrant the inclusion of all three methods in the difference statistic. Further, some items may be measured by more than one method. In a sense, this results in the definition of more than three measurement methods if a distinction is made, for example, between Method 1 applied to items for which no other method was used, and Method 1 applied to items for which also Method 2 was used. To accommodate this situation, the difference statistics, both the traditional $\hat{D}$ and the generalized $\hat{G}$, have been extended to include three or more measurement methods.

6. NEAR REAL TIME ACCOUNTANCY

The concept of near real time accountancy (NRTA) has been a part of the IAEA’s statistical analysis package for a number of years. At the IAEA, NRTA may be described as follows:

The operator provides at regular intervals and under normal plant operating conditions a snapshot of the distribution of nuclear material in the plant. Also under normal plant operating conditions and in the same sequence the IAEA strikes material balances and verifies the inventory.

At the aforementioned Albuquerque Conference [1], a rather detailed account and description was given of NRTA at an automated MOX fuel fabrication plant [3]. This paper presented NRTA as a powerful tool to assure that nuclear materials have been satisfactorily accounted for; the paper also considered a facility specific approach, and then treated aspects of an NRTA system implementation, concluding with a section on experience gained. To quote partially from the conclusions of this paper:

"There are many improvements achievable by implementing an NRTA system, but its dynamic nature which is capable of accumulating assurance against inventory falsifications over time should be particularly emphasized. The experience acquired should initiate a process of reconsideration of the
Safeguards criteria to allow improvements in effectiveness and efficiency of a Safeguards system which will permit the use of more sophisticated concepts, approaches and technology when they become available."

"In this respect a wider perspective should be gained on the improvements achieved in data consistency, detection sensitivity and assurance for falsifications in the inventory vis-à-vis the fulfilment of the Safeguards Criteria requirements on the sample sizes for inventory verification and the prescribed annual PIV. In fact, a reduction in sample size for inventory verification and a dispensation from the annual PIV could be realistic options to be considered. The savings in inspection effort to the Agency and the benefit to the Operator accruing from reduced inspection time and an uninterrupted process operation without clean-out for the PIV are solid incentives to move in that direction. A fully developed and implemented NRTA system is certainly a prerequisite for such a development."

In the ensuing two years, additional experience has been gained with NRTA. The IAEA has implemented NRTA systems in a few facilities. All these systems have been in routine use for several years. The experience gained shows that the decision to implement an NRTA system was worth while in all cases. At the very least, the knowledge about the facilities' state of control improved with the implementation of the NRTA system. When the operator co-operates with the IAEA, the results of the NRTA system help both parties to identify and resolve accountancy and measurement weaknesses in the operator's system. In this respect, implementation of an NRTA system can significantly improve the effectiveness and efficiency of a safeguards system.

On the other hand, experience with NRTA systems has pointed up weaknesses in the standard MUF evaluations based on State reports and leads us to suspect that their safeguards effectiveness is less than previously thought. The main reasons for this are that State reports do not provide operator measurement information and that reported batches are often not defined in the way envisaged, namely as a homogeneous amount of material measured by the operator with the same measurement method.

In all the facilities with an NRTA system in routine use, the current IAEA software capability allows only on-site evaluation of the operator's material balance. The Statistical Analysis Section of the IAEA has spent a lot of effort and time in recent years to develop the new software NMAX (nuclear material accountancy expert), which would also allow us to evaluate operator-inspector differences on the site. This problem is much more difficult than on-site material balance evaluation, since operator declared data as well as inspector measured data have to be analysed. Using an embedded expert system enabled us to develop that software. Access to inspectors' data turns out to be the main problem. In addition, all the instruments used for verification are stand-alone systems that are not designed to produce standardized
outputs and reports. Here is an area where more effort is necessary to cope with future requirements. A first version of the software for on-site operator–inspector difference evaluation is expected to be available for routine use in 1995.

7. ESTIMATION OF MEASUREMENT ERROR PARAMETERS

From the preceding discussions it is apparent that, for a variety of reasons, it is necessary to have good estimates of the error parameters for both the operator and the inspector. For some applications, it is only necessary to have an estimate of the standard deviation for the combined errors of the operator and the inspector. This is the case when calculating inspection sample sizes and when calculating reject limits in attributes inspection. In other applications it is necessary to have estimates of the error parameters for each party. This information is needed for material balance evaluations based on $MUF, \bar{D}$ and $MUF-\bar{D}$.

The basic data from which the needed estimates are derived consist of paired operator–inspector differences for a given facility, stratum, measurement methods and inspection groups. In the terminology of an analysis of variance, the within-inspection group mean square estimates the combined random error parameter for the two parties, and the between-inspections component estimates an important systematic error component, again due to the combined effects of both parties. This between-inspections component describes the shifting relative bias between the operator and the inspector, the assumption being that for a given inspection the relative bias is fixed.

Estimation of the error parameters due to the combined effects of the operator and the inspector is a much simpler problem than is the estimation of individual parameters for each party. However, even here a problem may arise inasmuch as the estimate of the systematic error parameter may be negative. Various estimators that yield positive estimates have been proposed for use at the IAEA [1]. The estimator being used at present has a Bayesian flavour since it requires a knowledge of the relationship between the random error and the systematic error parameters — information that is available from the large body of historical data.

When the problem is to obtain separate estimates of the parameters for the operator and the inspector, the difficulties become greatly magnified. In principle, these separate estimates can be obtained by application of the model known as the Grubbs' model. The estimators make use of the fact that the variance of the operator's values provides an estimate of the so-called process variance plus the operator's random error variance. A similar statement applies to the inspector's data. Since the covariance between operator and inspector provides an estimate of the process variance, it is a simple matter to obtain separate estimates of the random error variances for both parties. By performing a similar analysis on the inspection group means, estimates of the systematic error variances can be derived in a similar fashion.
Unfortunately, because the process variance is often large relative to the measurement error variances, one of the so-called Grubbs' estimates is in many cases negative. Thus, alternative estimators had to be developed to apply in most cases. The estimator being used at present is known as CELE (constrained expected likelihood estimator). Although this estimator always produces positive estimates of the parameters, it often happens that the two estimates are very nearly equal. Work is proceeding on several fronts within the Statistical Analysis Section of the IAEA to develop yet other estimators. A comprehensive simulation study has been outlined, and after it is completed, the set of estimators that are judged to be 'best' in some sense will be identified. It is planned that the TREND and ERREST programs will be modified accordingly.

8. OTHER STATISTICAL TOPICS

There are other areas of statistical development that have been or are being considered, and three important ones are mentioned. A new study is being made of the problem of detecting outliers in the operator-inspector data when such data are used to estimate error parameters. In quite a different problem area, some sets of operator declared data are clearly not based on independently measured values. As an example, there may be groups of items having the same $^{235}\text{U}$ enrichment factor. The generalized difference statistic may be used in this instance, but, as an alternative approach, one can sort out the data on the enrichment factor and obtain more direct estimates of the error parameters. In still another topical area, the statistical problems associated with the calibration of measuring instruments are many and complex. Better statistical methods are continually being sought.

9. DOCUMENTATION OF STATISTICAL METHODS

The IAEA has produced a manual of statistical methods used in its accountancy safeguards activities [4]. The statistical techniques in place as of 1989 — the date of the most recent manual revision — are contained in the manual. As indicated in this paper, there have been a number of improvements in the statistical methodology since 1989, and progress is still under way. All of these techniques, in addition to some that have been developed, evaluated, and then discarded for one reason or another, appear in internal IAEA memoranda. Plans are to revise the manual during 1994. The relevant parts of these internal memoranda will then replace or complement the parts of the manual that currently deal with these techniques. The resulting fifth revised edition will make transparent the methods of statistical analysis employed at the IAEA in its nuclear materials accountancy safeguards programme.
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RECENT DEVELOPMENTS IN AND PRESENT STATE OF VARIABLE SAMPLING

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Abstract

RECENT DEVELOPMENTS IN AND PRESENT STATE OF VARIABLE SAMPLING.
In order to verify the material balance data reported by plant operators, the safeguards authority performs independent measurements on a random sampling basis and compares the resulting data with the reported ones. It is shown that the so-called D-statistic, originally justified with heuristic arguments, and subsequently used in practice for many years, is optimal if the total assumed falsification is small. Furthermore, numerical calculations indicate that for larger total falsification, where a more complicated test procedure would be optimal, the D-test is still useful from a practical point of view. For very large total falsification, the optimal test statistic is complicated; this, however, is not so important since here one is approaching the attribute sampling area. For intermediate values of the total falsification, game theory provides a complicated but interesting solution.

1. INTRODUCTION

International nuclear material safeguards is by general agreement organized in such a way that the plant operators generate all the data necessary for the establishment of a material balance, that the inspectors verify the operators' data with the help of independent measurements and that — if there are no significant differences between the operators' data and the inspectors' findings — the material balance is established with the help of the operators' data.

In this paper recent developments in data verification are discussed. In doing so, only variable sampling [1] procedures are considered which take into account measurement errors, and with the help of which the expected differences between the operators' reported data and the inspectors' findings are quantitatively evaluated.

In the following we shall have in mind primarily the verification of inventory data, first, because it is easier from a methodological point of view, and secondly, because it represents an especially important part of safeguards: whereas flow measurement data sometimes can be verified by comparing shipper and receiver data, there is nothing which can replace inventory data verification using independent measurements.
Data verification presents a statistical problem because of the random sampling procedure and, in the case of variable sampling, because of the existence of statistical measurement errors. Furthermore, since at the end of the verification procedure a decision has to be taken whether or not the data of the operator are accepted, data verification in safeguards is basically a test problem. (An inspector may also be interested in estimating possible defects [2]; since, however, their use is not clear in international safeguards, estimation is not discussed here.) Finally, contrary to conventional statistical problems such as quality control, there is a conflict situation between an operator who may falsify data — otherwise there would be no reason for verifying his data — and the safeguards authority which has to detect any falsification. This means that data verification represents a game-theoretical problem.

Any game-theoretical model of variable sampling data verification problems requires the definition of pay-offs to both the inspector and the operator for detected and undetected illegal behaviour as well as for confirmed and not confirmed legal behaviour (false alarms). It should be mentioned that, at least in the case of false alarms, the interests of the two players need not be opposed. Therefore, non-zero-sum models have to be used.

It can be shown (see, for example, Ref. [3]) that models of this kind can be decomposed into two sub-models such that the first sub-model deals only with the decision of the operator whether or not to act illegally, and the inspector’s choice of the false alarm probability. The second sub-model deals with the operator’s choice of an illegal strategy and the inspector’s choice of a decision procedure for a given false alarm probability. This second sub-model, however, is a zero-sum game with the probability of detecting an illegal action as pay-off to the inspector. This means that, for a given value of the false alarm probability, pay-off parameters need not be known, which is very helpful for practical applications, and furthermore, that the Neyman-Pearson lemma can be used for the determination of the best test [4]. In the following, we consider only the second ‘statistical’ game-theoretical model.

2. GENERAL RESULTS

Let us assume that $N$ data $X_i$, $i = 1, \ldots, N$, have been reported by the operator, and that $n$ ($\leq N$) data are verified by the inspector with the help of independent observations $Y_i$, $i = 1, \ldots, n$, on a random sampling basis. Since the inspector is not interested in the true values of the random variables $X_i$ or $Y_i$ but only in the deviations between corresponding reported and independently generated data, he will construct his test procedure with the help of the differences of these corresponding data. We formulate this test problem as follows:
Definition 1. The differences $Z_i, i = 1, \ldots, n$, between the operator's reported data $X_i$ and the independent findings $Y_i$ of the inspector are assumed to be independent and identically normally distributed random variables with variances

$$\text{var}(Z_i) = \text{var}(Y_i) + \text{var}(Y_i) =: \sigma^2, \quad i = 1, \ldots, n$$

In the following we write $\sigma^2 = 1$ without loss of generality, and with expected values

$$E(Z_i) = \begin{cases} 0 & \text{under } H_0 \\ \mu_i > 0, i = 1, \ldots, n & \text{under } H_1 \end{cases}$$

where $H_0$ is the null hypothesis (no data falsification) and $H_1$ is the alternative hypothesis (data falsification).

As a result of the game-theoretical analysis outlined before, we are looking for that test procedure which maximizes the probability of detection $1 - \beta$, given a fixed value of the false alarm probability. In doing so, we assume that the operator — if at all — will falsify all data by the total amount $\mu$ in such a way that the probability of detection is minimized.

First, for maximum sample size $n = N$ an optimal test procedure is given:

Theorem 1 [5]. Let the sets of pure strategies $\Delta_\alpha$ and $\Gamma_\mu$ of the inspector and of the operator be given by the set $\Delta_\alpha$ of all test procedures for the test problem given by Definition 1 with $n = N$ and given false alarm probability $\alpha$, and

$$\Gamma_\mu := \left\{ (\mu_1, \ldots, \mu_N) : \sum_{i=1}^N \mu_i = \mu > 0, 0 \leq \mu_i, i = 1, \ldots, N \right\}$$

and let the pay-off to the inspector be the detection probability $1 - \beta$, i.e. the probability of rejecting $H_0$ if $H_1$ is true. Consider the two-person zero-sum game $(\Delta_\alpha, \Gamma_\mu, 1 - \beta)$, where the values of $\mu$ and $\alpha$ are known to both players.

Then the saddlepoint strategy of the operator is given by $(\mu/N, \ldots, \mu/N)$; that of the inspector is a test given by the critical region

$$\left\{ (Z_1, \ldots, Z_N) : \sum_{i=1}^N Z_i > \sqrt{NU(1 - \alpha)} \right\}$$
independent of the total falsification $\mu$, and the value of the game, i.e. the guaranteed optimal probability of detection $1 - \beta^*$, is

$$1 - \beta^* = \Phi(\mu\sqrt{N} - U(1 - \alpha))$$

where $\Phi(.)$ is the normal distribution and $U(.)$ is its inverse.

According to this theorem, the test statistic is

$$D : = \sum_{i} Z_i$$

This is the well-known D-test statistic which was proposed first in 1970 [6] for use in nuclear material safeguards; at that time, it was justified by heuristic arguments.

Second, let us assume that only one datum drawn at random out of $N$ reported data (minimum sample size) is verified by the inspector. The following solution to this problem is more complicated than the preceding one:

**Theorem 2** [7]. Consider the two-person zero-sum game $(\Delta, \Gamma, 1 - \beta)$ as in Theorem 1, however with $n = 1$. Let $\mu^*(N)$ be the unique solution of $f(\mu) = 0$, where the function $f(\mu)$ is defined by

$$f(\mu) = \Phi\left(U(1 - \alpha) - \frac{\mu}{N}\right) - \{\Phi[U(1 - \alpha) - \mu]$$

$$+ (N - 1)(1 - \alpha)\}/N$$

Then for $\mu < \mu^*(N)$ the saddlepoint strategy of the operator is $(\mu/N, \ldots, \mu/N)$, whereas for $\mu \geq \mu^*(N)$ it is $(\mu, 0, \ldots, 0)$ or ... or $(0, \ldots, 0, \mu)$.

The saddlepoint strategy of the inspector is the test given by the critical region $\{Z : Z > U(1 - \alpha)\}$; it is independent of the total falsification $\mu$.

The value of the game, i.e. the guaranteed optimal probability of detection $1 - \beta^*$, is

$$\begin{cases} 
\Phi\left(\frac{\mu}{N} - U(1 - \alpha)\right) & \text{for } \mu \leq \mu^*(N) \\
\Phi(\mu - U(1 - \alpha))/N + \left(1 - \frac{1}{N}\right)\alpha & \text{for } \mu \geq \mu^*(N)
\end{cases}$$
The sequence \( \{\mu^*(N)\} \) for \( N = 1, 2, \ldots \) of critical falsifications is strictly increasing; it starts with \( \mu^*(2) = 2U(1 - \alpha) \) and converges to a limit value \( \mu^* \) which is implicitly determined by the equation

\[
\phi(U(1 - \alpha))\mu^* + \Phi(U(1 - \alpha) - \mu^*) - 1 + \alpha = 0
\]

where \( \phi(.) \) is the normal distribution density.

This result recurs in some form or other throughout this problem area. It has an intuitive interpretation: if the total falsification is small, then from a falsification point of view it is best to distribute it over all \( N \) data in order to conceal the falsification by the measurement uncertainty. If, on the other hand, the total falsification is large, it can no longer be concealed in this way, so the number of falsified data has to be as small as possible in order to minimize the probability that the falsified datum is verified and thus discovered.

Third, we consider the general case \( 1 < n < N \), again without restriction to single falsifications. So far, no general solution to this problem is known. It is known, however, that for small total falsification \( \mu \) the D-statistic is again optimal:

**Theorem 3** [5]. If the total falsification \( \mu \) satisfies the condition

\[
\mu \leq \frac{\sqrt{n}}{\binom{N-1}{n-1}} \mu^* \left( \frac{N}{n} \right)
\]

where \( \mu^*(.) \) is given by Theorem 2, then the D-test as well as the equal distribution of the falsification are saddlepoint strategies of the game \( (\Delta_\alpha, \Gamma_\mu, 1 - \beta) \).

Furthermore, we formulate a theorem for very large total falsification, i.e. the attribute sampling case.

**Theorem 4.** For \( \mu \to \infty \), the saddlepoint strategies of the game \( (\Delta_\alpha, \Gamma_\mu, 1 - \beta) \) are given by the test characterized by the critical region \( \{Z_1, \ldots, Z_n : Z_i < U(\sqrt{1-\alpha}) \text{ for } i = 1 \text{ or } 2 \text{ or } \ldots \text{ or } n\} \), and by the falsification strategy \( (\mu, 0, \ldots, 0) \) or its permutations. The value of the game, i.e. the guaranteed optimal probability of detection, is \( 1 - \beta^* = 1 - (1 - \alpha)(1 - n/N) \).

Interestingly enough, this theorem means that the \( n \) observed differences are tested independently, as if there were only one datum, with a false alarm probability of \( \sqrt{1-\alpha} \) each.

There are further analytical results for the general problem considered here, e.g. for the case where there exists an upper limit \( \mu \), for the falsification of a single stratum [8].
3. TWO OUT OF THREE

With the help of extensive numerical calculations [9] and by use of some advanced game theoretical concepts, it was possible to determine saddlepoint solutions for the whole range of the total falsification $\mu$ for $N = 3$ and $n = 2$, which is, as already mentioned, the simplest case not covered by Theorems 1 and 2.

It has been shown that only three different falsification strategies contribute to the saddlepoint strategies of the operator, namely

$$\left(\frac{\mu}{3}, \frac{\mu}{3}, \frac{\mu}{3}\right), \left(\frac{\mu}{2}, \frac{\mu}{2}, 0\right), (\mu, 0, 0)$$

the latter two together with their permutations. Only against the first one is the best test the D-test, as can be shown quickly. For the last one, the critical area of the best test is given by the set of observations

$$\{(Z_1, Z_2) : \exp(\mu Z_1) + \exp(\mu Z_2) > s\}$$

where the relation between the significance threshold and the false alarm probability $\alpha$ is given by

$$1 - \alpha = \int_{-\infty}^{\ln(s)} \Phi\left(\frac{1}{\mu} \ln(s - \exp(\mu x))\right) \phi(x) dx$$

Explicitly, the saddlepoint strategies of the inspector and of the operator are given as follows for $\alpha = 0.05$: For $0 < \mu \leq 3.22$ the equal distribution $(\mu/3, \mu/3, \mu/3)$ of the falsification together with the D-test are saddlepoint strategies. For $5.34 \leq \mu$ they are the one-point falsification strategy $(\mu, 0, 0)$ together with the test as given above. Finally, for the intermediate interval $3.22 < \mu < 5.34$, the saddlepoint strategies of the operator are appropriate mixtures of each two of the above given strategies, together with the best test of the inspector against these mixtures (see also Ref. [10]).

4. D-STATISTIC

We have seen that for arbitrary values of $N$ and $1 < n < N$ and for large total falsification $\mu$, the D-statistic is not the best test statistic for the inspector. Nevertheless, for the moment we continue to consider the test based on the D-statistic and ask which falsification strategy minimizes its detection probability for a given value of $\mu$. Theorem 3 tells us that for small values of $\mu$ the equal distribution
\[ \beta(\mu_1, \ldots, \mu_N) = \sum_i \left( \Phi \left( U(1 - \alpha) - \frac{l \mu}{\sqrt{n} r} \right) \right) \]

\[ \text{with summation for } \max(0, n + r - N) \leq l \leq \min(n, r). \]

\[ \beta(\mu_1, \ldots, \mu_N) = \Phi \left( U(1 - \alpha) - \frac{\sqrt{n} \mu}{N} \right) \]

\[ \beta(\mu, 0, \ldots, 0) = \Phi \left( U(1 - \alpha) - \frac{\mu}{\sqrt{n}} \right) \frac{n}{N} + (1 - \alpha) \left( 1 - \frac{n}{N} \right) \]

For \( n = 1 \), these expressions are obviously the same as those given by Theorem 2; for \( r = n = N \), we get the expressions given by Theorem 1.

Let us consider now explicitly an example, where all marginal falsification strategies are optimal within well-defined ranges of \( \mu \). Figure 1 gives the probability of non-detection as a function of \( \mu \) for \( N = 6 \), \( n = 4 \) and \( \alpha = 0.05 \), which illustrates two very interesting aspects: First, in that region of \( \mu \) where the one-point falsification (1) is not optimal, all probabilities of non-detection are very close to that of the equal distribution for a given value of \( p \), except for that of the one-point distribution. Second, in the optimality region of the one-point distribution, the probability of non-detection approaches very quickly its asymptotic value, which is given by \((1 - \alpha)(1 - n/N)\) for arbitrary values of \( N \) and \( n \), as found already in Theorem 4.

These results are very important for practical applications. Quite generally, one may conclude that it is sufficient to consider only the equal and the one-point distribution of the falsification. This has been proven to be correct for \( n = 1 \), and shown numerically for \( n \leq N/2 \). For all other cases, the probabilities of detection...
of all other marginal falsification strategies which are optimal for some ranges of the total falsification $\mu$ are so close that the differences are practically irrelevant.

This means that one has just to look for that value $\mu^*(N; n)$ of $\mu$, where the probabilities of detection for the two extreme marginal falsification strategies intersect: $\mu^*(N; n)$ therefore is given as the unique solution of $g(\mu) = 0$, where $g(\mu)$ is given by

$$ g(\mu) = \Phi \left( U(1 - \alpha) - \frac{\sqrt{n} \mu}{N} \right) - \Phi \left( U(1 - \alpha) - \frac{\mu}{\sqrt{n}} \right) \frac{n}{N} - (1 - \alpha) \left( 1 - \frac{n}{N} \right) $$

For $N \to \infty$ the sequence $[\mu^*(N; n)]$ converges for fixed $n$ to a limiting value $\mu^*(n)$ which is implicitly determined by the equation

$$ \varphi(U(1 - \alpha)) \frac{\mu^*(n)}{\sqrt{n}} + \Phi \left( U(1 - \alpha) - \frac{\mu^*(n)}{\sqrt{n}} \right) - 1 + \alpha = 0 $$

If we compare this with the limiting critical falsification given by Theorem 2, we observe that we simply have to replace $\mu^*$ by $\mu^*(n)/\sqrt{n}$. 

FIG. 1. Probability of non-detection $\beta(\mu)$ for the D-test as a function of total falsification $\mu$ for $(N, n) = (6, 4)$ and various falsification strategies.
As a result, for given values of \( N \) and \( n \), we have to take for \( \mu < \mu^*(n) \) the equal distribution, and for \( \mu > \mu^*(n) \) the one-point distribution; for the latter case the probability of no detection is practically equal to its asymptotic value for the one-point falsification.

5. CONCLUDING REMARKS

Let us conclude our deliberations with some remarks on several strata. Stratified sampling means considering data in different classes or 'strata'. This is appropriate when there are different types of measured entities, such as volume, weight, etc., or to achieve a greater uniformity if the measurements are of widely different magnitudes, for example. Where several strata of data are to be verified, there is a new element, namely, how the inspector shall allocate his inspection resources (e.g. man-hours or money) to the different strata. Furthermore, how shall he do this if the verification of one datum in one class takes more effort than in another. There is no general solution to these problems, which one can understand in view of the difficult situation even for one stratum. Therefore, special cases and approximations are usually considered [3]. Both are oriented towards the existing solutions for one stratum, especially towards those for the case where there exists an upper limit \( \mu \), for the falsification of a single stratum.

One important special case, also from the practical point of view, is that if the operator acts illegally, he falsifies all the data of one stratum by the same particular amount which is typical for that stratum. In the light of Theorem 2, one expects this assumption to be reasonable for small total falsification \( \mu \). In this case, the set of illegal strategies of the operator is the set of the single falsifications for all classes. The inspector's set of strategies is the set of possible decision procedures for a given false alarm probability and the set of possible modes of allocating the inspection effort to the different strata.

Considering the strata sample sizes of the inspector as continuous variables, this problem can be solved in a satisfying way: The saddlepoint test statistic is the weighted sum of D-statistics for the single strata, and the optimal allocation of the inspection effort is determined by the well-known Neyman-Tschuprow formula (see, for example, Ref. [12]).

REFERENCES


DETECTION PROBABILITIES FOR RANDOM INSPECTIONS IN VARIABLE FLOW SITUATIONS*

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Abstract

DETECTION PROBABILITIES FOR RANDOM INSPECTIONS IN VARIABLE FLOW SITUATIONS.

Improvements in the efficiency and effectiveness of verification of inventory changes are necessary at certain nuclear facilities, for example low enriched uranium fuel fabrication facilities. The IAEA Safeguards Criteria suggest performing interim verifications of inventory changes by random inspections. The paper describes random inspection schemes for inventory change verifications and evaluates the detection probabilities for realistic plant receipt and shipment schedules, and stratum residence times as a function of the inspection frequency and effort, and compares these with the existing inspection strategies.

1. INTRODUCTION

Improvements in the efficiency and effectiveness of verification of inventory changes are necessary at certain nuclear facilities, for example low enriched uranium (LEU) conversion and fuel fabrication facilities.

Because of operational and economic considerations, it may be difficult or undesirable for facilities to retain receipts or shipments for an extended period of time (residence time). On the other hand, the IAEA does not have enough resources to inspect all nuclear materials before they are processed (receipts) or shipped (products). If the schedule for a limited number of inspections is known to the facility operator or if not all materials are covered by inspections, it is possible for operators to divert materials with impunity.

Thus, the method specified in the IAEA Safeguards Criteria to improve inspection effectiveness is carrying out interim verifications of inventory changes by random inspections. It is important to understand the detection probabilities achievable by random inspections for realistic plant receipt and shipment schedules and for

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stratum residence times as a function of the inspection frequency and effort, and to compare these with the detection probabilities achievable with the existing inspection practices.

2. DESCRIPTION OF LEU PLANTS AND CURRENT INTERIM INSPECTIONS

The reference LEU conversion and fabrication plant is assumed to produce 1200 LEU assemblies per year. Each assembly is assumed to contain 500 kg U with 15 kg $^{235}$U. Thus, the total shipment per year, assumed to be domestic, is 240 significant quantities (SQs). The plant receives 400 cylinders of 3% enriched uranium, each containing 1500 kg U with a content of 45 kg $^{235}$U. It is assumed that all these receipts are also domestic.

Figure 1 shows the typical cumulative residence time of fuel assemblies in a plant. Figure 2 shows the typical flow and inventory distribution of fuel assemblies throughout a year. Figure 3 shows the flow and inventory distribution of fuel assemblies with residence times of six days.

It is noted that the residence times for fuel assemblies are 30 days or more for about half of them, seven days or less for about 20% of them, and 4 days or less for 15% of them. The average residence time and the standard deviation of the residence time are 39.8 d and 37.3 d, respectively. The residence times for cylinders are much longer than those for fuel assemblies.

According to the 1991-1995 IAEA Safeguards Criteria, there are normally five interim inspections per year to verify the domestic receipts and shipments, and at least 20% of the annual throughput must be verified. At each interim inspection, the fuel assemblies available for verification but not covered by containment and surveillance (C/S) applications must be counted, and the medium (INFCIRC/153) or high (INFCIRC/66) detection probabilities for gross and partial defects of the assemblies must be verified. For materials under C/S, the effectiveness of C/S is evaluated. Seals must be verified either with low (INFCIRC/153) or with medium (INFCIRC/66) detection probability.

It should also be noted that, according to the Safeguards Criteria, samples of pellets are taken at the rod loading station for bias defect tests at least four times per year or at least two times per year when the facility is part of a zone.

Thus, for the material inspected during interim inspections (at least 20% of the annual throughput at a facility), the detection probability for a diversion of 1 SQ of LEU or more would be at least 50% for facilities with INFCIRC/153 (Corr.) types of Safeguards Agreements and 90% for facilities with INFCIRC/66 types of Safeguards Agreements. If the timing of interim inspections is not known to operators, so that they will not divert any material to be inspected, the overall detection probability for the entire throughput will be the fraction of material covered by
FIG. 1. Cumulative residence time of fuel assemblies.
FIG. 2. Flow and inventory distributions of fuel assemblies.
FIG. 3. Flow distributions of fuel assemblies (residence time = 6 d).
inspection, times 50% (INFCIRC/153) or times 90% (INFCIRC/66). In other words, the overall detection probability will be at least 10% for facilities with INFCIRC/153 (Corr.) types of Safeguards Agreements and 18% for facilities with INFCIRC/66 types of Safeguards Agreements.

However, interim inspections at an LEU facility are often carried out according to a schedule for an inspection trip, in conjunction with scheduled inspections at other facilities, and thus their timing will be known in advance to the operator of the plant. In this case, the operator could divert with impunity some of the material that is not scheduled for verification by inspectors. The diversion will most likely remain undetected if effective concealment measures are adopted at the LEU facility. (It is possible that diversion at one facility might be detected at another facility, for example at the reactor to which the fuel assemblies are shipped. It is also possible that the diversion will be detected because of inconsistencies created by the concealment measures. However, such indirect detection is not within the scope of this study and will not be addressed further here.)

3. BASIC MODEL FOR RANDOM INSPECTIONS

We discuss here the basic model for random inspections developed at Brookhaven National Laboratory (BNL).

Suppose that it is necessary at a certain time to carry out an inspection with an overall detection probability of \( 1 - \beta \) for a diversion of 1 SQ. Instead of carrying out an inspection with a detection probability of \( 1 - \beta \) in the regular inspection regime, it is possible (see Ref. [1]) for inspectors to decide first whether an actual inspection shall be carried out, the probability of which is indicated by \( p \), \( 0 < p < 1 \). (As will be shown below, \( p \) must lie between \( 1 - \beta \) and 1.) Thus, a probabilistic inspection may be omitted. However, if a probabilistic inspection is to be carried out, then the actual inspection will be more intensive than the inspection required in the regular inspection regime.

The number of samples randomly selected for verification during random inspections that must be carried out is indicated by \( n' \), the total number of items in a population is indicated by \( N \), and the number of defective items making up 1 SQ is indicated by \( m \). The overall non-detection probability \( B \) for the random inspection scheme can then be expressed as

\[
B = 1 - p + p \frac{C_{N-m}^{n'}}{C_N^{n'}}
\]

where \( C_N^{n'} \) is the number of combinations of \( n' \) items taken from \( N \) samples. The non-detection probability \( B \) for this inspection is the sum of two terms:

- The probability that the inspection is not carried out \((1 - p)\), and
- The product of the probability that the actual inspection is carried out and the probability that none of the defective items \( m \) is included in the samples \( n' \).
Note that $B$ must be less than $\beta$ to satisfy the detection probability goal. The exact sample size in this scheme can then be calculated from Eq. (1).

It can be seen that, after a variable transformation,

$$\beta' = \frac{\beta + p - 1}{p} = 1 - \frac{1 - \beta}{p}$$

(2)

analogous to the derivation of the original sample size formula used by the IAEA, the sample size in the random inspection scheme can be approximated by:

$$n' = N(1 - \beta'^{1/m})$$

(3)

Thus, in this scheme the sample size depends on the probability $p$ of actually carrying out an inspection; $p$ must be greater than or equal to $1 - \beta$, the desired detection probability. The higher $p$, the smaller is the sample size. When $p = 1$, Eq. (3) reduces to the formula used by the IAEA, as it should when all scheduled inspections are carried out. However, in a more general scheme proposed recently, it is considered unnecessary to carry out all scheduled inspections. Instead, whether a scheduled inspection should actually be carried out is determined on a random basis before the date of inspection. The probability that an inspection will be performed is denoted by $p$. The probability that an inspection will not be performed is denoted by $1 - p$. One 'cost' of this scheme is that when an inspection is actually implemented, more items than are required in the original approach must be sampled for verification. This disadvantage is generally more than counterbalanced by the benefits gained from carrying out fewer actual inspections.

4. EXPANDED MODEL FOR RANDOM INSPECTIONS

To enhance the inspection effectiveness, the basic BNL model is expanded so that inspections can be carried out at any time, as described below.

We consider first a fuel assembly that was produced (or a UF$_6$ cylinder that was received) and was immediately declared to the ‘mailbox’ at time $t_0$. We assume that the item will reside in the facility and will be available for inspection for $T$ units of time (say, days), beginning at $t_0$. A diversion is assumed to have occurred between $t_0$ and $T + t_0$. It is desired to have an inspection scheme with a detection probability of $1 - \beta$ for a possible diversion. In a regular inspection regime, an inspection is performed at $T + t_0$, and no inspections are needed before that time. Regular inspections may not be effective, since the operators may adopt concealment measures before the inspection scheduled at $T + t_0$.

One way to expand the coverage of the regular inspection regime is to include also the intermediate inspection opportunities at $t_i = i + t_0$, $i = 1 \ldots k$, such that

$$\prod_{i=1}^{k} \beta_i \leq \beta$$

(4)
where \( k \leq T \). In other words, the detection probability \( 1 - \beta \) at each opportunity of interim inspection is chosen such that a diversion, assumed to have occurred as soon as the item was declared, is detected with a probability of at least \( 1 - \beta \) before the item becomes unavailable for verification at time \( T + t_0 \).

In a regular inspection scheme with no interim random inspections, the aim is that, at each inspection opportunity \( t_i \) the detection probability be at least \( 1 - \beta_i \); this can be achieved by selecting a proper sampling size such that the probability of not including a defect in the sample \( s_i \) is no greater than \( \beta_i \), i.e. \( s_i \leq \beta_i \). This would increase significantly the work load of inspectors, since many interim inspections that are not required under the IAEA Safeguards Criteria would have to be carried out.

However, use of the basic model of random inspections described in Section 3 can alleviate the problem. According to this model, the mandated detection probability at each opportunity \( t_i \) can be accomplished via a combination of the probabilistic determination of whether or not to carry out an actual inspection and the requirement to execute the actual inspection more intensively than would be required in the case of regular, non-random inspections.

We denote the probability of carrying out the \( i \)-th actual interim inspection by \( p_i \) and the probability of not including a defect in the sample by \( s'_i \). The detection probability requirement becomes:

\[
(1 - p_i) + p_i s'_i \leq \beta_i
\]  

(5)

This requirement must be satisfied at each \( t_i \). Details of the mathematics and the solutions for \( p_i \) and \( s'_i \) are given in Ref. [1]. The probability of actually carrying out an interim inspection \( p_i \) must be higher than \( 1 - \beta_i \), and the sample size for random interim inspections must be larger than that for regular inspections, since \( s'_i \leq (\beta_i - 1 + p_i)/p_i \leq \beta_i \).

The main advantage of random unannounced inspections to be carried out at any time is that it becomes possible to detect diversions before they can be concealed. If concealment measures are possible — for example by borrowing materials from other facilities before inspections or by claiming that the diverted materials have been shipped and inserted into a reactor core — and if inspections can be carried out only at several scheduled dates, operators are likely to divert materials that are not covered by scheduled inspection or to take measures to conceal diversions before the scheduled inspection takes place. On the other hand, if unannounced inspections at any time are allowed, the operator may not have the necessary time to conceal diversions. Thus, an important advantage of the model of random unannounced inspections is that it makes it possible to detect diversions at any time and thus provides a more extensive coverage than that specified by the IAEA Safeguards Criteria.
5. RESULTS AND RECOMMENDATIONS

Several possible schemes have been developed for random inspections for flow verification at LEU conversion and fabrication plants. These inspection schemes are based on the basic model and on the expanded randomization model published earlier [1, 2]. Inspections implemented according to the schemes described will achieve the specified detection probability. Only a fraction of the possible inspection opportunities covering all materials need be implemented probabilistically. In order to achieve the specified overall integrated detection probability, the inspections that are actually carried out will involve more intensive sampling than that which would be necessary for regular inspections.

To achieve a certain detection probability for random inspections, the right of the inspectorate to inspect all materials in a facility must be laid down in an agreement. The inspection opportunities may be based on receipt or shipment dates, packaging dates, or any dates agreed upon between the facility and the inspectorate. Under the basic scheme of random inspections, materials are not required to remain at a facility for a minimum residence time if the accountancy values for them are declared (and committed) before any inspection opportunities and if the materials are available for verification at any time. Furthermore, it is necessary that the facility operator does not know in advance (i.e. before declaration of accountancy values) the times when inspections will not occur. Under this scheme, which is based on the actual operational experience of a large LEU conversion and fabrication plant (which is not currently under international safeguards), about 50 inspection opportunities could provide complete coverage for all materials. With an average of about five interim inspections per year, a 10% detection probability can be achieved. To achieve a higher detection probability, either the number of actual inspections must increase or the total number of inspection opportunities (needed to provide complete coverage) must decrease. For example, assuming two inspection opportunities during each of the months of peak production (say 4 months) and one opportunity during each of the months of off-peak production (say 7 months), about seven interim inspections per year would provide a detection probability of 50%.

Alternatively, the facility could commit itself to declare the accountancy values of all materials as soon as they are available and to retain the materials in the facility for a minimum residence time \( R \) for possible verification. With this agreement, there are two possible inspection opportunities: every \( R \)-th day, or at any time when there is an item in the facility (for example, 4 days for fuel assemblies during peak periods and 7 days during off-peak periods). In either case, with an average of about six interim actual inspections, a detection probability of 10% could be reached. To achieve a 50% detection probability, the residence time must be about three weeks during peak periods and one month during off-peak periods. It is emphasized again that for the expanded randomization model the inspection effectiveness increases
significantly, since the operator of the facility does not know when an actual inspection will take place and thus he cannot apply countermeasures.

Under some special circumstances, it may not be possible for a facility to adhere to the agreed schedule of inspection opportunities. In this case, the facility must notify the inspectorate as soon as possible in order to allow the inspectorate to change its inspection schedules, possibly shifting or adding inspection opportunities. Again, the inspectorate could use the scheme of random inspections to determine when an actual inspection must be carried out. If possible, the addition of an inspection opportunity should be compensated for by decreasing the number of possible future inspection opportunities. This might be accomplished, for example, by extending the intervals between future inspections. In order for this to work properly, effective and timely communication between the facility and the inspectorate must be maintained.

The inspectorate must select randomly the inspection opportunities at which actual inspections must be performed. According to the model, the inspectorate has the freedom to carry out any specific inspections. Furthermore, if the expanded randomization model is used, the inspectorate can skip some of the inspection opportunities as long as the total detection probability is achieved. For any scheme of random inspections to be effective, the total number of actual inspections must remain a random variable. If the total number of interim inspections is fixed and therefore is known to the operator of a facility, the diversion can be optimized against detection by timing of the inspections.

In this study, only LEU facilities have been analysed. For these it is shown that randomization will generally enhance the inspection effectiveness and that inspection resources can be saved. In the course of this study, a prototype computer program was developed to select inspection opportunities that provide complete coverage in a cost efficient manner.

It would be interesting to extend the randomization scheme to a full LEU fuel cycle for a country by a zone approach. Under such an approach, some of the fuel assemblies may be verified at the reactor sites, thus reducing the inspection effort at LEU fabrication plants. Some of the feed cylinders could be verified at the facilities where they are produced or even further upstream. With randomization in a zone approach, not all verifications must actually be implemented and thus the problems associated with the requirement of simultaneous inspections in a zone approach are mitigated. It is also important to study carefully the actual practice of each operations division for each fuel cycle in a State and to develop cost effective randomization schemes accordingly.
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Abstract

NEAR REAL TIME MATERIALS ACCOUNTANCY DEVELOPMENT AT THE TOKAI REPROCESSING PLANT.

The feasibility study on near real time materials accountancy (NRTMA) at Tokai Reprocessing Plant (TRP) was carried out from 1978 to 1981 as one of the tasks of the Tokai Advanced Safeguards Technology Exercise (TASTEX). In 1985, a co-operative field test with the IAEA was performed as one of the tasks of the Japan Support Programme for Agency Safeguards (JASPAS). One of the important results was that NRTMA would be effective in detecting possible abrupt loss. However, another result was that the greater cumulative material unaccounted for (CUMUF) tendency reduced the ability of NRTA to detect protracted loss. Therefore, an improvement of the pump operation and output accountancy procedure has been made in order to reduce this tendency. Consequently, it became fairly flat and slightly downward. The equation to estimate unmeasured inventory in solvent extraction contactors was improved; it had formerly shown a lower inventory than was actually correct.

1. NEAR REAL TIME MATERIALS ACCOUNTANCY

Current near real time materials accountancy (NRTMA) at the Tokai reprocessing plant (TRP) is based upon the measurements of transferred materials and cleanout physical inventory taking. From materials balance, it is confirmed that there is no abnormal loss of materials. NRTMA is one of the ways to assist timely detection of the abnormal loss of materials.

The data required for NRTMA consist of the input of solution of dissolved spent fuel, the output of product Pu nitrate solution and the in-process inventory of the chemical processing area. Most of the in-process inventory under process operation is distributed between the equipment shown in Fig. 1. As the inventory in the Pu evaporator is difficult to measure directly, in-process inventory taking is scheduled after the Pu product is drained into the output accountancy tank and before Pu solution is fed to the evaporator as the next batch operation.
The inventories of the seven tanks (feed tank, buffer tanks, other vessels containing Pu solution) can be measured when in-process inventory taking is performed. The inventory taking includes the volume of the seven tanks and the most recent Pu concentration measured in process control.

The inventories of extraction contactors cannot be measured directly. They are calculated by the equation inferred from Pu concentrations of the solution in the feed tank and the buffer tanks.

2. INVESTIGATION OF MATERIAL UNACCOUNTED FOR (MUF)

The cumulative material unaccounted for (CUMUF) taken in the plant operation campaigns from 1986 to 1989 showed an increase [1]. The MUF in the chemical processing area and the MUF in the Pu storage area were compared. The comparison showed that the former was compensated for by the latter, and that the wrap-up MUF of the chemical processing area and Pu storage area was found to be almost zero. The output accountancy method in the chemical processing area was the factor most affecting the MUF trend.
2.1. Evaporation effect on output accountancy

The usual output accountancy procedure involves the following steps (Fig. 2):

1. draining out Pu solution from the Pu evaporator to the Pu accountancy tank at 40°C
2. circulation of the solution for agitation
3. sample taking
4. reading the manometer before transfer of the solution
5. transferring the solution to the storage tank
6. reading the manometer after transfer of the solution.

The waiting time between steps (3) and (4) takes about ten hours to get the result of the impurity analysis for product quality assurance. Evaporation of the solution in the Pu accountancy tank may occur during these ten hours, because the temperature of the solution rises to about 45°C by mechanical circulation. Experiments to evaluate the evaporation effect revealed that loss of the water by evaporation is not more than 0.2 L, which cannot explain the magnitude of the CUMUF observed in the field test.

2.2. Pump operation in output accountancy

As a pump is used in the output accountancy both for circulation before taking samples and for Pu solution transfer to the Pu storage tank, the pump hold-up volume was investigated in both operation steps.

FIG. 2. Output accountancy procedure.
During circulation of the solution, the pump is almost filled with solution (Fig. 3). But in the transfer stage, the pump hold-up may decrease as the pump sucks in air just before the end of transfer (Fig. 4). It is possible that the difference in the pump hold-up between these two steps caused the upward MUF tendency since 1985. Therefore, experiments were done to evaluate the pump hold-up changing effect.

In order to determine the pump hold-up, the level, density and temperature of the solution in the accountancy tank were measured at the following stages of the output accountancy procedure:

1. after transfer from the Pu evaporator
2. before circulation
3. after circulation
4. before transfer to the Pu storage area
5. after transfer to the Pu storage area.
It was found that the decrease in volume of the pump hold-up after transfer to the Pu storage tanks was about 1 L. Therefore, the transfer operation had to be stopped as soon as the level of the accountancy tank indicated zero, to prevent air inhalation. The difference in the pump hold-up between stages 3 and 5 decreased from about 1/3 to 1/8 following the improvement in pump operation. Consequently, the positive MUF was decreased, and a fairly flat CUMUF trend was achieved (Fig. 5). The trend of CUMUF in the operation campaign after 1991 has turned negative (Fig. 6).

**FIG. 5.** CUMUF in Campaign 1990-I.

**FIG. 6.** CUMUF in Campaign 1991-I.
2.3. Change of output accountancy procedure

In the output accountancy procedure, there was a long time interval between taking a sample and reading the manometer before transfer to the Pu storage tank. It was necessary to make the interval short in order to perform the output accountancy more accurately. Therefore, the procedure was improved so that the waiting time could be as short as reasonably achievable. Then the effect of the improvement was evaluated.

In the previous procedure, both the sample for accountancy (analysis of Pu concentration, density, Pu isotopes) and the sample for product quality assurance (analysis of impurity) were taken at the same time. Before Pu product solution was transferred to the Pu storage tank, the operator needed to confirm the result of impurity analysis; the waiting time was about ten hours. Then, the procedure was improved by taking the following steps, so that the accountancy sample taking would be scheduled just before transfer:

1. draining out Pu solution from the Pu evaporator to the accountancy tank
2. circulating the solution for agitation
3. sample taking for product quality assurance
4. confirming the result of product quality
5. circulating the solution
6. sample taking for accountancy
7. reading the manometer before transfer of the solution
8. transferring the solution to the storage tank
9. reading the manometer after transfer of the solution.

MUF in the chemical processing area was evaluated from the 1991-II campaign, using a new accountancy procedure. The MUF was found to be slightly less. However, non-negligible MUF still existed. More investigation and careful consideration were necessary.

3. UNMEASURED INVENTORY ESTIMATION

The Pu inventory in the chemical processing area is about 20 kg during normal operation. The Pu inventory is distributed in the buffer tanks, Pu evaporator, extraction contactors and miscellaneous piping, etc. The inventory in the buffer tanks can be measured, but the inventory in the other equipment cannot be measured. As each in-process inventory taking is scheduled after the Pu product is drained into the output accountancy tank, the inventory in the Pu evaporator is zero at the time. The rest of the unmeasured inventory is distributed mainly in the solvent extraction contactors.
The unmeasured inventory should be estimated in order to determine the material balance in the chemical process during operation. The unmeasured inventory needs to be estimated with minimum operating information. Therefore, an attempt was made to calculate the unmeasured inventory using the equation inferred from the Pu concentration in the feed tank and the buffer tanks (Fig. 1). The equation was based on the field test data in 1985 [2]. The equation is as follows:

\[
UMI = 701.9 \times C_1 + 2684.3 \times C_2 + 842.9 \times C_3
\]  

(1)

where

UMI is the unmeasured inventory (g),

C1 is the Pu concentration in the feed tank (g/L),

C2 is the Pu concentration in the buffer tank before the partition cycle (g/L),

C3 is the Pu concentration in the buffer tank before the Pu purification cycle (g/L).

Considering NRTMA data in the recent campaigns, the unmeasured inventory calculated by Eq. (1) seemed lower than the inventories derived from the flow measurements. Therefore, an attempt was made to revise the equation using NRTMA data of campaigns from 1990 to 1992.

The new equation was calculated, with the method of least squares, from the Pu concentration in the tanks (shown in Fig. 1) and the Pu inventory in the solvent extraction contactors, which was estimated by flow measurement. However, the contactor inventory derived from flow measurements is strongly affected by the biases of the flow measurement. Therefore, it was assumed that the flow measurement biases accumulate in proportion to the amount of Pu throughput until clean-out (Fig. 7). The revised unmeasured inventory in the contactors was calculated by subtracting the assumed biases from the inventory estimated by flow measurements.

\[\text{CumMUF} = \text{CumMUF}_{\text{flow}} - \text{cumbias} \times \text{Pu throughput} \]

\[\text{CumMUF}_{\text{flow}} = \text{CumMUF}_{\text{flow}} \text{clean-out} + \text{cumbias} \times \text{Pu throughput clean-out} \]

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The revised unmeasured inventory was found to be almost constant and seemed more realistic (Fig. 7).

The new equation was deduced by using the revised inventory and the Pu concentration in the feed tank, three buffer tanks and the accountancy tank, where the number of parameters was increased to five in order to calculate the inventory more correctly. The new equation is as follows:

\[
UMI = 912 \times C1 + 2497 \times C2 + 25.1 \times C3 + 107 \times C4 + 3.5 \times C5
\]

where

UMI is the unmeasured inventory (g),
C1 is the Pu concentration in the feed tank (g/L),
C2 is the Pu concentration in the buffer tank before the partition cycle (g/L),
C3 is the Pu concentration in the buffer tank before the Pu purification cycle (g/L),
C4 is the Pu concentration in the buffer tank before the Pu solution evaporation cycle (g/L),
C5 is the Pu concentration in the Pu accountancy tank (g/L).

Values of CUMUF using Eqs (1, 2) are shown in Fig. 8. The unmeasured inventory calculated by Eq. (1) seems lower than the actual inventory in the contactors because positive CUMUF is observed in the Pu throughput range of 0–150 kg.
On the other hand, that calculated by Eq. (2) seems nearly equal to the actual inventory in the contactors, because CUMUF using the new equation accumulates gradually until clean-out, which leads to the conclusion that Eq. (2) gives a better estimation of the unmeasured inventory. However, data from several more campaigns may be necessary to confirm the validity of the new equation and to estimate its error.

4. CONCLUSION

Evaluation of the pump hold-up effect on the CUMUF shows that the positive MUF bias since 1985 has been caused by the difference in the pump hold-up between circulation and after transfer, and that the difference can be decreased by improvement of the pump operation. Since then, the CUMUF trend has become negative. In the chemical processing area MUF has decreased slightly, following a change in the output accountancy procedure. However, a non-negligible negative CUMUF bias still exists in the chemical process. Investigation of the cause of this negative CUMUF bias is under way.

The equation to calculate the unmeasured inventory in the solvent extraction contactors has been revised and the inventory in the contactors can be estimated more realistically.

Statistical analysis to detect the possible protracted loss and the measurement biases should be carried out in future. However, the procedure to perform NRTMA at TRP may be reasonable. In future, NRTMA will be applied at TRP as one of the ways to enhance the timeliness of the materials accountancy.

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PROCESS FAULT DETECTION AND NON-LINEAR TIME SERIES ANALYSIS FOR ANOMALY DETECTION IN SAFEGUARDS*

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Abstract

PROCESS FAULT DETECTION AND NON-LINEAR TIME SERIES ANALYSIS FOR ANOMALY DETECTION IN SAFEGUARDS.

The paper discusses process fault detection and non-linear time series analysis, which are applied to the analysis of vector-valued and single-valued time series data. Model based process fault detection methods for analysing simulated, multivariate, time series data from a three-tank system are investigated. The model predictions are compared with simulated measurements of the same variables to form residual vectors that are tested for the presence of faults (possible diversions in safeguards terminology). Two methods are evaluated, testing all individual residuals with a univariate $z$ score and testing all variables simultaneously with the Mahalanobis distance, for their ability to detect loss of material from two different leak scenarios from the three-tank system: a leak without replacement and a leak with replacement of the lost volume. Non-linear time series analysis tools have been compared with the linear methods popularized by Box and Jenkins. The paper compares prediction results using three non-linear and two linear modelling methods on each of six simulated time series: two non-linear and four linear time series. The non-linear methods performed better at predicting the non-linear time series and did as well as the linear methods at predicting the linear values.

1. INTRODUCTION

For process control and other reasons, there is increasing automation of chemical processing plants, including spent nuclear fuel reprocessing plants. Consequently, more data will be potentially available for safeguards in future reprocessing. These data will consist of control data and of physical and chemical measurements of process inputs and outputs during plant operations. Not only will more variables be monitored but also data collection will be more frequent than in the past. These data could assist the safeguards function if appropriate data analysis methods can be

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identified. We have investigated two different approaches for the analysis and interpretation of such data: (1) application of process fault detection to the monitoring of multivariate time series data, and (2) application of non-linear methods to the analysis of univariate time series data. The latter differs from the present linear methods used for time series analysis in safeguards.

The first method, *process fault detection and diagnosis*, monitors a vector-valued time series, such as the amount of nitric acid, plutonium and uranium in various tanks and other vessels over time. The second method, *non-linear time series analysis*, monitors a scalar-valued time series, such as the amount of plutonium over time, but allows the serial dependence to have an arbitrary functional form. Present methods used to analyse time series data for material control and accountancy, such as material unaccounted for (MUF), assume that the functional form of the time series is linear.

The goal of *process fault detection and diagnosis* is to develop improved methods for detecting, isolating and identifying deviations from nominal or desired process operating conditions [1]. Process fault detection and diagnosis involves comparing data from process measurements with redundant information to detect and identify faults so that appropriate action can be taken. The redundant information can be either from other process measurements or from process models. The concept is illustrated in Fig. 1 for a case in which a process model is used to provide the redundant information. The process model is developed for 'normal' operating conditions; it uses process knowledge and process inputs to make predictions about the expected state of selected output variables. These predictions are compared with measured values of the same output variables to form residuals (the differences between the measured values and the values predicted by the model), which are tested to determine the presence or absence of a fault at a desired degree of confidence. Because of the unavoidable presence of both modelling errors and measurement errors, non-zero residuals are expected, and therefore criteria are needed for deciding whether

![FIG. 1. Process fault detection.](image-url)
FIG. 2. (a) Time series plot of 100 observations of a non-linear time series. (b) Same time series as in (a), with a non-linear fit superimposed on the scatter plot of the present values versus the past values from the same time series.

A fault has occurred. Mass balance relations in the form of MUF are an example of a process model involving simple consistency relationships. We focus here on the analysis of multivariate residuals representing a single point in time. In another report [2] we address the analysis of cumulative residuals from successive multivariate residuals.
We have compared non-linear time series analysis tools [3, 4] with the linear methods popularized by Box and Jenkins (see Ref. [5]), for analysing time series. The Box–Jenkins analysis assumes that the expected value of any observation is a linear function of some subset of all previous observations and errors. Our approach relies on using historical data to estimate the functional dependence of the present observation on some subset of past observations. We then estimate the expected value of the next observation in the time series using a non-parametric procedure (no distribution assumptions) based on past observations. The next value of the time series is predicted and is then compared with the observed value. If the resulting residual is large, we suspect that an anomaly has occurred. We illustrate the idea in Fig. 2, where (a) is a time series plot of the first 100 values of a non-linear time series, and (b) shows our conditional expected value estimate (solid line) on a scatter plot of the present values versus the past values from the same time series. Note that the non-linear dependence of the present values on the past values is not readily apparent in Fig. 2(a), but is apparent in Fig. 2(b), and that our estimated conditional mean can be viewed as being a scatter plot that is smoother in the lag-one case. This estimated conditional mean can be used together with the estimated prediction error to assist in detecting anomalies.

2. METHODS AND RESULTS: PROCESS FAULT DETECTION

We have applied two multivariate fault detection techniques to simulated data from a three-tank system (Fig. 3) containing nitric acid, plutonium and uranium. The dynamics are described by a system of coupled differential equations based on total mass balances and on individual mass balances for each chemical species for each tank:

\[
\text{[Time rate of change of mass]} = \text{[mass in]} - \text{[mass out]} \quad (1)
\]

For given input flows, initial tank volumes, and initial concentrations of nitric acid, plutonium and uranium, the differential equations are solved to give the outputs, i.e. the volumes and concentrations in the tanks at various times. When density is a linear function of concentration, the equations for tank 1 are

\[
\frac{dV_1}{dt} = F_{11} + F_{12} - F_{21}
\]

\[
\frac{dH_1}{dt} = [(H_{11}^0 F_{11} + H_{12}^0 F_{12}) - (F_{11} + F_{12})H_1]/V_1
\]

\[
\frac{dPu_1}{dt} = [(Pu_{11}^0 F_{11} + Pu_{12}^0 F_{12}) - (F_{11} + F_{12})Pu_1]/V_1
\]

\[
\frac{dU_1}{dt} = [(U_{11}^0 F_{11} + U_{12}^0 F_{12}) - (F_{11} + F_{12})U_1]/V_1
\]
with analogous equations for tanks 2 and 3. The superscript zeros are the tank input concentrations for nitric acid (H), plutonium (Pu) and uranium (U), and the other symbols are defined in Fig. 3. The density of each tank solution is determined from empirical relationships between the density and the concentrations of nitric acid, plutonium and uranium [6, 7]. The system of equations is solved by the Euler method. The volumes, densities and concentrations of plutonium, uranium and nitric acid are the model predictions that are compared with measured values to give residuals, which are tested for faults. Simulated measured values are obtained by adding the following relative standard deviations to the known true values: flow rate: 0.05; tank volume: 0.002; density: 0.002; and nitric acid, plutonium and uranium concentrations: 0.01, 0.002 and 0.004, respectively.

\[ \text{FIG. 3. Three-tank system. } V \text{ and } \rho \text{ refer to the volumes and densities of nitric acid (H), plutonium (Pu) and uranium (U); } C^0 \text{ refers to the concentrations of } H, \text{ Pu and } U \text{ in the input flow } F. \]

The residuals have been evaluated by two different multivariate techniques. The first technique monitors each individual element of the residual vector separately and is a natural extension of the commonly used univariate approach. The univariate test statistic for the variable \( p \) of the vector-valued residual \( i, r_{ip} \), that is expected to be zero, \( H_0: E(r_{ip}) = 0 \), versus the alternative hypothesis \( H_1: E(r_{ip}) \neq 0 \), is

\[
\frac{z_{ip}(\alpha)}{\sigma/\sqrt{n}}
\]

assuming that the standard deviation \( \sigma \) is known. Here, \( n \) is the number of samples used to calculate \( r \) (\( n = 1 \) for this work), and \( E \) denotes the expected value. The critical values with which these test statistics are to be compared come from \( N(0,1) \). The user specifies what significance level (\( \alpha \) value) will be used to signal a fault, depending on the number of false alarms to be tolerated. For uncorrelated, multivariate
normal distributions, if we wish to maintain the same overall significance level for detecting a fault, tests for individual residuals use the Bonferroni method (see Ref. [8]), which replaces \( \alpha \) by \( \alpha/P \) to account for the multiple tests \( P \) is the number of individual \( z \) values being tested. If the standard deviation is not known but must be estimated, critical values from the Student’s \( t \) test distribution are used.

The second multivariate fault detection technique uses a multivariate statistical distance — the Mahalanobis distance — to jointly monitor \( P \) measured variables simultaneously. The Mahalanobis distance of the vector-valued residual \( r_i \) from the mean or target vector \( \bar{r} \) is

\[
Md_i = (r_i - \bar{r})' \Sigma^{-1} (r_i - \bar{r})
\]

if the covariance is known. The Mahalanobis distance for \( r_i \) is compared with user specified critical values from the chi-squared distribution. In the present application, the target vector \( \bar{r} \) is zero.

The covariance matrix \( \Sigma \) is necessary for calculation of the test statistics, \( z_{dp}(\alpha) \) and \( Md_i \). For the three-tank system, this was obtained by performing 1000 ten-hour simulations under no-fault operating conditions. The flow rates and initial tank conditions were the same for each simulation, except for the application of randomly distributed uncertainty to all measured variables. For the initial tank conditions, the uncertainties were applied once, at the beginning of each simulation. For the flow rates, we assumed that new measured values became available every 0.1 h, at which time the model values were updated. Model predictions at the end of each ten-hour simulation were compared with measured values obtained at the same time, and the residuals were calculated.

We also performed a principal component analysis of the 1000 simulated residual vectors. Principal components analysis is often used as a dimensionality reduction method when correlated variables are present. For data vectors containing \( P \) elements, it may be assumed that the components corresponding to the \( Q (Q \leq P) \) largest eigenvalues explain the important internal structure of the data. We used an approach presented in a paper on multivariate process control by Jackson [9]. Jackson suggested rescaling the eigenvectors by the eigenvalues so that each score has a mean of zero and a variance of one. Thus the score, \( t_{ij} \) for residual \( i \) and principal component \( j \), can be tested directly against critical values from the \( N(0,1) \) distribution to determine whether the particular principal component score may be an outlier. In addition, the Mahalanobis distances are easily calculated directly from the scaled scores. For observation \( i \), this is

\[
Md_i = t_i t'_i
\]

We have investigated two diversion scenarios: (1) a steady leak from the second tank without replacement (scenario 1), and (2) the same leak, but with the
FIG. 4. $z$-scores for each variable in the three-tank system for a leak rate of 0.5 L/h. In (a) the lost volume is not replaced. In (b) the lost volume is replaced with water.

FIG. 5. Mahalanobis distances for three leak rates under scenarios 1 and 2.
lost solution replaced by water (scenario 2). In practice, the model would not know about a leak and thus would make erroneous predictions because it would assume 'normal' operations. The true condition, i.e. the loss due to leakage, is reflected in the measured data. The results of fault detection tests for a leak of 0.5 L/h, which was easily detected, are summarized in Fig. 4. The concentrations of plutonium and uranium plus the density were detected as outliers in the second tank (Fig. 4(b)) under scenario 2, whereas only the volume (Fig. 4(a)) was detected as a fault under scenario 1. Replacing the removed volume by water diluted the concentrations sufficiently to make a large difference in all concentration variables as well as in the density, which is based on concentrations. The Mahalanobis distances are shown in Fig. 5 for three different leak rates. In all cases the values are larger and thus more statistically significant for scenario 2. The 0.5 L/h rate was the only one which was significant at the 5% level.

3. UNIVARIATE TIME SERIES ANALYSIS

We have considered univariate time series such as MUF values or other statistics arising from safeguards.

In computer simulations we have experimented with several non-linear estimation methods using both linear and non-linear simulated data sets. We assume in our approach that the same functional dependence between an observation and some subset of the previous observations holds throughout the entire time series. If this assumption is not valid, the time series must be divided into subsets in which the assumption is satisfactory. This requires detailed knowledge of the process that is generating the sequence. Except for this potentially serious problem, the implementation of our procedures is straightforward.

We have implemented Fortran computer codes to perform the two main activities in estimating the conditional mean: choosing the degree of smoothing, and estimating the lag, i.e. the number of previous observations directly affecting the present observation.

Empirical estimates of the lag may enhance understanding of the processes generating the data. In many cases, we have a good idea of a value for the lag. For example, in ordinary MUF sequences, a value of one is often a good first approximation for the lag. This is because the ending inventory for MUF_{j-1} is the beginning inventory for MUF_j. Therefore, if we ignore the effects of systematic errors, the lag is one.

Our main goal is a better detection of anomalies through the use of the best techniques for predicting future values of the time series. By 'best' we mean that the standard error of residuals (the mean-square error of prediction, MSEP) is minimized.
To illustrate non-linear modelling, we have compared the MSEPs using two linear and three non-linear estimation methods as follows:

1. The time series vector is divided into testing and training sets, and it is assumed that no loss has occurred.
2. The training set is used to estimate the conditional mean, making either no assumption about the functional form of the conditional mean or assuming that the conditional mean is linear.
3. The MSEPs evaluated in the test set for the linear and the non-linear methods are compared.

Regarding the MSEP, we have analysed simulated data from the following six time sequences, each observed with error:

(a) \( x_t = 1 - 1.4x_{t-1}^2 + 0.3x_{t-2} \) (non-linear)
(b) \( x_t = 4x_{t-1}(1 - x_{t-1}) \) (non-linear)
(c) \( X_t = a_0 + b_1 e_{t-1} + b_2 e_{t-2} + e_t \) (linear)
(d) \( X_t = a_0 + a_1 X_{t-1} + a_2 X_{t-2} + e_t \) (linear)
(e) \( X_t = a_0 + b_1 e_{t-1} + e_t \) (linear)
(f) \( X_t = a_0 + a_1 X_{t-1} + e_t \) (linear).

We use lower case \( x \) for the two non-linear time series (a) and (b), because we generated the data deterministically and then added observational error. Specifically, we generated the \( x_t \)'s and then added independent \( N(0, 0.05^2) \) random variables to represent observational errors. For the linear series, the errors were

<table>
<thead>
<tr>
<th>Time series</th>
<th>( s^2 )</th>
<th>Linear 1</th>
<th>Linear 2</th>
<th>Non-linear 1</th>
<th>Non-linear 2</th>
<th>Non-linear 3</th>
</tr>
</thead>
<tbody>
<tr>
<td>(a)</td>
<td>0.53</td>
<td>0.47</td>
<td>0.46</td>
<td>0.06</td>
<td>0.01</td>
<td>1.21</td>
</tr>
<tr>
<td>(b)</td>
<td>0.13</td>
<td>0.13</td>
<td>0.13</td>
<td>0.02</td>
<td>0.10</td>
<td>0.94</td>
</tr>
<tr>
<td>(c)</td>
<td>1.36</td>
<td>0.98</td>
<td>1.14</td>
<td>1.48</td>
<td>0.82</td>
<td>0.37</td>
</tr>
<tr>
<td>(d)</td>
<td>1.68</td>
<td>0.98</td>
<td>0.99</td>
<td>1.05</td>
<td>0.60</td>
<td>0.84</td>
</tr>
<tr>
<td>(e)</td>
<td>1.14</td>
<td>1.01</td>
<td>0.99</td>
<td>1.01</td>
<td>0.87</td>
<td>1.48</td>
</tr>
<tr>
<td>(f)</td>
<td>1.30</td>
<td>1.00</td>
<td>0.98</td>
<td>1.00</td>
<td>0.76</td>
<td>0.80</td>
</tr>
</tbody>
</table>
independent \(N(0,1)\) random variables. Therefore, the theoretically lowest achievable MSEp is 0.0025 for series (a) and (b), and 1 for series (c–f). For each of the six time series, we generated training vectors with 1000 observations and testing vectors with 1000 observations. The MSEPs for two methods of linear estimation and for three types of non-linear estimation are shown in Table I for the six cases. We include the sample variance \(s^2\) for each case because \(s^2\) would be the MSE if we used the sample mean as the predicted value.

In Table I, the first linear method is denoted Linear 1, the second linear method is denoted Linear 2, and similarly for the three non-linear methods. The first linear method fits the best possible autoregressive moving average (ARMA) model to the observed sequence. The second linear method fits a linear model to the regression of \(X_t\) on \(X_{t-1}\) or of \(X_t\) on \(X_{t-1}\) and \(X_{t-2}\), depending on which gives a better fit.

The first non-linear method is a conditional mean estimator. For the lag = 1 case, our estimator is

\[
\hat{M}(x) = \left[ \frac{1}{n-1} \sum_{j=1}^{n-1} X_{j+1} \kappa \left( \frac{x - X_j}{h} \right) \right] / \left[ \frac{1}{n} \sum_{j=1}^{n} \kappa \left( \frac{x - X_j}{h} \right) \right] 
\]

where \(\kappa\) is called the kernel. It is usually assumed that \(\kappa(x)\) evaluated at \(x = 0\) is the maximum value of \(\kappa\), and that \(\kappa(x)\) is a decreasing function of \(|x|\). It is usually further assumed that \(\kappa\) is a symmetric probability density function such as the standard normal density. The parameter \(h\) is the bandwidth, which determines the amount of smoothing. For more detail, see Ref. [5], but the idea in Eq. (5) is straightforward. We have \(n\) observations, \(X_1, X_2, \ldots, X_n\), and seek an estimate of \(X_{n+1}\), given the value \(X_{n} = x\). The idea is to use all of the first \(n-1\) observations, but to weigh most heavily the observations that are nearest to the value \(x\). The extension to higher lags is straightforward. We present results here only for the lag-one case for the first non-linear method. The second method differs from the first one, in that the second method does attempt to estimate the lag and uses a different method to choose the bandwidth. Using the second method, the best estimates of the lags for time series (a) through (f) were \(d = 5, 2, 2, 2, 2, 1\), respectively, whereas the correct lags are 2, 1, 2, 2, 1 and 1. The third non-linear method is a computationally intensive method, which appears to be an inconsistent performer in our experiments to date. The method uses the \(k\) nearest neighbours of each point to fit a local linear model at that point. The overall model is then piecewise linear, but can be made to look rather smooth if the pieces are sufficiently short.

In Table I there is not much difference in the MSEP between the two linear methods, except for series (c), where Linear 1 gives a lower value. We expect the second linear method to perform worse on moving average (MA) models than the first linear method because the second linear method relies on the true time series being an autoregressive series. Similarly, all three of our non-linear methods are
designed for autoregressive series. However, it is possible to extend our non-linear methods to accommodate MA models. The details of this extension can be found in Ref. [10].

In comparing the linear methods with the non-linear methods, note that cases (a) and (b) are the only non-linear series. For series (a) and (b) there is a clear advantage in using the Non-linear1 method compared with using the linear methods. However, note that the other two non-linear methods do not perform consistently. In fact, because the error variance for the two linear time series was 1.0, the theoretically lowest achievable variance for predicting them is 1.0. Therefore, the Non-linear2 and Non-linear3 methods sometimes give misleadingly low estimates of the true MSEP. Currently, we have no explanation for this behaviour. At present, we prefer the relative simplicity of the Non-linear1 method and are pleased with its performance on both linear and non-linear time series.

4. SUMMARY AND CONCLUSIONS

Both the multivariate process fault detection method and the non-linear time series method are fairly easy to implement. With respect to safeguards, the main issue is whether international inspectors will be granted access to the larger amounts of data expected to be available at modern reprocessing plants. Until such data are available, we can only test our methods on simulated data.

For the three-tank problem, univariate tests on individual variables as well as on individual principal components were equally effective at detecting losses of material. Because the principal components are linear combinations of individual variables, they might be expected to provide a more sensitive detection of outliers or faults for situations where two or more correlated variables are affected by a fault. With this simulation, the correlations were not strong enough to observe this effect. The multivariate tests based on the Mahalanobis distances were never as sensitive as the univariate tests, probably because the sensitivity is diminished somewhat by those variables that are not affected by a leak. Scenario 2 (leak with replacement) was detected with slightly more sensitivity than was scenario 1 (leak without replacement), perhaps because replacement of the lost volume by water affects four variables (the density, and the concentrations of nitric acid, plutonium and uranium) rather than just one variable, as does a leak without replacement.

For univariate time series, our current recommendation is to include techniques that can test for non-linearity in a package of evaluation methods for time series. If tests do not indicate non-linearity, there is no need to apply non-linear estimation methods. If tests do indicate non-linearity, we recommend using non-linear techniques for estimating the expected value of an observation in the time series sequence. Presumably, if a time series sequence fails the tests for linearity, the expected value of the present observation will be a non-linear function of some subset
of the previous observations. Another potential advantage of non-linear modelling could result from an improved understanding of mechanisms generating the data through detection of unexpected functional dependences. Thus, for reasons other than anomaly detection, we may wish to analyse time sequences containing many elements, using non-linear methods should they become available in the future.

Judging from results obtained using simulated data, the multivariate process control methods will improve our ability to detect loss by exploiting the redundancies between modelled and measured variables. The non-linear time series methods perform as well as the linear methods when the true functional form is linear and they outperform the linear methods when the true functional form is non-linear. However, the price paid in both cases is that more measurements must be made.

REFERENCES

WARNING AND ACTION LINES IN SEQUENTIAL TESTING

A new method of doing near real time accounting

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Abstract

WARNING AND ACTION LINES IN SEQUENTIAL TESTING: A NEW METHOD OF DOING NEAR REAL TIME ACCOUNTING.

The usual statistical process control methods are based on fixed sampling rates. Recently, a combination of cumulative sum (CUSUM) and sequential probability ratio tests (SPRTs), each with their respective sample rate, has been suggested for applications in control problems. When the initial CUSUM statistic shows evidence that the process might be out of control, a high sample rate mode sets in. The control statistic snaps into SPRT to settle quickly the question whether the process has indeed got out of control. From the high sample rate mode the SPRT statistic can either go back to the low sample rate regime with CUSUM or call an alarm. Motivated by this modification of the classical CUSUM scheme, a more general and flexible control statistic process with adaptive variable sample rate, AVAS, is constructed. A key feature is the application of the Markov chain probability model. There are situations, as in safeguards inspections, where obtaining measurements is costly. The authors believe that the proposed approach will significantly reduce costs, yet provide as effective a control as the traditional methods provide.

1. INTRODUCTION

Sequential testing procedures are usually applied in situations where data are observed during successive time intervals until sufficient evidence accumulates to permit making a decision between two hypothesized models that describe the statistical distribution generating the data [1, 2]. For materials accounting applications, where process control techniques are referred to as near real time accounting (NRTA), frequently assumed hypotheses are a no-loss model and a loss model.

The common techniques of sequential analysis assume that the underlying process is a random process of independent and identically distributed (IID) random measurements. In safeguards applications the frequently used measurement quantities, such as material unaccounted for (MUF) and shipper–receiver difference (SRD) series, do not satisfy these conditions. There are, however, procedures available for
transforming the data into an IID sequence. Most statistical process control techniques suggest the use of statistics derived from such sequences. Typical examples are the cumulative sum (CUSUM) and the sequential probability ratio test (SPRT) statistics, which are briefly described as follows.

Let \( x_1, x_2, \ldots \) be independent and normally distributed with mean \( \mu \) and unit variance. Testing \( H_0: \mu = \mu_0 \) against \( H_1: \mu = \mu_1 \) (assume \( \mu_0 < \mu_1 \)), the SPRT yields the stopping rule \( T = \text{first } n \) such that \( S_n = 0.5n(\mu_0 + \mu_1) \notin (a, b) = \infty \) if no such \( n \) exists, where \( S_n = \sum x_i, a = \log(A)/(\mu_1 - \mu_0), b = \log(B)/(\mu_1 - \mu_0) \).

The SPRT rejects \( H_0 \) if and only if \( S_T > b + 0.5T(\mu_0 + \mu_1) \). Testing \( H_0: x_1, x_2, \ldots, x_n \in f_0 = N(\mu_0, 1) \) against \( H_1: x_1, x_2, \ldots, x_{n-1} \in f_0 \) and \( x_n, x_{n+1}, \ldots \), \( x_n \in f_1 = N(\mu_1, 1) \), the CUSUM statistic takes the following form:

\[
\text{CUS}(n) = 0 \quad \text{CUS}(t) = \max[0, \text{CUS}(t-1) + x_t - k]
\]

where the constant \( k \) is called reference value (a function of \( \mu_0 \) and \( \mu_1 \)). Hypothesis \( H_0 \) is rejected and \( H_T \) is accepted, with the random stopping time \( T = \min\{n: \text{CUS}(n) > b\} \), where \( b \) is a test parameter constant.

2. THE MARKOV CHAIN MODEL

Note that both the SPRT and the CUSUM statistics are discrete parameter time homogeneous Markov processes due to the IID assumption on the original \( x_1, x_2, \ldots \) sequence. In any practical application the \( x_1, x_2, \ldots \) random variables are discrete as well. Hence the Markov processes referred to are further simplified to denumerable, i.e. finite or countable, Markov chains. For obvious reasons, this paper will be concerned with the finite case. In the language of Markov chains, the range of values for the CUSUM or SPRT statistics form the state space of the process. The states for an MUF based CUSUM Markov chain mode, for instance, can express the magnitude of cumulative values of unaccounted material. In a state space \( \Sigma \) of \( N \) states, state 1 can therefore represent the lowest possible running MUF value, which is always non-negative following the CUSUM test statistic rules. Likewise, state \( N \) can correspond to the highest admissible CUSUM value. In our examples we will be using \( N = 100 \), thereby indicating the percentage interpretation of the state space.

Discretized process statistics, such as SPRT or CUSUM, leads to Markov chains with transition probability matrices of special structures. The \( S_n = \sum x_i \), cumulative sum process, for example, yields a transition matrix \( P = [p_{ij}] \), with \( p_{ij} = q_{ij} \). Recall that the entries of transition matrix \( P \) are conditional probabilities defined as \( p_{ij} = \text{Prob}(x_n = j|x_{n-1} = i) \). Markov chains are, however, more general
than SPRT or CUSUM processes. This probability model has a number of other advantages as well:

(i) One can rely on a well developed Markov chain theory in computing probabilities and expected values related to the sample paths of the Markov process [3, 4].

(ii) Computation of probabilities and expected values relying on analytical means is always more accurate, efficient and satisfying than that based on Monte Carlo simulations, the primary tool for many of the statistical process control schemes.

(iii) The performance of statistical process control schemes against various modes of process changes, such as level drift, can be best accounted for by modifying the transition probability matrix of the Markov chain model rather than by investigating special process change scenarios.

(iv) A large class of derived processes based on Markov chains are Markov chains themselves with computable transition probability matrices, which makes analytical calculations possible.

Once we adapt the Markov chain probability model to our statistical process control scheme, the probability law governing the control statistics involved in the scheme will be contained in the initial probability distribution over the states and the transition probability matrix of the Markov chain.

In the sequel we denote a Markov chain process as \( y = \{y_i, i = 1,2, \ldots\} \), the associated transition probability matrix as \( P \), and initial distributions as \( v \).

3. WARNING AND ACTION LINES

There are situations in safeguards where it is costly or just not feasible to obtain frequent measurements.

Figure 1 shows a case in which the process control statistic hits the preset alarm level of 90 at about \( t = 90 \), or, equivalently, after 90 sample takings. The chart suggests that until time 60 or so the process was in control and sample taking could have been less frequent than was carried out.

In attempting to remedy the situation, one starts with a low \( f_0 \) sample rate and establishes a warning line \( W \). Once the warning line is reached, the sample taking frequency changes to a higher \( f_1 \) level. Some procedures [5] suggest using CUSUM statistics in the lower \( f_0 \) sampling regime, and SPRT in the higher \( f_1 \) sample taking region. Depending on the unfolding of the SPRT trajectory, the control process either switches back to the CUSUM mode or, by calling an alarm, concludes that there has been a change in the process level.
Figure 2 presents the variable sample rate process overlaid on the fixed sampled process of Fig. 1. Here the warning line is defined at level 80, and the action line determining the return to the lower sample rate regime is at level 75. Between level 1 and 80 the process is sampled at the rate of every fourth available sample. The variable sample process needed only 22 samples to alarm, while the original process needed around 90 samples. Thus the savings in sampling costs is almost fourfold. Notice how close the two processes are near to the alarm line level.

Next we shall generalize the variable sample rate process.
4. THE ADAPTIVE VARIABLE SAMPLING RATE (AVAS) PROCESS

The above described action and warning line scheme is a special case of the following more elaborate design. Take a collection of interval type subset \( \{V_i \subseteq \Sigma, i = 1, 2, \ldots, k\} \) in Markov chain \( y \) with state space \( \Sigma = \{1, 2, \ldots, N\} \) for some \( N \), satisfying the conditions:

(i) \( \bigcup V_i = \Sigma \)
(ii) any three \( V_i \) have empty intersection
(iii) there exist no \( V_i \) and \( V_j \) such that \( V_i \subseteq V_j \).

The \( V_i \) subsets are naturally ordered by their lower bounds, or, due to condition (iii), equivalently by their upper bounds. Each \( V_i \) is associated by a sample rate \( r_i \), by which the Markov chain is sampled when the process is in \( V_i \). The collection of triplets \( \{\{a_i, b_i, r_i\}, i = 1, 2, \ldots, k\} \), called a **sampling rule**, defines the sampling rates by which the original control statistic process is to be sampled. There is only one conflict left to be resolved. When the process \( y \) leaves a set \( V_i \), then at the next time point it might be contained in \( V_j \) and \( V_k \) for some \( j \) and \( k \), where either \( k = j + 1 \) or \( j = k + 1 \) due to the ordering on \( \{V_i, i = 1, 2, \ldots, k\} \). Let us agree on the following reasonable conflict resolution. If the process \( y \) leaves a \( V_i \) then it will enter set \( V_j \) of \( V_j \) and \( V_k \) above and will be sampled by \( r_j \), provided \( V_j \) is closer to \( V_i \) in the ordering on \( \{V_i, i = 1, 2, \ldots, k\} \). A moment’s reflection shows that with this additional conflict resolution rule the sampled process is uniquely defined. If the successive entrance times are denoted by \( \tau_n \), then a new process \( z_n = y_{\tau_n}, n = 1, 2, \ldots \) is defined as **adapted to the \( \{\{a_i, b_i, r_i\}, i = 1, 2, \ldots, k\} \) sampling rule**. It is called the **adaptive variable sampling (AVAS) process**.

![FIG. 3. Trajectory of the original process y.](image-url)
FIG. 4. Trajectories of the original process $y$ adopted to (a) rule 4; (b) rule 5; (c) rule 6.
Example 1: Consider a Markov chain with state space $\Sigma = \{1, 2, \ldots, 100\}$. Its transition probability matrix is then a 100 by 100 stochastic matrix, i.e. row sums are equal to unity. We fix the alarm hitting set as $\{90, 91, 92, \ldots, 100\}$. The transition matrix for this particular example was derived from a normal random walk with no drift and standard deviation equal to 4. Figure 3 shows a trajectory of the original process $y$. The trajectories in Figs 4(a–c) are the sampled trajectories of those of Fig. 3 as adapted to the following rules.

rule 4 = $\{\{1,80,4\}, \{70,100,1\}\}$,

rule 5 = $\{\{1,70,6\}, \{65,85,3\}, \{83,90,2\}, \{88,100,1\}\}$,

rule 6 = $\{\{1,45,10\}, \{45,70,6\}, \{65,85,3\}, \{83,90,2\}, \{88,100,1\}\}$.

Note the very different nature of the sample paths. They all, however, appear similar close to the alarm set. The number of samples for the trajectory of Fig. 3 is about 60, as opposed to the sample size of about ten when the same trajectory is adapted to rule 6.

Example 2: Next, consider a normal random walk with a drift of two units (states) in one time step and a standard deviation 3. Then rules 4–6 yield the sample paths shown in Figs 5 and 6(a–c).

It appears that the process adapted to rule 6 performs just as well as the original $y$ in detecting process shift except that it requires a great deal less sampling than does $y$. The adaptive nature of the derived AVAS process is more transparent here than in the previous example.

**FIG. 5. Trajectory of the original process $y$ with a drift of two units $+3$ SD.**
FIG. 6. Random walk with a drift of two units adapted to (a) rule 4; (b) rule 5; (c) rule 6.
5. PROBABILITY LAW FOR THE AVAS PROCESS

The AVAS process \( z_n = y_{rn} \), \( n = 1, 2, \ldots \), derived from a Markov chain \( y \), is a (time homogeneous) Markov chain itself, since it is the stopped process along a sequence of so-called \textit{optional random variables}. The derivation of its probability transition matrix is essential for calculating performance measures of an AVAS process. One could rely on simulation techniques, but these are notoriously time-consuming and carry large variations. Figures 7 and 8 compare the first and second moments of run lengths as exactly calculated and based on batches of runs of 64 sample paths to hitting the alarm set, the lower bound of which is the ordinate axis of the plots. These charts are rather convincing examples in supporting the above argument.

Suppose that Markov chain process \( y \) has a transition probability matrix \( P \) and is to be adapted to \( R = \{ V_i, r_i \}, i = 2, \ldots, k \} = \{ (a_{ij}, b_{ij}, r_i) \}, i = 1, 2, \ldots, k \}. \) Let the new process be denoted by \( y_R \) with state space \( S_R \) and transition matrix \( P_R \). The first step in constructing \( P_R \) is augmenting the original state space \( S \). Let \( \#V \) denote the number of elements in set \( V \). With this we set \( S_R = \{ 1, 2, \ldots, \#V_1 + \#V_2 + \ldots + \#V_k \}. \) Transition matrix \( P_R \) is to be composed from submatrices of \( P^{\#1}, P^{\#2}, P^{\#k} \). The actual construction is depicted in this pattern.

In Fig. 9 the striped diagonal submatrices are those of \( P^{\#1}, P^{\#2}, P^{\#k} \). The solid parts of \( P_R \) represent forbidden transitions, thus they are filled with zeros. The rest of the submatrix elements determined by the grid of subsets \( V_1, V_2, \ldots, V_k \) are filled appropriately from the powers of the transition matrix \( P \). It is easy to see that the resultant transition matrix is indeed a Markov chain matrix corresponding to state space \( S_R \), and it defines an equivalent process to \( y_R \).

Example 3: In this example we use the same Markov chain set-up as in Example 2. Applying rule 5, \( \{ (1, 70, 6), (65, 85, 3), (83, 90, 2), (88, 100, 1) \} \), to its transition matrix we obtain the new transition matrix for which the density plot is shown in Fig. 10.

The darker shades in Fig. 10 correspond to lower entries in the matrix. The largest section comprises states 1, 2, \ldots, 70. Observe that it is more dispersed than other parts of the matrix. This is readily explained by the 6th power of the original transition matrix, which corresponds to a regular Markov chain.

Example 4: Using its transition matrix the AVAS process admits the same type of calculations as its original Markov chain. Figures 11–13 present the graphs of first and second moments for average run lengths to reaching alarm levels 86, 87, \ldots, 100.
FIG. 7. Hitting time averages: simulation based and exact.

FIG. 8. Hitting time standard deviation: simulation based and exact.
\[ P_R = \]

**FIG. 9.** Diagonal submatrices.

**FIG. 10.** Density plot of new transition matrix.
FIG. 11. Calculated first and second moments for average run length to alarm rule 4 = \{1, 80, 4\}, \{70, 100, 1\}.

FIG. 12. Calculated first and second moments for average run length to alarm rule 5 = \{1, 70, 6\}, \{65, 85, 3\}, \{83, 90, 2\}, \{88, 100, 1\}.
FIG. 13. Calculated first and second moments for average run length to alarm rule $6 = \{\{1, 45, 10\}, \{45, 70, 6\}, \{65, 85, 3\}, \{83, 90, 2\}, \{88, 100, 1\}\}$. 

6. SUMMARY AND CONCLUSIONS

A flexible and cost effective statistical process control technique has been suggested in this paper. It is based on the Markov chain probability model. The main feature is a rule set comprising a collection of subsets of the underlying state space satisfying some plausible conditions. The new process is the stopped process of the original Markov chain along stopping times corresponding to first entry times to elements of the rule set. The main advantage of the scheme is its adaptive nature; it gives rise to another Markov chain in which the transition matrix can be found easily.

REFERENCES

A NEW BOOTSTRAP SCHEME
TO COMPARE CALIBRATION CURVES*

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Extended abstract

Comparing calibration curves is a quantitative or qualitative procedure whereby one can
decide whether two regression curves can be considered not different from each other. When
the procedure is sufficiently formalized it normally takes the form of a test of hypothesis.
Let $F(\theta_1)$ and $F(\theta_2)$ denote two regression curves with parameter vectors $\theta_1$ and $\theta_2$.
Sample $x_i = (x_1, x_2, ..., x_n)$ with response $y_i = (y_1, y_2, ..., y_n)$ gave rise to curve $F(\theta_i)$,
where $i = 1, 2$. We assume that both samples cover the same domain $[a, b]$ of measurement
data. By covering the same interval we mean that there is a ‘fine enough’ partition $\pi_{[a, b]}$ on
$[a, b]$ such that each member of $\pi_{[a, b]}$ contains one sample element from both samples.
In this case we say that $x_1$ and $x_2$ are the same relative to $\pi_{[a, b]}$, and we denote this relation as
$x_1 = \pi_{[a, b]} x_2$. The notion of ‘fine enough’ partition, not to be rigorously defined for the time
being, refers to an adequate number of measurement points and their spatial distribution, from
which the response curve of an instrument can be determined by using regression techniques.

Out of the two regression curves a whole range of new, so-called bootstrap curves can
be built in the following way. Take the first member of the partition $\pi_{[a, b]}$ and flip an un­
baised coin. If it is ‘heads’ then the new curve will have its first measurement point drawn
from the first data set for that partition member, otherwise from the second data set. The same
random selection mechanism will be applied for the other partition elements as well. As a
result we will end up with a mixed data set of observation and response values to
which a
regression is to be applied. Let the corresponding actual regression parameter vector be
denoted by $\theta$, where the superscript $b$ stands for bootstrap. Repeating the pseudo or boot­
strap regression curve construction in all possible manners, we form the bootstrap distribution
$\Theta_b$ of the corresponding bootstrap regression parameter vectors. There are clearly $2^n$ differ­
et bootstrap curves. The difference between $F(\theta_1)$ and $F(\theta_2)$ can now be measured by how
far the original $\theta_1$ and $\theta_2$ vectors are located from the centre of the bootstrap distribution $\Theta_b$.

The technique can easily be applied to piecewise regression curves as well, which are very
frequent in tank calibration. Next, a real life example is shown to demonstrate the performance
of this novel non-parametric technique.

* The full text of this paper was not available at the time of going to press.

1 By a partition $\pi_{[a, b]}$ of $[a, b]$ we mean a subdivision of the $[a, b]$ interval
in the following manner. $[a, b] = [d_1, d_2] \cup [d_2, d_3] \cup \ldots \cup [d_{n-1}, d_n]$, where $a = d_1 < d_2 < \ldots < d_n = b$. The sub-intervals $[d_i, d_{i+1}]$ are called the members or the elements of the
partition.
The two sets of points shown in Fig. 1 are from two calibration runs of a plutonium tank.

On visual inspection, the curves determined by the points look very much alike, yet we wish to quantify the closeness in terms of a statistic for which a distribution can be derived. Both sets of points, thirty of them, are defined in this particular example on the same mass values. For each mass value we can take a pressure value either from set No. 1 or from set No. 2. The set of all possible assignments thus contains $2^{30} = 1.073741824 \times 10^9$ pseudo or bootstrap curves. By fitting a quadratic regression curve for each member of this set we obtain three distributions for the three parameters of the regression. These are called the bootstrap distributions and are presented in Fig. 2.

In each histogram there is a pair of arrows shown as pointing to the positions of the actual coefficients of the regression curves determined by the original curves. These positions on the bootstrap distributions serve as the measure of closeness of the curves. Formal test hypotheses can be based on them, readily calculating the size of the test. Attempts are being made to calculate the power of this testing procedure and also comparisons may be made with classical methods such as analysis of covariance.
FIG. 2. Histograms of regression parameters.
SAFEGUARDS APPLICATIONS OF PATTERN RECOGNITION AND NEURAL NETWORKS*

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Abstract

SAFEGUARDS APPLICATIONS OF PATTERN RECOGNITION AND NEURAL NETWORKS.

The number, complexity and throughput of nuclear materials is increasing at nuclear facilities all over the world. It is becoming necessary to invest less inspector time per facility without losing safeguards effectiveness. Continuous unattended radiation and surveillance monitoring systems significantly reduce inspector time in facilities. However, continuous measurement systems produce large safeguards databases and require inspector time for thorough review and analysis. Technology based on neural network pattern recognition software has been developed to automate the review and analysis of safeguards databases. The overall effectiveness of safeguards evaluations is enhanced through on-line integration of video digital data, radiation monitoring and other sensor data. Neural network pattern recognition analysis of these integrated safeguards data sets increases the evaluation effectiveness by showing trends, discovering anomalies, and highlighting specific activities for detailed review by inspectors.

1. INTRODUCTION

Large quantities of safeguards data are automatically collected by continuous unattended monitoring instruments used in nuclear chemical processing plants, nuclear material storage facilities, fuel fabrication facilities and on-load nuclear reactors. With complex and diverse data, it is time consuming and demanding for safeguards inspectors to examine effectively all the data for consistency and to find subtle anomalies that could be caused by diversion. When normal trends and patterns in the data can be characterized, such anomalies reveal themselves. This radiation signature forms a ‘fingerprint’ that can signal inappropriate activity.

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Software that can recognize patterns through neural networks can be an efficient aid to inspectors. The software can analyse all data and provide interpretations, predict trends and identify anomalies. Using these computing and analysis techniques, we can thoroughly analyse large volumes of data to provide inspectors with information that allows them to focus on anomalies and on the data of interest for effective safeguards, thereby isolating the 'needle in the haystack'. Systems utilizing these techniques must be capable of classifying, clustering and recognizing features within the data; extrapolating features from the data; checking for proper operation of the systems; and detecting anomalies. The potential benefits of this intelligent software include:

- Correlating large quantities of diverse information;
- Identifying abnormal activity, such as diversion of material or faulty data from intermittent sensors;
- Developing inferences and conclusions;
- Increasing inspector efficiency by providing routine analysis of all data, identifying specific trends and flagging data to be checked by an inspector, and relieving the tedium of reviewing masses of information;
- Automatically verifying plant operation.

In this paper we describe several projects using pattern recognition software that interpret data and detect anomalies. In each case, existing systems for collecting safeguards data are used to gather data for neural network analysis. The aim of this analysis is to characterize a large volume of correlated data to discover patterns that enable us to distinguish normal activity from abnormal activity.

2. PATTERN RECOGNITION USING NEURAL NETWORKS

A neural network is an iterative numerical technique that facilitates the solution of a number of different types of problems including pattern recognition and categorization of data. A neural network learns to recognize patterns by repeatedly examining examples of those patterns [1]. For example, a network could be trained to learn to recognize radiation signatures and video images. The name neural network is used because these networks are similar to biological neurons and their connections. Neural networks have attracted attention recently, and now that hardware implementations are available they have greater appeal for data acquisition and control systems. We have demonstrated the usefulness of neural network pattern recognition for a variety of safeguards applications [2].

The basic pieces of a neural network are layers (usually three or more), consisting of several neurons or nodes connected by adjustable weights. The weights are selected such that for a given pattern of input values, a particular output value will be chosen. The most commonly used neural network paradigm is the back-
propagation algorithm. The network is developed through a training step and a recall step. During the training step, examples of input vectors are presented to the network together with a desired output. The actual output is computed and compared with the desired result. Any difference between the two is a measure of the error in the chosen weights. This error is then ‘back-propagated’ through the network (fed through the layers in reverse order) and used to update the weights. This type of learning is called ‘supervised’ because the desired-result value controls and directs the process. Learning without such a control is called ‘unsupervised’.

During the testing step, new vectors that have not been used for training are presented to the network. The accuracy of classification of these new testing phase vectors provides a measure of how well the network was trained. The back-propagation paradigm has some limitations when it is used for categorization. It tends to place an anomalous item into the category it fits best without recognizing its anomalous nature. It also does not give a measure of confidence with which the categorization was made.

The neural network self-organizing map (SOM) [3] is one methodology that addresses these issues by creating a two dimensional feature map of the input data. If two input vectors are close in some sense, they will be placed in the same area of the map. A key difference between the SOM and many other networks is that the SOM learns relationships among data elements without supervision (without the use of a desired output value), hence the term self-organizing. A hybrid of SOM and back-propagation gives the best results for anomaly detection.

3. SPENT FUEL MONITORING

Effective safeguards requires monitoring the reactor refuelling process and tracking spent fuel to ensure that it is not diverted. On-load reactors are refuelled at power, preventing inspectors from visually observing the core during the refuelling process as is done with light water reactors. On-load refuelling consists of pushing fresh fuel bundles in at one side and discharging spent fuel bundles out at the opposite side of the core. The on-load refuelling operation can be monitored with a continuous unattended core discharge monitoring (CDM) system that detects radiation signals from spent fuel bundles as they are discharged from the core [4]. Continuous unattended monitoring of on-load reactors and of spent fuel movements produces large volumes of data that must be reviewed to evaluate and account for refuelling and to verify fuel movements.

Figure 1(a) is a time series plot showing data from two detector channels that are monitoring the core of an on-load reactor. Each large spike on the time series graph corresponds to a pair of fuel bundles being discharged from the reactor. An inspector using a graphical review programme must locate and count each of the spikes to determine the total number of spent fuel bundles discharged from the
FIG. 1. (a) Time series radiation monitoring of an on-load reactor, and (b) spent fuel burnup from an on-load reactor.
reactor. This number is then compared with that given in the facility declaration of spent fuel discharges for safeguards verification. The safeguards inspector can benefit from modern, high performance computers and software tools that automatically analyse and review unattended monitoring data.

We have developed a prototype neural network pattern recognition analysis program, CDM Analysis, to automate the review of CDM data. CDM Analysis significantly reduces the time and tedium required to manually review continuous unattended monitoring data and provides additional information that is not available through manual reviews [5]. The CDM Analysis tool

- Counts fuel bundle discharges and records the time when they occurred,
- Identifies sections in the CDM data for an inspector to examine,
- Indicates the reactor power level,
- Correlates events between detector channels and checks for monitoring problems,
- Identifies the channel from which the spent fuel was discharged, and
- Predicts the burnup of discharged fuel bundles.

Determining burnup is a difficult, multivariate problem requiring a large data set. For this study, the total amount of data available (30 days) was sparse and fuel burnup fell into one of four distinct regions. We built the neural network to examine CDM data and to classify each of the discharged fuel events into one of the four burnup categories, as shown in Fig. 1(b). Data from neutron monitoring channels were used to train the neural network to predict fuel burnup. In spite of the extremely small data set, the network correctly classified the burnups of discharged fuel into one of the four regions with an accuracy of 92%.

The CDM Analysis tool has shown the potential for automated analysis of CDM data to determine the refuelling activity and to monitor the reactor power level. Neural network implementations for determining the location of fuel discharge and the burnup of fuel bundles appear successful enough to warrant further research. It appears that neural network models could be developed to provide almost 100% accuracy in determining the position and burnup category if a more complete set of representative data from an operating on-load reactor were available.

4. PROCESS MATERIAL TRANSFERS

Another pattern recognition project on which we are working involves material transfers in a chemical processing plant that follow very specific patterns of activity as liquid is moved from one tank to another and valves are opened and closed. Transfers occur in a small number of patterns and leave the system in specific ways. Because of the complexity of the processes and the large and diverse amount of data, efficient automatic algorithms are necessary to interpret the data and improve
safeguards evaluations [6]. Success for many applications may also rely on data representation (preprocessing and encoding). A neural network analysis and prediction model can be adapted [7, 8] to provide continuous unattended monitoring of material flow in a plant, to identify anomalous transfers and to predict material loss.

The process material transfer project is aimed at improving anomaly detection and thereby reducing inventory frequency, enhancing assurance of physical inventory and gaining a better perspective on overall facility operations. Neural networks are versatile in modelling the input–output relationships of complex non-linear systems. For this reason, this technique has been proposed to provide support for nuclear power plant operations [9–11]. Our experiments with neural networks on data from a process monitoring system indicate that a neural network based module may provide an anomaly detection system as an adjunct to a material control and accounting system. If a trained network represents a good model of normal plant operation, it can be a reliable tool for recognizing non-normal activity. The neural network will then be able to flag any anomalous facility state, such as loss or diversion of material, instrument failures, or other abnormal or unusual events. This would improve the sensitivity of any existing system and reduce inspector effort.

Modern safeguards systems for nuclear materials handling typically use distributed control systems that record and store information in large databases [12]. From these databases, we are using characteristics of the physical system to predict future correlated characteristics. By comparing predicted states with states reported by a control system, we can detect anomalies. Recently, we used raw data obtained from the process monitoring computer system (PMCS) at the Idaho Chemical Processing Plant (ICPP) [13]. Uranium fuel processing at ICPP consists of a number of head-end dissolution processes, one cycle of tributyl phosphate solvent extraction, two cycles of hexone extraction, and one denitration step, to end with uranium trioxide as a product. The PMCS acquires process data from plant instruments and sensors (valves, pumps and steam jets). This database is used by safeguards personnel to monitor solution movements to and from the measurement vessels. These vessels are positioned so that all input and output streams can be measured. Data from the four inventory vessels were the focus of this project. Figure 2 shows the flow of material into and out of the inventory vessels.

Two sets of process data were available: campaign 1 and campaign 2. We developed software to preprocess the data, producing snapshots of the facility at four-minute intervals. Our input to the neural network consisted of a window of intervals taken from this data set. Because of the large number of sensors (144), it was important to incorporate in our model only those that were correlated, keeping the problem as small as possible without loss of information. There were a number of normal transactions and no known anomalies in these data. The neural network model correctly predicted the volume in 85–90% of the transactions.

We believe that the best approach is to work with an expert in plant operation to perform appropriate feature selection, data aggregation, data fusion, concept
extraction and data correction. Further efforts in the areas of architecture design and data preprocessing should lead to neural networks that can accurately model processes and detect anomalies.

5. INTEGRATION OF VIDEO AND RADIATION ANALYSIS DATA

For the past several years, the integration of containment and surveillance (C/S) with non-destructive assay (NDA) sensors for monitoring the movement of nuclear material has focused on the hardware and communications protocols in the transmission network. Not much progress has been made in using the combined C/S and NDA data for safeguards and reducing the time spent by inspectors in nuclear facilities. One of the fundamental problems in integrating the combined C/S and NDA data is that the two methods operate in different dimensions. The C/S video data are spatial, whereas the NDA sensors provide radiation levels versus time, i.e. the problem is to integrate spatial data (C/S) with radiation time data (NDA).

We have developed a new method to facilitate this integration of spatial and radiation time information, i.e. to transform the video spatial data into the time domain as a function of physical motion in the video data. We call the method video time and radiation analysis program (VTRAP). Every 1–2 seconds a video picture of the area of interest is compared with the baseline picture, and the quantitative amount of change or movement is converted to a single pixel difference number.
This provides a data compression of $\sim 10^5$ to $10^6$ and allows a comparison in time with the radiation sensors that are synchronized with the video motion data. More sophisticated versions of the VTRAP concept split the video signal into several regions of interest, such as a vault door and a vault area, and the motion within each region is separately digitized and read out.

The interplay between the multiple regions of interest in the video motion data and the multiple radiation sensors as a function of time becomes very complex. We have introduced neural network analyses to evaluate the data to distinguish abnormal events from normal activities. The neural network methods can be automated to handle the large quantities of data that are generated with frequent collection periods (1–5 s). The software can easily compress the data in time periods when there are no motion or radiation changes. When there are physical motion or radiation changes, the digital picture can be saved for later review by inspectors.

The short time intervals (1–5 s) give a continuity of knowledge that is not possible with normal collection of video data. In addition, the essentially continuous data collection avoids the problem of missing a crucial trigger event.

To develop the VTRAP concept, we have set up combined video and radiation monitors in a nuclear material laboratory at the Los Alamos National Laboratory. The NDA radiation sensors include a neutron coincidence counter and a compact neutron detector positioned on the wall of the room. Each of these NDA sensors has its own electronics, a local computer, and data collection software. The video camera is connected to a Sun Sparcstation to store and analyse data. The pixel difference numbers versus time are stored as well as full video images when present thresholds are exceeded.

To train the neural network, we have defined a set of normal activities in the room where nuclear material is removed from storage and placed in the neutron coincidence counter for measurement and then returned to storage. The normal activities also include movement of people and equipment in the room with no movement of nuclear material. After the normal activity training period was completed, we provided the sensors with abnormal activities, such as removal of a radiation source from the storage vault and no returning of it to the vault.

Typically, there would be multiple (3–10) sets of video and radiation sensor data synchronized in time with coupled information related to the location, direction and speed of motion of the nuclear material. Because of the complex relationships in the multiple integrated data sets, neural networks can be used effectively to evaluate the data. During times of abnormal activity, full (or compressed) video images are saved. Each sensor would retain data in its local computer system to provide redundancy for a prescribed time interval.

The VTRAP approach is well suited for large automated facilities where the movement of nuclear material is restricted by the constraints in the robotics system. The VTRAP would also find applications at reactors, spent fuel storage facilities, reprocessing plants and nuclear material storage vaults.
We have developed software to automatically review and analyse safeguards data from continuous unattended monitoring systems. This technology is based on pattern recognition by neural networks, provides significant capability to analyse complex data, and has the ability to learn and adapt to changing situations. Recent demonstrations of prototype software tools have shown the ability to accurately determine fuel discharges from on-load reactors, determine reactor power levels and predict fuel burnup. The application of neural network analysis to chemical processing plant data provides the capability to detect material transfer anomalies as valves are opened and closed and liquid is moved from one tank to another.

We have introduced a new method to integrate spatial (digital video) information with time (radiation monitoring) information. This method involves transforming a video image to a single pixel difference on the basis of the amount of change between video frames and allows a comparison in time with the radiation sensors that are synchronized with the video motion data. The compression of the data is enhanced by the interplay between the synchronized video and NDA data, especially in facilities with activity that is unrelated to nuclear material movement. The essentially continuous time information provided by the VTRAP approach using multiple and redundant sensors gives the potential of significantly reducing the frequency of inspection visits to key facilities without a loss of safeguards effectiveness.

REFERENCES


Abstract

Since the last IAEA International Safeguards Symposium, a revised standard for volume calibration methodology has been issued in the United States of America. Because the new standard reflects the advent of high precision volume measurement technology, it is significantly different from the earlier standard, which it supersedes. The new standard outlines a unified data standardization model that applies to process tanks equipped with differential pressure measurement systems for determining liquid content. At the heart of the model is an algorithm to determine liquid height from pressure measurements that accounts for the major factors affecting the accuracy of those measurements. The standardization model also contains algorithms that adjust data from volumetric and gravimetric provers to a standard set of reference conditions. A key component of the standardization model is an algorithm to take account of temperature induced dimensional changes in the tank. Improved methods for the statistical treatment of calibration data have also appeared since the last Safeguards Symposium. A statistical method of alignment has been introduced that employs a least-squares criterion to determine 'optimal' alignment factors. More importantly, a statistical model has been proposed that yields plausible estimates of the variance of height and volume measurements when significant run-to-run differences are present in the calibration data. The new standardization model and statistical methods described in the paper are being implemented in a portable, user friendly software program for use by IAEA inspectors and statisticians. Perhaps these methods will eventually find their way into appropriate international standards.

1. INTRODUCTION

Reliable measurements of liquid volume are a vital component of any system for the control and accountability of materials in a nuclear processing facility. In most facilities, tanks are equipped with differential pressure measurement systems...
to determine liquid content. Volumes are then determined from a measurement equation that relates the pressure produced by a given quantity of tank liquid to its volume. Tank calibration refers to the process required to estimate a tank's measurement equation. A typical tank calibration exercise involves collecting data from several calibration runs. Each run consists of adding a series of carefully measured increments of some liquid of known density to the tank and recording the corresponding measurement system response (pressure) for each. The resulting data are used to estimate the tank's measurement equation.

Many factors can adversely affect the measurements obtained during a calibration run. Of these, temperature is the most important. When factors such as temperature change during a calibration, the data must be adjusted to a standard set of reference values. Procedures for data standardization are presented in a document of the American National Standards Institute, ANSI N15.19-1989 [1].

The purpose of this paper is to outline the key features of the standardization model in ANSI N15.19-1989, together with methodological improvements that have occurred since 1989. The underlying calibration model is presented in Section 2. The data standardization model is presented in Section 3. Its essential components are an equation for determining liquid height from pressure, algorithms for thermal adjustments to volumetric measurements, and a model that deals with dimensional changes in the tank. Statistical methods are presented in Section 4, where the emphasis is on a model that yields realistic estimates of variance for calibration measurements when the calibration data exhibit significant run-to-run differences. Some concluding remarks are given in Section 5.

2. THE VOLUME CALIBRATION MODEL

The calibration equation for a process tank expresses the height of liquid in the tank as a function of its volume. The measurement equation, which is the inverse of the calibration equation, gives the volume of the tank as a function of elevation relative to a specified reference point. At a fixed reference temperature $T_r$, the measurement equation can be written as

$$V_r = f^{-1}(H_r) = \int_{-\epsilon}^{H_r} A_r(H) \, dH \quad (1a)$$

where $A_r(H)$ is the cross-sectional area of the tank at elevation $H$ above the primary reference point, and $\epsilon$ is the vertical distance between the primary reference point and the lowest point in the bank.\(^1\)

\(^1\) The nomenclature in Sections 2 and 3 conforms to that in Ref. [1].
Unless a tank can be completely emptied, a calibration begins at some unknown elevation $H_0 > -\varepsilon$. Equation (1a) can be rewritten in terms of $V_0 = V_r(H_0)$, the volume of the tank below $H_0$:

$$V_r = V_0 + \int_{H_0}^{H_r} A_r(H) \, dH$$

(2)

The heel volume $V_0$ cannot be determined by the tank's measurement system. In particular, $V_0$ cannot be determined by calibration, but must be determined by some method independent of the tank's measurement system.

The purpose of the tank calibration exercise is to determine the functional relationship between volume and height given by Eq. (1a). This is done by estimating the calibration function

$$H_r = f(V_r)$$

(3)

and inverting the resulting estimate to obtain an estimate of

$$V_r = f^{-1}(H_r)$$

(1b)

3. DATA STANDARDIZATION MODEL

Data standardization refers to the steps taken to adjust the calibration data to a predetermined set of standard reference conditions. Key components of the data standardization model are discussed in this section.

3.1. Liquid height

The majority of the tanks in nuclear processing facilities are equipped with differential pressure measurement systems to determine liquid content. The fundamental relationship for determining liquid height from pressure is

$$g \rho H_1 = P_1(0) - P_3(H_1)$$

(4)

where $g$ is the local acceleration due to gravity, $\rho$ is the average density of liquid in the tank, $P_1(H)$ is the pressure in the major probe at an elevation $H$ above the primary reference point, and $P_3(H)$ is the pressure in the reference (vapour head) probe at an elevation $H$ above the primary reference point.

In practice, the differential pressure

$$\Delta P_1 = P_1(E_1) - P_3(E_1)$$

(5)
is measured by a gauge at a convenient elevation $E_1$ above the primary reference point and is used to approximate the right hand side of Eq. (4). Differences between $\Delta P_1$ and $P_1(0) - P_3(H_1)$ may result from several factors, which include: (i) the density of gas in the pneumatic lines, (ii) differences in flow resistance in the pneumatic lines, (iii) differences in the density of air in the reference pressure line and in the vapour head, and (iv) surface tension and pressure effects associated with the formation of bubbles at the tip of the major probe. When all of these factors are taken into account, the difference between $\Delta P_1$ and $P_1(0) - P_3(H_1)$ is given by

$$[P_1(0) - P_3(H_1)] = \Delta P_1$$

$$= gE_1(\rho_1^a - \rho_s^a) + gH_1\rho_s^a - gE_3(\rho_3^a - \rho_s^a) + (\delta_3 - \delta_1) - 2\gamma/r_b \quad (6)$$

where

- $\rho_1^a$ is the average density of air in the major probe line,
- $\rho_3^a$ is the average density of air in the reference probe line,
- $\rho_s^a$ is the average density of air in the tank above the liquid surface,
- $\delta_1$ is the pressure drop in the major probe line (due to the gas flow resistance),
- $\delta_3$ is the pressure drop in the reference probe line,
- $E_3$ is the elevation of the pressure gauge above the tip of the reference probe,
- $E_1$ is the elevation of the pressure gauge above the primary reference point,
- $\gamma$ is the surface tension for the liquid and air,
- $r_b$ is the radius of curvature of the bubble at its lowest point.

The terms on the right hand side of Eq. (6) are discussed in greater detail in Ref. [1].

From Eqs (4) and (6) it is possible to derive the following expression for the height of liquid $H_1$ in the tank:

$$H_1 = [\Delta P_1 + gE_1(\rho_1^a - \rho_s^a) - gE_3(\rho_3^a - \rho_s^a) + (\delta_3 - \delta_1) - 2\gamma/r_b]/[\rho - \rho_s^a] \quad (7a)$$

In practice, differences between $\Delta P_1$ and $P_1(0) - P_3(H_1)$ may often be ignored, in which case Eq. (7a) reduces to

$$H_1 = \Delta P_1/(g\rho) \quad (7b)$$

If a tank contains more than one probe, the equations given here can be used to compute the height of liquid above the tip of each probe. Many of the quantities in Eqs (6) and (7) are temperature dependent. Consequently, these equations should
not be used to infer the height of liquid at one temperature directly from its height at some other temperature. This procedure can produce erroneous results, especially for tanks with large changes in cross-sectional area or large heel volumes.

3.2. Thermal adjustments

Of all the factors that affect the quality of calibration measurements, temperature changes have the greatest effect. The data standardization model includes thermal adjustments for the height of liquid in the tank, the prover data, and dimensional changes in the tank.

3.2.1. Volumetric provers

Temperature differences affect the volume of liquid delivered by a prover; standardization is required when differences during a calibration are significant (e.g. more than 3°C).

Let \( v_c \) denote the calibrated volume of the prover at the temperature \( t_c \). Then the volume of liquid delivered by the prover at temperature \( t_m \) is

\[
v_m = v_c (1 + 3\beta \Delta t_m)
\]

where \( \Delta t_m = t_m - t_c \) and \( \beta \) is the (linear) coefficient of thermal expansion of the prover. The mass of the liquid delivered is estimated by

\[
\hat{m} = v_m \rho_m
\]  \hspace{1cm} (8a)

where \( \rho_m \) is the density of the calibration liquid at temperature \( t_m \).

3.2.2. Gravimetric provers

Weight measurements are essentially independent of temperature. Thus, the mass of liquid delivered by a gravimetric prover is estimated by

\[
\hat{m} = w + \delta
\]  \hspace{1cm} (8b)

where \( w \) is the measured weight of the volume delivered and \( \delta \) is a correction for the buoyancy of air. Depending on whether a prover is volumetric or gravimetric, either Eq. (8a) or Eq. (8b) is used in subsequent data standardization calculations.

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2 Prover temperatures are designated by lower case \( t \) to emphasize that they may differ from the tank temperatures, designated by upper case \( T \).
3.2.3. Dimensional changes in the tank

From Eq. (1b), the measurement function valid at temperature $T_r$, which gives the volume of the tank below some point at elevation $H_r$ relative to the primary reference point, is

$$V_r = f^{-1}(H_r)$$

If the temperature of the tank now changes to $T_m$, then the volume of the tank below the indicated point changes to

$$V_m = V_r(1 + 3\alpha \Delta T_m)$$  \hspace{1cm} (9)

and the height of the indicated point, relative to the primary reference point, changes to

$$H_m = H_r(1 + \alpha \Delta T_m)$$  \hspace{1cm} (10)

where $\alpha$ is the coefficient of thermal expansion for the tank and its probes, and $\Delta T_m = T_m - T_r$. When the expressions for $V_r$ and $H_r$ given by Eqs (9) and (10) are substituted into Eq. (1b), the following form of the measurement equation is obtained:

$$V_m/(1 + 3\alpha \Delta T_m) = f^{-1}[H_m/(1 + \alpha \Delta T_m)]$$  \hspace{1cm} (11)

The corresponding form of the calibration equation is

$$H_m/(1 + \alpha \Delta T_m) = f[V_m/(1 + 3\alpha \Delta T_m)]$$  \hspace{1cm} (12)

Equations (11) and (12) define the adjustments to liquid volume and height measurements that compensate for dimensional changes in the tank induced by temperature.

3.3. Standardized calibration data

Standardized calibration data are obtained by applying the appropriate standardization steps of the preceding section to the data from each increment of each calibration run.

3.3.1. Liquid height

Liquid heights are determined from pressure measurements made at some temperature $T_m$ by means of Eq. (7a) or Eq. (7b), as appropriate. If $T_m$ differs from the
reference temperature $T_r$, then Eq. (10) is used to adjust the heights to compensate for dimensional changes in the tank. The standardized liquid height is given by

$$ Y = H_r = H_m/(1 + \alpha \Delta T_m) $$

Equation (13) is applied to each increment $i$ of a calibration run.

### 3.3.2. Volume

For each increment $i$, the cumulative volume of liquid delivered to the tank by the prover is determined as follows: The cumulative mass of all previous calibration increments of the current run is divided by the density of the calibration liquid $\rho_{mi}$ at the measured temperature $T_{mi}$ of the current increment. Thus,

$$ V_{mi} = \frac{\sum_{j<i} \dot{m}_j/\rho_{mi}}{\sum_{j<i} \dot{M}_j/\rho_{mi}} = \frac{\dot{M}_i/\rho_{mi}}{\sum_{j<i} \dot{M}_i/\rho_{mi}} $$

For each increment, the quantities $\dot{m}_j$ are determined from either Eq. (8a) or Eq. (8b), as appropriate.

If $T_{mi}$ differs from the reference temperature $T_r$, then Eq. (9) is used to adjust the volumes to compensate for dimensional changes in the tank. The standardized cumulative volume of the $i$-th calibration increment is given by

$$ X_i = V_{ri} = V_{mi}/(1 + 3\alpha \Delta T_{mi}) $$

If necessary, the data from several calibration runs can be aligned at this point. A statistical procedure for alignment is described in the following section. An estimate of the heel volume can then be added to each of the (aligned) standardized cumulative volume measurements to complete the data standardization process. The resulting data can then be used to estimate the calibration function (Eq. (3)) or its inverse at the reference conditions.

### 3.4. Run alignment

A statistical procedure for run alignment is based on the following model:

$$ Y_{ij} = \alpha + \beta(X_{ij} + \delta_j) + \epsilon_{ij} $$

where $Y_{ij}$ and $X_{ij}$ denote, respectively, the standardized height and cumulative volume measurements (see Section 3.3) from the $i$-th increment of an initial segment.

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3 A method of estimating the heel volume of a tank is given in Section 5.2 of Ref. [1].
of the \( j \)-th calibration run (where \( i = 1, 2, \ldots, n; \) and \( j = 1, 2, \ldots, r \)). The quantity \( \delta_j \) represents the amount by which the cumulative volumes of the \( j \)-th run must be adjusted to bring the run into alignment. The quantity \( \epsilon_{ij} \) represents the measurement error. Data used for alignment should be selected from an initial segment in which the calibration function is reasonably linear.

The quantities \( \delta_j \) are obtained by minimizing the quantity

\[
Q = \sum_{i} \sum_{j} [Y_{ij} - (\alpha - \beta (X_{ij} + \delta_j))]^2
\]

with respect to \( \alpha, \beta \) and \( \delta_j \). The values \( \alpha^*, \beta^* \) and \( \delta_j^* \) that minimize \( Q \) are obtained by solving the following equations:

\[
\alpha = \bar{Y} - \beta \bar{X}
\]

\[
\beta = \frac{\sum \sum Y_{ij}(X_{ij} + \delta_j) - nr\bar{Y}\bar{X}}{\sum \sum (X_{ij} + \delta_j)^2 - nr\bar{X}^2}
\]

\[
\delta_j = \beta^{-1}(\bar{Y}_j - \alpha) - \bar{X}_j
\]

Although these least-squares equations must be solved iteratively, this can usually be accomplished in two to four steps.

The alignment method presented here is an extension of that given in Appendix B of Ref. [1]. The method presented here is superior unless \( \alpha = 0 \), in which case the two methods yield identical results.

4. STATISTICAL ANALYSIS

The calibration (or measurement) function can now be estimated from the (aligned) standardized calibration data. The main challenge is to provide realistic estimates of uncertainty for liquid height and volume measurements.

4.1. Estimation of the calibration and measurement functions

Many statistical methods are available to estimate either the calibration function or the measurement function. The choice depends largely on the nature of measurement errors in the calibration and tank measurement systems. The literature on
this topic is extensive and estimates of height and volume are relatively insensitive to the choice of a specific model. We therefore focus on one method that yields suitable variance estimates for height and volume measurements.

4.2. Variance estimates for liquid height measurements

Tank calibration data often exhibit significant run-to-run variability. An ad hoc procedure that produces realistic estimates for such cases was presented in Ref. [2] and later in Ref. [1]. However, this procedure lacks a firm theoretical foundation. A more recent model [3] provides a sound theoretical basis for making realistic variance estimates and is applicable whenever the calibration function, Eq. (3), is piecewise linear. The estimation procedure is outlined here; additional details and examples are given in Ref. [3].

The model is expressed as follows:

$$Y_{ij} = \beta_0 + \gamma_j + (\beta_1 + \phi_j)x_{ij} + \epsilon_{ij}$$

where $Y_{ij}$ and $x_{ij}$ denote, respectively, the standardized height and cumulative volume for the $i$-th increment of the $j$-th calibration run (for $i = 1, 2, \ldots, n$ and $j = 1, 2, \ldots, r$). The quantities $\beta_0$ and $\beta_1$ denote the intercept and slope of the underlying calibration function, respectively. The variable $\gamma_j$ represents the deviation of the intercept of the $j$-th run from its true value $\beta_0$; similarly, $\phi_j$ represents the deviation of the slope of the $j$-th run from its true value $\beta_1$. The residual error of the $i$-th increment of the $j$-th run is denoted by $\epsilon_{ij}$.

The variables $\gamma_j$, $\phi_j$ and $\epsilon_{ij}$ are all assumed to be mutually independent, both between runs and within runs. Moreover, $\gamma_j$, $\phi_j$ and $\epsilon_{ij}$ are assumed to be distributed as normal (Gaussian) random variables, each with mean 0, and with variances $\sigma^2$, $\sigma^2$, and $\sigma^2$, respectively.

It is assumed for computational simplicity that the cumulative volume of the $i$-th calibration increment is the same for all runs. In symbols, this means that

$$x_{ij} = x_i \text{ for } j = 1, 2, \ldots, r$$

Variance estimates for liquid heights calculated from Eq. (16) are based on the sums of squares (SSE) calculated under various assumptions about the variables $\gamma_j$ and $\phi_j$. Specifically, the quantity

$$\text{SSE} = \sum_i \sum_j (\epsilon_{ij})^2$$

In this section, lower case $x$ is used to denote the selection of volume as the control variable.
is computed for the following three cases:

**Case 1:** \( \phi_j \neq 0 \) and \( \tau_j \neq 0 \). Estimates of the intercept and slope are computed individually from the data of each calibration run (one equation for each run, different slopes and different intercepts).

**Case 2:** \( \phi_j = 0 \) and \( \tau_j \neq 0 \). Data from all calibration runs are pooled to estimate a common slope, but intercepts are estimated individually for each run (one equation for each run, a common slope and different intercepts).

**Case 3:** \( \phi_j = 0 \) and \( \tau_j = 0 \). Data from all calibration runs are pooled to estimate a single equation (with a common slope and a common intercept).

Now let \( SSE(1), SSE(2) \) and \( SSE(3) \) denote the values of Eq. (17) for Cases 1, 2 and 3, respectively. Clearly, \( SSE(1) < SSE(2) < SSE(3) \). Under the assumptions of the model, it can be shown that

\[
E[SSE(3) - SSE(2)] = n(r - 1) \left[ \sigma^2 + \bar{x}_j^2 \sigma^2 + \frac{\sigma^2}{n} \right]
\]

(18)

\[
E[SSE(2) - SSE(1)] = (r - 1) \left[ \sigma^2 \sum_i (x_i - \bar{x}_i)^2 + \sigma^2 \right]
\]

(19)

In all cases, \( SSE(1) \), the Case 1 value of Eq. (17), yields a good estimate of the residual measurement error variance \( \sigma^2 \):

\[
\hat{\sigma}^2 = \frac{SSE(1)}{[r(n - 2)]} \quad (20)
\]

Now, if \( \hat{\sigma}^2 \) is taken as an estimator of \( \sigma^2 \), Eqs (18) and (19) yield the following estimates of \( \sigma^2 \) and \( \sigma^2 \):

\[
\hat{\sigma}^2 = \left( \frac{SSE(2) - SSE(1)}{r - 1} - \hat{\sigma}^2 \right) \sum_i (x_i - \bar{x}_i)^2 \quad (21)
\]

\[
\hat{\sigma}^2 = \frac{SSE(3) - SSE(2)}{n(r - 1)} - \bar{x}_j^2 \hat{\sigma}^2 - \frac{\hat{\sigma}^2}{n} \quad (22)
\]

In Case 3, the least-squares estimators \( (b_1 \) and \( b_0) \) for \( \beta_1 \) and \( \beta_0 \) are, respectively,

\[
b_1 = \frac{\sum_i (x_i - \bar{x}_i)(\bar{Y}_i - \bar{Y}_..)}{\sum_i (x_i - \bar{x}_i)^2}
\]

\[
b_0 = \bar{Y}_.. - b_1 \bar{x}.
\]

\(^5\) Explicit formulas for \( SSE(1), SSE(2) \) and \( SSE(3) \) are given in Ref. [3].
The Case 3 estimates of $\beta_1$ and $\beta_0$ are used for prediction. Thus, for any volume $x_0$, the corresponding estimate of liquid height is

$$\hat{Y}_0 = \hat{Y}|x_0 = b_0 + b_1 x_0$$

(23)

Under the given model, the variance of the predicted (mean) height measurement $\hat{Y}_0$ that corresponds to $x_0$ is

$$\sigma^2(\hat{Y}_0) = \frac{1}{n} \left\{ \sigma^2 + x_0^2 \sigma^2_{\beta} + \sigma^2 \left[ \frac{1}{n} + \frac{(x_0 - \bar{x})^2}{\sum_i (x_i - \bar{x})^2} \right] \right\}$$

(24)

With the aid of Eqs (21) and (22), an estimator of $\sigma^2(\hat{Y}_0)$ can be written in the form

$$\sigma^2(\hat{Y}_0) = \sigma^2 + x_0^2 \sigma^2_{\beta} + \sigma^2 \left[ \frac{1}{r} + \frac{(x_0 - \bar{x})^2}{\sum_i (x_i - \bar{x})^2} \right]$$

(25)

The final step is to determine the degrees of freedom for the estimator in Eq. (25). A simple approximate way to do this is to assign $r(n - 2)$ degrees of freedom to $\sigma^2$ and $(r - 1)$ degrees of freedom to each of $\sigma^2_{\beta}$ and $\sigma^2_{\phi}$, and then apply the Welch-Satterthwaite formula. A more precise (and more complicated) algorithm is given in Ref. [3]. After the degrees of freedom have been calculated, Eq. (25) can be used to compute confidence intervals for the predicted (mean) height corresponding to a given volume $x_0$.

For a given volume $x_0$, the variance of a new (future) individual predicted liquid height measurement $Y_p$ is

$$\sigma^2(Y_p|x_0) = \sigma^2(\hat{Y}_0) + \sigma^2_{\varepsilon}$$

(26)

The first term on the right hand side of Eq. (26) is the systematic component of variance given by Eq. (24). The second term is the random component of variance for the new prediction. This term should reflect the variability of the system used to make the new predicted measurement of liquid height. If the new measurement is to be made with the same system as that used for calibration, then a suitable estimator of $\sigma^2_{\varepsilon}$ is

$$\sigma^2_{\varepsilon} = \sigma^2_{\epsilon} + x_0^2 \sigma^2_{\beta} + \sigma^2_{\phi}$$

(27)

If the new measurement is made with some other measurement system, then $\sigma^2_{\epsilon}$ should be replaced by an estimator that reflects the variability of the system being used.
If Eq. (23) is used for estimating the volume \( x_0 \) that corresponds to a given liquid height \( Y_0 \) (as is frequently the case), then a slightly different procedure is required. First, the variance of the new measurement is estimated. This estimate can then be used in connection with Eq. (24) to compute a suitable discrimination interval for \( x_0 \). One method of combining these estimates is given in Appendix B of Ref. [1]. When the measurement and calibration systems are the same, Eqs (24) and (26) are identical. More detailed examples of the estimation procedures outlined here are given in Ref. [3].

4.3. Variance estimates for volume measurements

The discussion in the previous section applies to the calibration function. In practice, height measurements are used to determine volumes; so the calibration function must be inverted, and confidence intervals for liquid height measurements must be translated into prediction intervals for volume determinations. One method of doing this is given in Section 7.4 of Ref. [1] and exemplified in Appendix B of Ref. [1] (see also Ref. [2]). The determination of simultaneous prediction intervals for volume measurements is an area of current research.

5. CONCLUDING REMARKS

The main features of the data standardization model presented in ANSI N15.19-1989 are described in Section 3 of this paper. Statistical procedures that provide realistic estimates when the calibration data exhibit significant run-to-run differences are outlined in Section 4. A portable, interactive software package that implements these methods is being developed for the IAEA. The software will contain both data acquisition and standardization routines that can be used by inspectors and the analytic capability required by statisticians.

ANSI N15.19-1989 is currently undergoing review, and a revised standard is expected within a few years. Major topics under review include adjustments for thermal differences, the treatment of buoyancy corrections, and the inclusion of procedures for additional systems to measure tank content. The IAEA calibration software will be updated to reflect any changes that are approved for the standard. Finally, we plan to recommend the procedures in ANSI N15.19 to the International Organization for Standardization for adoption.

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REFERENCES


1. INTRODUCTION

Making decisions on the basis of measurements is complicated by the uncertainties inherent in experimental results. Usually, two components of such uncertainties are distinguished: a component varying in an unpredictable way, which is denoted random uncertainty, and a component remaining constant or varying in a predictable way in the course of a measurement, which is denoted systematic uncertainty. The accepted methods of statistical hypothesis testing, as the term indicates, incorporate only random uncertainties. Systematic uncertainties cannot be incorporated in hypothesis tests, but they deteriorate the performance of these tests. Neglecting systematic uncertainties may thus lead to failing to meet the test design specifications of applications, in particular it may lead to an underestimation of the probabilities of erroneous decisions.

We try to quantify these effects, specifically to determine allowable upper levels of systematic uncertainties. The arguments will be developed using an example from the practical application of safeguards. The adopted method of analysis is based on computer simulations.

2. DESCRIPTION OF THE TEST CASES

2.1. Test application

The test application concerns LEU/HEU verification of the uranium enrichment of UF₆ gas in pipework of centrifuge enrichment plants. On the basis of measurements it has to be determined whether the enrichment is less than 20%. In
TABLE I. VALUES OF DECISION PARAMETERS, MEASUREMENT CONDITIONS, DETECTION EFFICIENCIES AND MAXIMUM RELATIVE UNCERTAINTIES

<table>
<thead>
<tr>
<th>Quantity</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>$E$ (wt%)</td>
<td>3 or 20</td>
</tr>
<tr>
<td>$p$ (Pa)</td>
<td>60</td>
</tr>
<tr>
<td>Deposit-gas $^{235}$U activity ratio</td>
<td>20 for $E = 3%$</td>
</tr>
<tr>
<td></td>
<td>3 for $E = 20%$</td>
</tr>
<tr>
<td>$\phi_{in}$ (mm)</td>
<td>36</td>
</tr>
<tr>
<td>$\phi_{out}$ (mm)</td>
<td>42</td>
</tr>
<tr>
<td>$\alpha_{preset}$ (%)</td>
<td>1</td>
</tr>
<tr>
<td>$\beta_{preset}$ (%)</td>
<td>10</td>
</tr>
<tr>
<td>$T$ (s)</td>
<td>1000</td>
</tr>
<tr>
<td>$\epsilon_{1G}$ $(m \cdot s^{-1} \cdot Bq^{-1})$</td>
<td>$3.97 \times 10^{-4}$ (0.5%)</td>
</tr>
<tr>
<td>$\epsilon_{1D}$ $(m \cdot s^{-1} \cdot Bq^{-1})$</td>
<td>$2.17 \times 10^{-4}$ (6.8%)</td>
</tr>
<tr>
<td>$\epsilon_{2G}$ $(m \cdot s^{-1} \cdot Bq^{-1})$</td>
<td>$0.83 \times 10^{-4}$ (1.1%)</td>
</tr>
<tr>
<td>$\epsilon_{2D}$ $(m \cdot s^{-1} \cdot Bq^{-1})$</td>
<td>$2.04 \times 10^{-4}$ (7.9%)</td>
</tr>
</tbody>
</table>

order to make the problem suitable for hypothesis tests, two enrichment levels must be defined:

(a) A low level, below which an 'HEU' decision is considered a serious error. The enrichment value declared by the operator is taken for this level; in practice, it amounts to $\leq 5\%$.

(b) A high level, above which an 'LEU' decision is considered a serious error; a value of $20\%$ enrichment is adopted for this level.

The problem is therefore equivalent to deciding between two hypotheses:

(i) The null hypothesis $H_0$ that the measured enrichment $E$ is consistent with the value $E_{H0}$ declared by the operator.

(ii) The alternative hypothesis $H_1$ that $E$ is larger than $E_{H1} = 20\%$.

2.2. Decision errors and decision time

Two types of decision errors can be made: rejecting $H_0$ when it is true, and rejecting $H_1$ when it is true. The probability of making an error of the first type is
called the false alarm probability, which is indicated by $\alpha$; the probability of making an error of the second type is called the non-detection probability, which is indicated by $\beta$. The false alarm and the non-detection probabilities are both fixed in the example application, at design values of $\alpha_{\text{design}} = 1\%$ and $\beta_{\text{design}} = 10\%$, respectively.

In practice, the (average) decision time $T_{\text{DEC}}$, i.e. the total measurement time (the sum of times per observation of the sample) required for a decision, is at least as important as $\alpha$ and $\beta$.

2.3. Measurement method

In order to correct for the contribution of uranium deposits on the inner pipe wall to the 186 keV signal, two 186 keV $\gamma$ ray measurements with distinct detection geometries are performed simultaneously [1], yielding numbers of counts, $C_1$ and $C_2$, respectively. From these, the value of the enrichment $E$ is obtained:

$$E = \frac{\frac{C_1}{T} - \frac{C_2}{T}}{\frac{1}{\epsilon_{1G} \epsilon_{2D}} - \frac{1}{\epsilon_{1D} \epsilon_{2G}}} K$$

where $\epsilon_{1G}$, $\epsilon_{1D}$, $\epsilon_{2G}$ and $\epsilon_{2D}$ are the detection efficiencies of the two geometries (indicated by the subscripts 1 and 2) for gas and deposit (indicated by the subscripts G and D), $T$ is the measurement time, $p$ is the pressure and $K$ is a calibration constant. $K/p$ is assumed to be known. Table I lists the equipment parameters, realistic design test parameters and worst-case measuring conditions.

2.4. Experimental uncertainties

The detection efficiencies are obtained from 186 keV calibration measurements using gas-only and deposit-only test pipes. For the sake of clarity, only uncertainties from counting statistics in these measurements are considered to contribute to the total systematic uncertainty $\sigma_E$ in $E$.

The random uncertainty $\sigma_{E,\text{rand}}$ is estimated in the usual way as:

$$\sigma_{E,\text{rand}} = \frac{\left(\frac{\epsilon_{2D}}{T}\right)^2 C_1 + \left(\frac{\epsilon_{1D}}{T}\right)^2 C_2}{\left(\frac{1}{\epsilon_{1G} \epsilon_{2D}} - \frac{1}{\epsilon_{1D} \epsilon_{2G}}\right)^2} \left(\frac{K}{p}\right)^2$$

The effect of background and complications arising from the fact that $\sigma_{E,\text{rand}}$ is not a given constant but a random number will be ignored in the discussion.
2.5. Decision rules

Two decision rules applicable to the example application have been analysed and are discussed below.

2.5.1. The sequential probability ratio test (SPRT)

The SPRT [2] is the optimum method with respect to the (average) decision time. The procedure consists of performing a sequence of measurements of equal duration, which is terminated after the \( m \)-th measurement:

(a) If
\[
\sum_{i=1}^{m} E_i \geq \frac{\sigma^2 \ln A}{E_{H_1} - E_{H_0}} + m \left( \frac{E_{H_1} + E_{H_0}}{2} \right)
\]
where
\[
A = \left( \frac{1 - \beta_{\text{preset}}}{\alpha_{\text{preset}}} \right)
\]
then \( H_0 \) is rejected.

(b) If
\[
\sum_{i=1}^{m} E_i \leq \frac{\sigma^2 \ln B}{E_{H_1} - E_{H_0}} + m \left( \frac{E_{H_1} + E_{H_0}}{2} \right)
\]
where
\[
B = \left( \frac{\beta_{\text{preset}}}{1 - \alpha_{\text{preset}}} \right)
\]
then \( H_1 \) is rejected.

Otherwise, the procedure is continued with the \( m + 1 \)-th measurement. The number of measurements is not bound. The measurement sequence can be truncated to remedy this shortcoming (Ref. [2], p. 61 ff.). Computer simulation (see Section 3.2) is also a suitable tool for calculating the effect of this shortcoming on \( \alpha \) and \( \beta \).

2.5.2. The Bayes decision rule

The rule gives optimum performance for cases in which the sample size is fixed [3]. The procedure consists of two steps:
(a) Performing a sequence of measurements until $\sigma_E$ has decreased below the value $\sigma_{\text{DEC}}$:

$$
\sigma_{\text{DEC}} = \frac{E_{H_1} - E_{H_0}}{G^{-1} (1 - \alpha_{\text{preset}}) + G^{-1} (1 - \beta_{\text{preset}})}
$$

where $G^{-1}$ refers to the inverse of the cumulative standard normal distribution.

(b) Comparing $E$ with the critical value $E_{\text{DEC}}$:

$$
E_{\text{DEC}} = \frac{E_{H_1} G^{-1} (1 - \alpha_{\text{preset}}) + E_{H_0} G^{-1} (1 - \beta_{\text{preset}})}{G^{-1} (1 - \alpha_{\text{preset}}) + G^{-1} (1 - \beta_{\text{preset}})}
$$

If $E \leq E_{\text{DEC}}$, $H_0$ is accepted; otherwise, $H_1$ is accepted.

3. QUANTITATIVE CRITERIA AND ANALYSIS METHOD

3.1. Criteria for bounds on systematic uncertainties

First, quantitative criteria on the allowable level of systematic uncertainties are required. The mean values of $\alpha$ and $\beta$ are not suitable parameters for defining these criteria, since the detection efficiencies, once they have been calibrated, are fixed and consequently they deviate from their true but unknown values by fixed amounts. It is proposed here to use criteria of the following form:

*At the 99% confidence level:*

$$
\alpha \leq 2\alpha_{\text{design}}
$$

$$
\beta \leq 2\beta_{\text{design}}
$$

$$
T_{\text{DEC}} \leq 1.1T_{\text{design}}
$$

Use of these criteria requires determination of the distributions of $\alpha$ and $\beta$ due to systematic uncertainties, for worst-case operational conditions.

3.2. Method of analysis

The distribution of $\alpha$ and $\beta$ due to systematic uncertainties can be obtained in several ways. Both analytical derivation and experimental determination are tedious. A much faster and flexible method is computer simulation of both enrichment measurements and calibration measurements. Calibration and enrichment verification measurements have been simulated by means of random generators. Normal
FIG. 1. Cumulative distributions of the actual $\alpha$ and $\beta$ values for the SPRT and the Bayes rule. $\sigma_{\text{sys}}/E = 0\%$ (solid curves), 13\% (dash-dotted curves), 29\% (dotted curves), 56\% (short-dashed curves) and 130\% (long-dashed curves).
distributions have been used, since they are adequate approximations to the actual Poisson distributions for practical measurements. The simulated enrichment measurement was obtained from Eq. (1), using the simulated numbers of counts, \( C_1 \) and \( C_2 \), and the simulated detection efficiencies. The enrichment values obtained in this way are used as inputs for the SPRT and the Bayes rule.

4. RESULTS

Figure 1 shows the cumulative distributions of the actual \( \alpha \) and \( \beta \) values for various values of systematic uncertainties; expressed as fractions of the random uncertainty at decision, these levels are 0.05, 0.10, 0.21 and 0.48 for the SPRT, and 0.08, 0.18, 0.36 and 0.82 for the Bayes rule. The distributions were based on 100 `calibration runs`. Estimates of the actual values of \( \alpha \) and \( \beta \) for one calibration were obtained from typically \( 10^4 \) `enrichment verification runs` with LEU (\( H_0 \) true) and HEU (\( H_1 \) true), respectively. The criterion outlined above is met for both the SPRT and the Bayes rule if \( \sigma_{E, \text{syst}} \) is \( \leq 10\% \) of the random uncertainty at decision.

In practice, it may be difficult to achieve this stringent constraint. Approximate methods to compensate for the effect of systematic uncertainties may therefore be of use, aiming at attaining the design values for \( \alpha \) and \( \beta \). Two candidate methods were analysed:

(a) Combination of systematic and random uncertainties in one overall uncertainty, by quadratic addition;
(b) Use of lower values of \( \alpha \) and \( \beta \) than the design values.

For the SPRT, method (a) is obviously wrong: The importance of the term containing the uncertainty \( \sigma \) in Eqs (3) and (4) diminishes at every step in the sequence, which is the characteristic feature of the random uncertainty of a combined result, opposed to the constant size of its systematic uncertainty. Method (b) could be applied to both the SPRT and the Bayes decision rule.

Analyses have shown that both method (a) and method (b) have the same two effects: First, the values of \( \alpha \) and \( \beta \) improve slightly compared with the design values, with an increase in \( T_{\text{DEC}} \) as a penalty. Second, the distributions of \( \alpha \) and \( \beta \) do not approach those for the case of `no systematic uncertainties`. In particular, the probabilities for the occurrence of large \( \alpha \) and \( \beta \) values are still significant. When considering these points, it becomes clear that these conclusions must hold generally. It is therefore concluded that approximate methods do not work.

5. DISCUSSION AND CONCLUSIONS

Systematic uncertainties worsen the performance of statistical tests. A wrong treatment of these uncertainties in statistical verification applications for safeguards
purposes would lead to false assessment of the strength of the safeguards measure and thus would undermine the safeguards system.

Systematic uncertainties cannot be incorporated in statistical hypothesis tests. The proper way to manage systematic uncertainties is to reduce them to allowable levels on the basis of appropriate criteria. Requiring 99% confidence level bounds on $\alpha$ and $\beta$, an upper level of 10% of the (average) random uncertainty at decision was found, using computer simulation analyses of measurements and decision methods.

Although the qualitative conclusions were obtained from a study of the behaviour of two decision rules applied to a single practical application, it is believed that they are generally applicable.

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STRENGTHENED AND
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(Session 7)

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TECHNOLOGIES FOR DETECTION OF UNDECLARED FACILITIES

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Abstract

TECHNOLOGIES FOR DETECTION OF UNDECLARED FACILITIES.

The United States of America is developing a number of technologies which could be used by the IAEA for the detection of undeclared nuclear activities in States with comprehensive Safeguards Agreements. The US Department of Energy (DOE) provides a substantial portion of the overall US support to IAEA safeguards and is heavily involved in developing technologies for use in searching for undeclared activities. The majority of relevant technologies available from the DOE to support international safeguards in this area, identified as a priority for IAEA investigation by the Director General in his address to the Board of Governors on 2 December 1993, fall within three broad categories: (1) environmental monitoring, (2) alternative approaches to traditional safeguards, and (3) enhanced information management. In the area of environmental monitoring, the USA is currently assisting the IAEA in three major areas: (1) ‘Programme 93+2’ Task 3 – environmental monitoring field trials; (2) technical and financial assistance to the Safeguards Analytical Laboratory (SAL) at Seibersdorf, and (3) general technical support. In the area of alternative safeguards approaches, the USA continues to support development of technologies and approaches such as remote and long term monitoring techniques, which are being explored to reduce the number of inspectors required to conduct on-site inspections and to increase the cost effectiveness of safeguards. The DOE is also developing a number of technologies and technical approaches to augment the concept of long term monitoring, such as the creation of detection capabilities for various signatures of nuclear facilities operations. Diverse types of information are now being utilized at the IAEA, including environmental sampling results, photographs, video film, lists of equipment and activities, and open source materials such as unclassified publications. The ability to acquire, review, store, analyse, validate and retrieve large volumes of such information will provide a challenge to the existing IAEA information management system. Consequently, the USA has been working with the IAEA to implement advanced information management tools which will greatly enhance inspection efficiency and could facilitate the search for clandestine nuclear operations.
1. INTRODUCTION

The past decade has witnessed a multitude of changes in the area of nuclear non-proliferation and international safeguards. In response to these changes, the IAEA has had to face the inherently difficult situation of handling enormous responsibilities, but with severely limited resources. In order to assist the IAEA with the challenges it faces today, the United States of America has offered technical and financial resources to support the IAEA's safeguards activities.

The Office of Export Control and International Safeguards at the Department of Energy (DOE) co-ordinates the development of equipment and training, and offers technical advice in several major areas of principal concern to the implementation of more effective and efficient IAEA safeguards, and which are directly related to and supportive of the IAEA 'Programme 93+2' initiative. The primary areas of support are environmental monitoring, alternative safeguards approaches and information management, as briefly described in this paper.

2. ENVIRONMENTAL MONITORING

One of the most important areas in which the IAEA and the USA are cooperating is the field of environmental monitoring. Techniques for environmental monitoring have been extensively utilized for many years by the DOE to assist in the characterization and remediation of nuclear sites in the DOE complex. In order to strengthen the effectiveness of the IAEA safeguards system, the USA is now assisting the IAEA in the acquisition and assessment of additional data to evaluate integration of environmental monitoring techniques into the application of IAEA safeguards.

2.1. Environmental sampling field trials

In order to help address questions regarding the technical capabilities and the cost effectiveness of environmental monitoring techniques, a variety of field trials are being held by the IAEA in co-operation with Member States. These field trials are focused on hydrological sampling, biota sampling, soil surface deposition sampling, and surface wipe sampling.

Technical assistance has been provided by the USA and DOE laboratories, in particular when requested by the IAEA, on a case by case basis. The DOE has provided technical assistance in design, planning and implementation of such environmental sampling field trials. This assistance includes provision of technical expertise, sampling equipment, and sample analysis services by DOE laboratories which are members of the network of analytical laboratories. The USA has also assisted the IAEA in analysis and interpretation of field trial measurement data.
These data are then compiled in a final report to provide comprehensive information for IAEA evaluation.

Field trials will be held by the IAEA in 1994. In March 1994, the USA is hosting a field trial of environmental monitoring techniques at US uranium enrichment facilities in Oak Ridge, Tennessee. This field trial will also be utilized to train IAEA personnel in meteorological approaches for planning and for gaining practical experience in carrying out environmental sampling operations.

2.2. Clean room and Safeguards Analytical Laboratory

The USA is also assisting the IAEA in the design, construction and operation of a Class 100 'clean room' facility at the Safeguards Analytical Laboratory at Seibersdorf, capable of handling and analysing environmental samples. DOE laboratories have considerable experience in the operation of such facilities and will assist in facility design, planning and operation, development of operating procedures and protocols, personnel training, analytical techniques, and development of a quality assurance/quality control system, both for operation of the laboratory and for interactions with other network laboratories involved in environmental sample analysis on behalf of the IAEA. Initial training of personnel in laboratory operations and development of long term training activities/programmes will also take place at DOE facilities.

2.3. General technical support

In addition, the DOE provides the IAEA with other general technical support in environmental sampling, including the performance of studies on the use of environmental monitoring in routine and non-routine inspections, and assistance in developing technical expertise to evaluate and interpret data.

3. ALTERNATIVE SAFEGUARDS APPROACHES

While IAEA responsibilities increase and the budget remains at zero real growth, the IAEA has been seeking safeguards approaches utilizing advanced technology which could reduce inspection effort. These approaches are designed to increase cost effectiveness while streamlining and enhancing safeguards, potentially becoming a part of a traditional safeguards regime.

3.1. Technical options study

The DOE is conducting a technical options study designed to explore new or improved technologies, methods, safeguards approaches and procedures in an effort
to identify and describe pathways to proliferation, evaluating the attractiveness and probability of particular pathways, and the expected effectiveness of each enhanced option for addressing such pathways, while also addressing significant factors such as cost, intrusiveness, and legal constraints. Signatures that an enhanced system might detect will be listed and options identified for detection of such signatures. This study will emphasize collection methods, co-operative measures, shareable technologies, information, multilateral arrangements, and specific proliferation scenarios anticipated to be of concern. A prioritized set of technical options that could be developed to enhance traditional safeguards will be projected and provided to the IAEA for its review and comment.

3.2. Long term monitoring concepts

In co-ordination with the IAEA, the DOE is also assessing long term monitoring concepts as a part of an alternative safeguards approach. The objective of long term monitoring is to develop inspection regimes which aim at detecting signatures resulting from various nuclear activities.

3.2.1. Radiometric technologies

Radiometric technologies based on alpha, beta, gamma and neutron measurements are key to long term monitoring. For example, large scintillation detectors may be used for detecting gross gamma activity. A multichannel analyser with a high resolution detector may be used to detect gamma signatures. A real time $^{238}$U analyser based on detecting low energy beta particles on the surface of the ground is being considered for deployment by the IAEA.

3.2.2. Positioning and meteorology

In many cases, both a global positioning system (GPS) and a meteorology instrumentation package would be included in any long term monitoring regime. Accurate positioning of where samples are taken and knowledge of the atmospheric conditions would be required. Both are required for any type of backtracking analysis to locate a source term.

3.2.3. Lightweight neutron emission search instrument

The DOE is developing a lightweight neutron emission search instrument. This instrument is being designed to detect neutron emission from radioisotopes and to generate an alarm when the neutron fluence is significantly higher than background. This instrument is small and portable, consisting of a self-contained neutron sensor system.
3.3. Review of the provisions of the Chemical Weapons Convention

The DOE is also conducting a review and analysis of the provisions of the Chemical Weapons Convention which may be relevant to enhanced IAEA safeguards, including issues such as general terms of access and their attendant technical, procedural and legal implications.

3.4. Remote readout

Remote readout is one method being developed to increase transparency of nuclear operations at facilities and in countries subject to IAEA or regional safeguards. In the field of remote readout, the DOE is identifying sensors and sensor combinations, promoting international acceptance of new advanced safeguards approaches utilizing remote readout systems, and performing system modifications to enhance the usefulness and utility to the international safeguards community by deploying integrated safeguards systems for joint field testing at nuclear related facilities worldwide.

Remote readout is capable of integrating advanced technologies into a state of the art integrated monitoring system (IMS) and can be used to demonstrate secure, unattended, unauthenticated remote readout systems for the collection, storage and transmission of safeguards relevant information from remote locations to a central site. As the DOE and its international partners establish worldwide remote monitoring links between nuclear facilities, data collected will be stored for facility operator review and transmitted to the national nuclear material control authorities in the country where the field trial is being sponsored. The central ground station for the US remote readout project has been established at Sandia National Laboratories.

3.4.1. Remote readout field trials

International field trials of a remote readout system will be conducted in certain Member States. Field trials are held to establish the feasibility of remote readout, assess its cost effectiveness and evaluate its possible use in increasing the effectiveness and efficiency of IAEA safeguards. An initial field trial has been under way in Australia since February 1994.

3.5. Special inspections

Because enhancing safeguards effectiveness is of high priority to the IAEA, the DOE is exploring the innovative use of special inspections. The DOE is conducting an analysis of lessons drawn from United Nations Security Council Resolution 687 inspections in Iraq as well as from other non-routine inspections, in order to extract and formulate potential options for future non-routine inspections. The effectiveness
of non-routine inspections will be analysed, including how such inspections were precipitated and conducted, and how inspection information has been assessed and acted upon. Inspection options will then be formulated in the light of the lessons learned, and evaluated in terms of feasibility and potential effectiveness in credible situations and in terms of cost–benefit trade-offs. This study will be presented to the IAEA as a study in non-routine inspection formulation, to use as appropriate.

As a potential tool for use in non-routine inspections, the DOE is developing a rapid response inspection package, the first step of which is comprised of general and specialized kits of technical equipment which would be available to satisfy inspectorate needs. Such kits would contain logistical equipment such as communication, documentation, measurement and containment/surveillance equipment. Specialized kits would be developed for specific theme inspections: radiation detection, chemical and biological agent measurement equipment, etc. Training IAEA inspectors in the use of these kits would also be conducted by the DOE.

Each of these alternative safeguards technologies being developed by the DOE may be used by the IAEA to assist it in achieving an appropriate balance between its increased responsibilities and fiscal restraints. With such a large number of resources available from around the world to draw expertise from, the IAEA is in a unique position to expand its capabilities in the detection of undeclared facilities.

4. INFORMATION MANAGEMENT

The world has entered an age of information — data must be collected, stored, and effectively retrieved and analysed — all in a matter of moments. The IAEA provides a prime example of just how much information must be dealt with. Data received from around the world must be placed in a retrievable format for immediate use.

4.1. Safeguards Information Management System

To assist the IAEA with this task, the USA is working closely with the IAEA to design and develop a Safeguards Information Management System (SIMS) relevant to the IAEA's emerging needs. This system will be designed to augment the existing information management systems at the IAEA, allowing personnel to monitor nuclear activities on a global scale more effectively by making better use of information obtained through inspections as well as other information sources.

The development of this system will provide continual monitoring, search and analysis capabilities regarding worldwide nuclear and safeguards related data and events. Thus IAEA personnel will have a higher likelihood of being alerted to developments of potential proliferation concern.
4.1.1. Contents

It is envisioned that when developed and deployed at the IAEA, SIMS will utilize hardcopy documents (entered into the system with optical character recognition and optical scanner capabilities), electronic text, video, images and graphics (maps and diagrams). Information will be retrieved from IAEA inspection reports, nuclear facility and related databases, the State Systems for Accounting and Control reports provided to the IAEA, open literature, and other databases.

4.1.2. Site modelling capability

In addition to this, it is planned that SIMS will have a site modelling capability. Inspectors will have the ability to become familiar with a site through textual, geographical and video references in order to facilitate pre-inspection planning and post-inspection analysis.

4.1.3. International Nuclear Safeguards Inspection Support Tool

The first step in the SIMS process, and the platform from which SIMS will be built, is the International Nuclear Safeguards Inspection Support Tool (INSIST). INSIST provides the IAEA with certain capabilities for utilization and retrieval of the information received as a result of the non-compliance of Iraq. SIMS will greatly expand the scope of information management for the IAEA as it moves into the twenty-first century.

The IAEA will continue to assess and implement approaches and technologies for the detection of undeclared facilities within the context of limited resources and competing demands. The USA and DOE are fully committed to assisting the IAEA in this area and will continue making available technical and financial resources in the area of enhanced safeguards as the IAEA deems appropriate.
THE ROLE OF SATELLITES AND REMOTE DATA TRANSMISSION IN A FUTURE SAFEGUARDS REGIME

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Abstract
THE ROLE OF SATELLITES AND REMOTE DATA TRANSMISSION IN A FUTURE SAFEGUARDS REGIME.

The revelations of the Iraqi nuclear weapons programme, and the production of milligram and gram quantities of plutonium in Romania and in the Democratic People's Republic of Korea (DPRK) respectively indicated the need to upgrade the safeguards system of the IAEA. This was further emphasized by the subsequent declaration by South Africa that it had produced several hundred kilograms of highly enriched uranium (HEU). Whereas Safeguards Agreements were already in force for Iraq and Romania, the DPRK only agreed to sign a Safeguards Agreement with the IAEA in 1992. These States have compromised IAEA safeguards. South Africa, on the other hand, produced HEU prior to its recent accession to the Non-Proliferation Treaty (NPT), leading to questions about the country's non-proliferation position. The paper, therefore, proposes an improvement in the effectiveness of the IAEA's safeguards system to detect undeclared nuclear activities of a State party to the NPT. In order to exercise the right to conduct 'special inspections' the IAEA needs reliable and independent information on undeclared activities in a State derived from preferably open sources. It is suggested that the IAEA should be provided with a tool enabling it independently to generate the necessary information. In this connection, the possibility of monitoring States parties to the NPT by satellites equipped with suitable sensors is discussed. It should be emphasized that the implementation of special inspections should not lead to a reduction in the traditional safeguards activities of the IAEA in declared facilities. This aspect is already being taken into account insofar as the current technical developments of safeguards essentially aim at improving efficiency without a loss in effectiveness. Efficiency is increased, for example, by using integrated safeguards systems capable of collecting automatically and non-intrusively safeguards data over a prolonged period, which can then be extracted from the system by inspectors at a later date. A more extensive approach is the transmission of data on safeguards via the integrated services digital network (ISDN) or using satellite communication channels.
1. INTRODUCTION

In the early 1990s, several events rekindled extensive discussions on the worldwide nuclear non-proliferation regime in general and the IAEA safeguards system in particular. These were the disclosure of the Iraqi nuclear weapons programme, the declarations by the governments of Romania and of the Democratic People's Republic of Korea (DPRK) of being in possession of plutonium, and South Africa's declaration of having highly enriched uranium.

Ever since, political consequences as well as technical and methodological improvements have been discussed with regard to strengthening the safeguards effectiveness [1]. The goal is to enable the IAEA to detect undeclared nuclear activities and to verify the completeness as well as the correctness of the declarations. This second aspect particularly applies to States which for several decades operated an extensive uncontrolled nuclear programme prior to acceding to the Treaty on the Non-Proliferation of Nuclear Weapons (NPT).

As these challenges add on to the normal IAEA activities, an appropriate increase in the IAEA's resources is required. Obviously, there is no consensus among the Member States on increasing these resources. Therefore, it is necessary both to reallocate the limited resources and to improve the efficiency of safeguards techniques. Current developments in safeguards technologies aim at improving the efficiency without the loss of effectiveness by using integrated safeguards systems capable of collecting safeguards data automatically and non-intrusively, evaluating the acquired data, and transmitting them via satellites. The remote transmission of safeguards relevant information raises both technical and political questions which have to be solved.

The paper discusses the possibilities and problems of both remote data transmission and acquisition by satellite.

2. ASPECTS OF A FUTURE SAFEGUARDS REGIME

The IAEA's right to carry out special inspections is enshrined in the INFCIRC/153 type Safeguards Agreement. But a systematic search of a whole country by IAEA inspection teams for undeclared nuclear activities is not politically possible and cannot be implemented by the IAEA because of limited resources. Both reasons suggest that the IAEA should be provided with a non-intrusive tool enabling it independently to generate the necessary information on clandestine nuclear activities in a Member State. The advantages of such a tool could be greater transparency and, thus, also higher political acceptance of the measures derived.

Remote sensing of territories with the aid of satellites equipped with suitable sensors could be an effective measure. Until now, this method has only been used in the military area. However, owing to the considerable improvements in the quality
of modern commercial satellite systems, they could provide the IAEA with a good opportunity to enhance its safeguards procedures. The usefulness of their images has to be investigated with regard to technical aspects, and the prospects of implementing such new verification methods with regard to political aspects.

While the human resources of the IAEA may be reallocated, e.g. for the implementation of special inspections, the overall effectiveness of safeguards in declared facilities should not be reduced. Current technical developments of safeguards essentially aim at improving the efficiency without a loss in effectiveness. Efficiency is increased, for example, by using integrated safeguards systems capable of collecting safeguards data automatically over an extended period. At a later date, the inspector will extract the data from the system for evaluation. Such systems used exclusively for safeguards purposes may even have interactive functions so that the plant operator, or the staff of the State System of Accounting and Control (SSAC) of nuclear materials, can carry out safeguards relevant tasks in the absence of the IAEA inspector, operating the system without being able to intervene in the database.

In Germany, for example, a safeguards specific multicamera video system with an electronic seal interface has been implemented in two facilities for long term intermediate storage of spent fuel. The facility operator handles the electronic seal under camera surveillance, while the data regarding opening and closing of the seal are being inserted into the video scenes. At any time in the future the IAEA inspector may verify the seal and review the video scenes in order to check their consistency.

A more extensive approach is the remote data transmission of safeguards relevant information. The first phase could involve the possibility of protected data carriers in safeguards systems being replaced by the plant operator or SSAC staff and sent to Vienna for evaluation by the IAEA. A further step could be remote transmission of safeguards data. In this case, not only technical but, more importantly, political questions will have to be solved. This approach would enable the IAEA to reduce on-site inspections without loss of timeliness.

The pending technical questions of remote data transmission concern development of the components for authentication and encryption, while the political questions refer to the release of the data, whereby such questions as transmitting data by mail, e.g. sending video cassettes, will need to be addressed. Not all States and plant operators are prepared to perform safeguards relevant functions or accept an uncontrolled flow of safeguards data to Vienna. The boundary conditions under which the discussed improvements can be applied must be studied in detail.

Many of the technical aspects ruling the authentication and encryption of data for remote transmission have already been addressed in connection with the previous development of the tamper resistant TV link (TRTL) [2], the MOS video system [3], and the MORE review station [4]. Therefore, the implementation of a new task on ‘Design and Testing of Video Data Authentication and Encryption’ under the German Support Programme to the IAEA was a logical step. The components for the mail-in option of video cassettes will be basically the same as for the remote trans-
mission of video data. Here appropriate transmission media, such as the integrated services digital network (ISDN) and satellite communication channels, have to be tested and evaluated, which can be done separately, e.g. under bilateral co-operation with different partners. Preceding field tests have been carried out by Australia (telephone network), Japan and the USA (satellite communication). Apart from the questions of reliability and tamper resistance, cost aspects have also to be investigated.

The expectation is that upon implementation of remote transmission of safeguards data, less effort of the IAEA staff will need to be spent on routine activities.

3. SATELLITE MONITORING

Images taken by satellite based cameras have been successfully used for the verification of many arms control treaties, but not openly for the verification of compliance with the NPT [5]. As long as the IAEA safeguards were concentrated on the verification of declared nuclear materials there was no role for satellite images. However, the recent detection of a clandestine nuclear programme in an NPT Member State has demonstrated the importance of observation satellites, particularly when special inspections are required. A request for a special inspection in the DPRK was triggered because information obtained from satellite imagery corroborated the IAEA’s own findings.

Ideally, information should be collected by observations from space, the air and the ground. While ground and aerial inspections can only be performed with the agreement of the inspected country, monitoring from space does not require it.

Obviously it is not practicable to scan the entire globe in order to find undeclared sites. However, situations might arise in which monitoring of specific areas might be undertaken. It should be remembered that a satellite can image a large area at a time. For example, the French civil SPOT satellite can photograph an area of 3600 km$^2$ and a Russian high resolution satellite covers an area of 1600 km$^2$. The burden of cost could be reduced if initial information came from national technical means (NTM).

Today, the military satellites could be a source of information for the IAEA. But one has to ask whether data from civil satellites such as Landsat, SPOT or RESURS will be useful for the detection of undeclared activities. Will their quality be sufficiently high and can the right field of view be taken to enable proper evaluation? Another aspect will be the cost for both acquisition and evaluation of the data. The data should not be restricted to panchromatic images but should comprise multispectral and even radar images.
One of the basic problems is to establish a catalogue of features for the detection of nuclear facilities from space. The existing fuel cycle facilities could serve as a reference basis. In the following, some of these features are discussed.

— Mining and milling is a vast operation. There have to be appropriate buildings and equipment such as shafts, hoists, and ventilation machinery. Removals from underground could be detected by observing above-ground deposits. As many of these features are common to other mining operations, some characteristic features of uranium/thorium mining could be explored, e.g. specific host rock of uranium deposits or waste heat effluents.

— Gas centrifuge enrichment plants may look like supermarkets. However, they consume significantly more electricity, may not be freely accessible, and they require a significant material flux and storage in specific containers.

— Reactors are relatively easy to detect as long as they are above ground. However, the possibility of underground reactors cannot be ruled out and adequate criteria for the identification of such concealed reactors have to be developed. For example, their waste heat generation could be detected and their detection may be possible by radars. This could depend on the depth at which the facility is located.

— Distinctive features of large reprocessing plants are, for example, the size and concrete shielding of the process buildings, as well as the thermal output, which can be observed using infrared sensors. However, it has to be investigated whether small scale underground reprocessing laboratories emit sufficient heat and radionuclides to enable monitoring by satellite. Again, radars may be useful in this context.

— Laboratories and weapons production plants offer only indirect signatures for their detection. The most recognizable features are security fences, air defence systems, hardened shelters or buildings for the assembly of nuclear warheads, highly protected transports, etc. Possibly some or all of these features could be detected with civil satellite images.

At present there are six countries which have launched and operated civil remote sensing satellites. The European Space Agency (ESA) joined these when the ERS-1 radar satellite was launched on 16 July 1991. Of these countries, only France and the USA have been actively distributing their satellite data on a commercial basis. Now also India, Japan and the Russian Federation are offering satellite data commercially. Several other countries are studying the use of satellite data for arms control verification.
There are nine civil long duration remote sensing satellites in orbit at present. As their launches were not co-ordinated, their orbits are such that observations of the Earth's surface cannot be performed in any logical fashion. Nevertheless, an examination of their orbital parameters indicates that they are not all in one orbital plane; they are widely dispersed relative to each other. This means that a more frequent and wider coverage of certain parts of the Earth's surface is possible.

Of the nine satellites, SPOT 1 and 2, MOS 2 and ERS 1 are all in the same orbital plane. From this plane, the orbital plane separations of IRS 1B, MOS 1, Landsats 4 and 5, and IRS 1A are 2°, 5°, 15° and 28°, respectively. The Landsat 4 and 5 satellites are in the same plane. The similarity of parameters for the two Landsats indicates that Landsat 5 was intended as a replacement of Landsat 4. On the other hand, the two IRS satellites are being used in order to obtain frequent coverage. Their orbital planes are separated by 25.75°. The period of these satellites is such that a point observed by IRS 1B would be under IRS 1A 103.1 min later, during which period the Earth would have rotated 25.78°.

Consider the constellation of all the nine satellites. While SPOT 1 and 2, MOS 2 and ERS 1 are in the same orbital plane, their perigee positions, relative to that of SPOT 2, are at 33.1°, 30.1° and 44.8°, respectively. Other satellites could observe the same target just over 9 minutes after SPOT 2 had left it. During this time the Earth would have rotated just over two degrees. The ERS 1 would be over the target in 12.5 minutes, during which time the Earth would have rotated about three degrees. This would still enable the satellites to observe the target with the radar sensor. With the Landsats and the IRS 1A following the target from different orbits, the revisit time could be much less, if data from different satellites were used.

The analysis of panchromatic images usually enables the identification of civil nuclear facilities because of the more or less standardized configuration of reactor buildings and auxiliary buildings in the case of specific reactor types. In contrast to military installations, passive protection in the form of security fences is much less massive and active security arrangements such as anti-aircraft and missile defence systems are lacking. Using multispectral images, particularly in the infrared region of the spectral bands, it is possible, for example, to monitor the time periods of operation of nuclear reactors. For the production of weapons grade plutonium, common reactors would have to be shut down frequently in order to remove the fuel so as to avoid a high burnup and the buildup of undesirable plutonium isotopes. The cooling water from a production reactor is usually discharged into a river or a lake or a cooling pond. Unless the water is efficiently mixed, it is difficult to hide its rise in temperature from satellite based thermal sensors. Exceptions to this are the CANDU and the Magnox reactors. It would be difficult to monitor such facilities from space because the fuel could be changed without turning the reactor off.
5. PRELIMINARY EVALUATION OF SOME SATELLITE IMAGES

So far, only some SPOT images and one Russian panchromatic image have been acquired, showing presumably military areas with known but unsafeguarded nuclear activities. The present investigation is limited to a brief examination of only panchromatic images and is of a preliminary character.

Images taken over Iraq show the Tuwaitha nuclear research complex some 15 km south of Baghdad. The images were acquired after the 1991 Gulf War. Three nuclear installations, the IRT-5000 and the Tamuz-2 reactors and the fuel storage facility, all declared to the IAEA, can be identified. From reports it is known that all of these were damaged during the war. This would account for several dark patches in the complex.

Another site north of Baghdad, near Al Tarmiya, is identified where an enrichment facility using the electromagnetic isotope separation method was employed. This site does not have any perimeter fences or any other form of detectable defences. However, the whole area has been declared a military area.

In another country, four key nuclear related installations could be identified: a 40–70 MW(th) reactor, a building large enough to cover an underground plant, two mining areas linked to the site by road and railway, and a long building probably used for ore processing and fuel manufacture. In addition, three perimeter fences give an impression of a highly protected military area. A closer examination discloses also active protection, anti-aircraft guns or missiles. Moreover, two very large airfields are situated north and south of the reactor complex which may act as further active defences. On these military airfields conventional ammunition and other storage sites can also be identified. To the northwest, some 25 km from the site, a large ammunition storage site was detected. There are widely spaced bunkers compared to those near the airfields that could be housing nuclear weapons.

In a third country, two images of a main research centre were available, one SPOT image and one Russian image with a higher resolution. They show, especially in an enlarged display, a number of details of a research reactor site and of a nuclear science related research institute. Because of the high resolution of the Russian image, details of defence related activities are detectable. Whereas the institute has a strong perimeter fence, only a small area at the research reactor complex seems to have a security fence. On the Russian image a military airport and a surface-to-air missile site are also detectable.

6. CONCLUSIONS

A first evaluation of commercially available satellite images has shown the potential for detection of unsafeguarded nuclear activities. More work is necessary
to establish a comprehensive list of evaluation criteria and a reference basis comprising different types of commercial nuclear facilities. The use of commercially available satellite images by the IAEA seems to be promising, even though only panchromatic but no infrared or radar images have been evaluated yet.

Remote data transmission of data from safeguards equipment can enhance the IAEA’s efficiency. Development projects are under way aiming at reducing on-site inspections.

REFERENCES

PROSPECTS FOR ENVIRONMENTAL MONITORING IN INTERNATIONAL SAFEGUARDS*

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Abstract

PROSPECTS FOR ENVIRONMENTAL MONITORING IN INTERNATIONAL SAFEGUARDS.

As part of its ongoing programme to strengthen the safeguards system, the IAEA is currently evaluating possible approaches that might be used to improve confidence in the absence of undeclared nuclear activities. One such approach involves the analysis of environmental samples ('environmental monitoring') to detect releases of radionuclides and other species that are indicative of key nuclear fuel cycle activities. The results of such monitoring can then be compared with the nuclear activities that have been declared by States. The paper reviews the main indicators for processes leading to the production of special fissionable materials. The nature of environmental releases is discussed, together with the analytical methods that would form the basis of any monitoring programme. Conclusions of a Consultants Group which met to advise the IAEA on the use of environmental monitoring are presented, and the prospects for the successful implementation of environmental monitoring are discussed.

1. INTRODUCTION

Over the last few years, considerable attention has been given to strengthening the IAEA's safeguards system with respect to the detection of undeclared nuclear activities. One of the tools available to the IAEA for this purpose is the collection and analysis of 'environmental' samples to obtain indications of the nature of nuclear related activities in a particular area. To help the IAEA decide how best to proceed with the evaluation of environmental monitoring techniques, an IAEA Consultants Group meeting was convened in March 1993. The present paper draws heavily on the technical advice and on the recommendations and conclusions of that meeting, and examines the potential of the recommended techniques.

Whilst it might be desirable for the IAEA to develop the capability to detect any undeclared nuclear activities, the first priority will be the detection of processes leading to the production of nuclear material that could be used directly for the

* Views expressed in this paper are those of the author and should not be taken as necessarily representing Government policy.
manufacture of nuclear explosive components (e.g. separated plutonium and highly enriched uranium, HEU). For plutonium, the key processes to be detected are: undeclared reactor operations, reprocessing of irradiated fuel, separation and purification of plutonium, and the conversion of plutonium nitrate to metal. For HEU, attention focuses mainly on the enrichment of uranium or the reprocessing of irradiated HEU fuel, and on the conversion of HEU product to HEU metal.

In evaluating opportunities for environmental monitoring it will be important to take into account possible complicating factors. Radionuclides and other species of interest in the environment are either from nuclear weapons testing or from environmental releases from legitimate nuclear operations; detection of any undeclared activity has to be accomplished against the background level. There are also various control strategies which a potential proliferator might adopt to limit releases or to mask operations. These strategies might include, for example, siting a facility underground, collocating an undeclared activity with a similar declared activity, and the use of abatement technology to substantially reduce emissions. However, it is considered very unlikely that, even with such countermeasures over a period of time, all indicative emissions from facilities can be prevented. Accidents and unplanned releases, such as minor contamination incidents, can occur even at facilities adopting the most advanced measures of control.

Environmental monitoring offers possibilities for both short range and long range detection of undeclared nuclear activities. Short range application of environmental monitoring would be at an existing declared nuclear facility or, for example in the context of special inspections, at a site which had not been declared as having any nuclear activities. In the former case it is possible that a potential proliferator might seek to utilize the facilities and the expertise available at an existing nuclear site as part of a covert programme at the same site. For long range application of environmental monitoring there might be no specific target facility. A brief summary of some of the important features and indicators relevant to the detection of undeclared plutonium or HEU production is given below.

2. UNDECLARED PLUTONIUM PRODUCTION

Detection of plutonium production at an undeclared reactor by environmental monitoring involves, inter alia, the measurement of radioactive gases in the atmosphere and of radionuclides in surface water. Close-in detection of gases (such as $^{133}$Xe, $^{41}$Ar, tritium and $^{14}$CO$_2$) is a powerful tool to confirm the presence of an operating reactor, although success will be dependent on the interception of the released plume. Analysis of wipes, articles of clothing and aqueous effluents for traces of activation and fission products and of actinides can also provide good indications. Effluent signatures can also be used to detect reactors over longer distances (hundreds of kilometres); indicators can often be traced back to their source: many
radioactive species remain in stream sediments for several years, gradually migrating downstream and concentrating in aquatic biota.

In a reprocessing plant, the first possibility for the release of significant quantities of radioactive species occurs when the fuel elements are dismantled and the irradiated fuel is dissolved. Fuel dissolution releases a wide range of volatile isotopes (e.g. $^{85}$Kr, $^{133}$Xe, $^{131}$I, $^{129}$I, $^{14}$C and $^{103}$Ru/$^{106}$Ru) together with some non-volatile isotopes (e.g. Pu, Zr/$^{95}$Nb, $^{144}$Ce and $^{134}$Cs/$^{137}$Cs) as an aerosol of fuel solution droplets. Non-aqueous dissolution processes also need to be taken into account and will require careful study in terms of defining potential signatures.

A wide range of possibilities exist for the separation of plutonium from fission products. In the context of detecting undeclared nuclear activities, the presence of chemicals that are indicative of processes such as solvent extraction, ion exchange, pyrochemical methods and precipitation may be of significance. Examples of some of the more common chemicals associated with the separation process are organophosphorous compounds, secondary and tertiary amines, alkanes, aromatic hydrocarbons, and reducing agents such as hydroxylamine. The conversion of plutonium nitrate to plutonium metal may involve typically a precipitation step and calcination to the oxide, and then high temperature fluorination with HF and reduction of plutonium fluoride to metal; non-radioactive indicators include HF, H$_2$O$_2$, oxalate ions and iodine.

For species which cannot be totally removed by trapping, each of the above steps brings with it the possibility of tell-tale releases. Emissions may lead to trace contamination of actinides (e.g. characteristic plutonium isotopes and $^{241}$Am) and other indicators inside a plant, for example in dust deposits; there may also be releases to the outdoor environment, through liquid discharges via drains or pipelines, or airborne releases of gases or aerosols. Clearly, isotope specific analysis of plutonium will be a very important part of any programme to detect its undeclared production.

The use of a mobile laboratory, suitably equipped for gamma spectrometry, tritium liquid scintillation counting and noble gas detection, has been suggested as a way for the IAEA to develop an in-field analytical capability of its own. Field based analytical methods for the detection of plutonium production include portable sodium iodide and germanium gamma spectrometers, portable X ray fluorescence (XRF) equipment, ion mobility spectrometers, and gas chromatography mass spectrometer (GCMS) equipment.

A number of possibilities exist for long range detection of plutonium production. Aquatic sampling is considered to provide the most cost effective approach for an IAEA based detection system because continuous sampling would probably not be needed; aquatic sampling would include the collection of sediments and biota in which some indicator species are concentrated. Air sampling for indicators such as $^{85}$Kr is another potentially powerful technique, but the cost and complexity of its widespread application over large areas are high compared with those of water sampling.
3. UNDECLARED HEU PRODUCTION

Several technologies can be used for the production of HEU, including gaseous diffusion, gas centrifuge, vortex tube and electromagnetic isotope separation. In any separation technology, some enriched uranium will inevitably be released to the environment, and samples taken at or near an enrichment facility may contain some of the enriched (or depleted tails) material. Releases are possible from the feed, product and waste streams of the enrichment process.

Detection of altered isotopic ratios in uranium provides an unambiguous indicator of enrichment activities. Accurate measurements of the isotopic abundances of $^{234}\text{U}$, $^{235}\text{U}$ and $^{238}\text{U}$ are necessary; changes in $^{234}\text{U}$ or $^{235}\text{U}$ abundance provide evidence that man-altered uranium is present; the ratio of $^{234}\text{U}$ to $^{235}\text{U}$ can provide some information about the type of enrichment technology used; and the presence of $^{236}\text{U}$ indicates the presence of uranium that has been exposed to a neutron flux.

Detection of uranium oxide conversion to UF$_6$ may indicate preparation for enrichment activities; possible signatures include uranyl fluoride (UO$_2$F$_2$) resulting from the hydrolysis of any UF$_6$ released, HF, and fluoride ions. Relevant analytical techniques include thermal ionization mass spectrometry (TIMS), for the determination of uranium isotopes and for UO$_2$F$_2$, ion chromatography and electron spectrometry.

The main transfer mechanisms for uranium, on a site and in the immediate vicinity of a site, are air transport and deposition of particulate matter, and surface water transport. Corresponding sample types include wipe samples (for interior surfaces), vegetation (e.g. pine needles, mosses, lichens), faecal matter, and surface soil or road surface samples. Transport of material to longer distances can occur via waste water streams, natural surface drainage and surface streams; samples are therefore likely to include water and sediments. As in all sampling regimes for environmental monitoring where ultra-trace quantities of materials are being analysed, close attention must be paid to the sample handling procedures in order to reduce the risk of accidental sample contamination.

4. ANALYTICAL METHODS

The Consultants Group Meeting also provided the IAEA with detailed advice on the available sampling and analysis methods, quality assurance, training and data evaluation. As might be expected, a wide range of techniques can be used for environmental analysis, depending on the species to be analysed, the sample type and the sampling area. It would not be appropriate to list details here, but a few of the main points of advice are noted.

Mass spectrometry in its various forms — TIMS, inductively coupled plasma mass spectrometry (ICPMS) and accelerator based mass spectrometry (AMS) — is
### TABLE I. DETECTION OF SMALL RELEASES TO THE ENVIRONMENT

<table>
<thead>
<tr>
<th>Type of sample</th>
<th>Indicative maximum useful ranges</th>
<th>Most promising isotopes</th>
</tr>
</thead>
<tbody>
<tr>
<td>Air</td>
<td>5 to &gt; 100 km</td>
<td>$^3\text{H}$, $^{85}\text{Kr}$, $\text{Pu}(\alpha)$, $^{241}\text{Am}$, U, $^{95}\text{Zr}$</td>
</tr>
<tr>
<td>Deposition</td>
<td>50 m to 5 km</td>
<td>$^{129}\text{I}$, $\text{Pu}(\alpha)$, U</td>
</tr>
<tr>
<td>Soil/dust</td>
<td>50 m to 5 km</td>
<td>$^{90}\text{Sr}$, $^{95}\text{Zr}$, $^{95}\text{Nb}$, $^{106}\text{Ru}$, $^{129}\text{I}$, $\text{Pu}(\alpha)$, $^{241}\text{Am}$, U</td>
</tr>
<tr>
<td>Vegetation</td>
<td>50 m to 5 km</td>
<td>$^3\text{H}$, $^{129}\text{I}$, $^{95}\text{Zr}$, $^{95}\text{Nb}$, U</td>
</tr>
<tr>
<td>River water</td>
<td>100–1000 km</td>
<td>$^3\text{H}$, $^{125}\text{Sb}$, $^{129}\text{I}$, $^{137}\text{Cs}$, $^{237}\text{Np}$, $\text{Pu}(\alpha)$, $^{241}\text{Pu}$, $^{241}\text{Am}$</td>
</tr>
<tr>
<td>Freshwater sediment</td>
<td>&gt;1000 km</td>
<td>$^{95}\text{Nb}$, $^{110}\text{Ag}^{m}$, $\text{Pu}(\alpha)$, $^{241}\text{Pu}$, $^{241}\text{Am}$, Cm</td>
</tr>
<tr>
<td>Freshwater vegetation</td>
<td>&gt;1000 km</td>
<td>$^{95}\text{Nb}$, $^{110}\text{Ag}^{m}$, $\text{Pu}(\alpha)$, $^{241}\text{Pu}$, $^{241}\text{Am}$</td>
</tr>
<tr>
<td>Sea water</td>
<td>&gt;5 km</td>
<td>$\text{Pu}(\alpha)$, $^{241}\text{Am}$, $^{237}\text{Np}$</td>
</tr>
<tr>
<td>Marine sediment</td>
<td>&gt;1000 km</td>
<td>$^{95}\text{Zr}$, $^{95}\text{Nb}$, $^{164}\text{Ce}$, $^{237}\text{Np}$, $\text{Pu}(\alpha)$, Cm</td>
</tr>
<tr>
<td>Marine algae</td>
<td>&gt;5 km</td>
<td>$\text{Pu}(\alpha)$, $^{241}\text{Pu}$, $^{241}\text{Am}$</td>
</tr>
</tbody>
</table>

An important cornerstone of the analytical capability needed for ultra-trace measurements. Mass spectrometry is recommended as a measurement method for a wide range of species, including organics, tritium, $^{129}\text{I}$, $^{14}\text{C}$ and, importantly, actinide isotopes. ICPMS is regarded as an important technique, particularly for water sampling programmes.

Techniques such as electron spectrometry and ion microprobe are important for the analysis of specific compounds such as $\text{UO}_2\text{F}_2$. Scanning electron microscopy (SEM) is suggested as one of the possible methods for screening particulate material on air filters, water filters and smear samples. Radiometric techniques, such as low level alpha, beta and gamma counting, are also available for measurements of suitably prepared samples of actinides, and of fission product and activation product species.

Table I provides a summary of indicative maximum useful ranges for detecting small environmental releases of radioactivity, given optimum sampling conditions [1]. The most promising isotopes are also listed. The ranges quoted have been calcu-
lated taking into account possible release rates, typical background levels of contamination and the limit of detection of commonly available analytical techniques. The figures are therefore conservative and, in practice, the detection ranges, for state-of-the-art techniques and equipment, could be somewhat longer. It should be noted that, especially in aquatic systems, the possibility of detection at ranges of > 10 km is strongly dependent on environmental conditions, and interferences from legitimate nuclear industry releases may also be a critical factor.

The existing analytical capabilities of the IAEA are already significant, but they will need to be supplemented in order to cover the analytical problems in connection with environmental monitoring. However, it is also evident that, for practical reasons and on grounds of cost effectiveness, the IAEA will need to continue to make appropriate use of some of the more specialized analytical capabilities of Member States' laboratories (e.g. techniques such as AMS). An important element of the overall analytical system is the application of quality assurance, covering matters such as training, reference materials and standards, sample treatment and archiving, and the performance standards of participating laboratories. Given the large amount of information likely to be generated from any environmental monitoring regime, it is considered important that data evaluation methodologies (use of databases, mathematical and statistical analysis, generation of sampling plans, etc.) should be carefully developed at the outset.

5. CONCLUSIONS OF THE CONSULTANTS GROUP

The Consultants Group provided the IAEA with a large number of detailed conclusions and recommendations, which included the following main points:

A. For both reprocessing and uranium enrichment a range of signatures exist that in principle would allow the detection of undeclared activity. For uranium enrichment, the primary signature would be the detection of disturbed isotopic ratios. For the detection of reprocessing and nuclear facilities a wide range of signatures have been identified.

B. It is unlikely that detection of such signatures will always provide unambiguous identification of undeclared activity. It would, however, focus the attention of the IAEA on particular sites or countries and trigger further investigations.

C. In the context of the IAEA programme and likely resources, the most cost effective, wide area monitoring approach would be based on the collection of water samples and their analysis in the laboratory.

D. For a sampling programme close to, or within a facility, there is a very high probability of being able to confirm the existence of an undeclared nuclear activity. The probability of detection may be at least an order of magnitude lower for a longer range environmental sampling programme.
E. The probability of success of a long range monitoring programme would be improved if supplementary information (e.g. supplied by Member States) is taken into account.

F. It is realised that significant environmental monitoring experience, available in Member States, remains unpublished. If access to this information is given to the IAEA it would help the IAEA in selecting and demonstrating (testing) the most suitable techniques.

G. Quality assurance will be an essential component of an environmental monitoring programme. This will include: training, selected laboratories, standardized and documented sampling, analysis, data management, model verification and assessment procedures.

H. The use of a structured environmental sampling programme as part of routine safeguards inspection activities has the potential to provide a cost effective and valuable source of information on the activities being undertaken, declared or otherwise, at nuclear facilities.

I. Although the IAEA’s analytical facilities in Austria and at Monaco provide a valuable existing resource, they do not include the extensive capabilities that would be needed for the type of environmental analysis required for safeguards applications.

Further research and evaluation work in several specific technical areas was also recommended.

6. DISCUSSION

Possible ways in which the IAEA might use environmental monitoring are listed below:

— Environmental monitoring in support of Special Inspections: for short range detection of nuclear activities when the IAEA has certain information which leads it to suspect that there may be undeclared nuclear activities at a particular location.

— Environmental monitoring for short range detection: this should be done not only when there is cause for suspicion but also as a routine part of safeguards inspections at or near existing declared nuclear facilities.

— Wide-area routine environmental monitoring in a State. There is a wide spectrum of possibilities for this option, ranging from water sampling programmes to comprehensive monitoring of water, soil, biota, vegetation and air.

The IAEA has already accumulated some experience in the application of environmental monitoring relevant to each of these cases. Its evaluation Programme 93 + 2 addresses detailed technical possibilities, as well as considerations regarding cost–benefit and practicability, and the associated legal questions.
This paper has concentrated on undeclared plutonium and HEU production, but any future programme would also need to address the possibility of undeclared thorium fuel cycle activities and the associated production of $^{233}$U. Further consideration might also be given to the possible use of certain physical 'monitoring' techniques by the IAEA, for example commercial satellite infrared imagery to identify cooling water outflows from reactors or enrichment plants.

It is suggested that the IAEA focus its attention initially on the possibilities of developing and proving a sound capability for short range environmental monitoring. Although the scope for immediate application of short range environmental monitoring is excellent, full development of a routine system of environmental monitoring, especially if it is applied over wider areas, may require many years of sustained effort. By way of analogy, some conventional safeguards methods (e.g. the use of near real time materials accountancy, as well as of NDA and C/S systems), which were applied for the first time some decades ago, continue to be developed and improved. We should therefore look at environmental monitoring not as a dramatic 'quick fix' solution but as a range of techniques that can be adopted as their effectiveness is established. In due course, developments in analytical science are likely to lead to improvement in the limits of detection and discrimination available to the IAEA, and monitoring programmes will need to be reassessed from time to time in the light of such developments.

In assessing the scope for environmental monitoring in international safeguards it will be important to bear in mind that this approach constitutes only one of a number of sources of information concerning the possible existence of undeclared activities. It is likely that the utility of techniques such as environmental monitoring will be greatly enhanced when they are coupled with information from other sources. Full integration of environmental monitoring with routine safeguards inspections and other strengthening measures should help to minimize the complexity and cost of the required logistical support (e.g. the number of duty visits). Once the potential and the cost of the techniques are more clearly established, as a result of the IAEA's evaluation programme, it will be necessary to consider carefully how they might be included in the framework of the safeguards system.

Environmental monitoring, like other sources of information, is unlikely to be able to deliver definitive proof, one way or another, as to the existence of undeclared nuclear activities. Whilst the available techniques are powerful, they will not provide an absolute guarantee that no undeclared facilities exist in a State. However, subject to confirmation by the ongoing evaluation programme of the IAEA, environmental monitoring should provide the IAEA with a range of potentially powerful tools that will allow reasonable questions about the nuclear programme of a State to be raised, and hopefully to be resolved.

In taking this initiative forward, it will be important for the IAEA to involve as many Member States as possible in the evaluation and development phase. Confidence in the application of environmental monitoring will grow through experience
with its application, and through objective scientific assessment of the utility, limitations and costs of the techniques.

REFERENCE

ENVIRONMENTAL SAMPLING FOR THE DETECTION OF REACTOR OPERATIONS IN A COASTAL AREA

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Abstract

ENVIRONMENTAL SAMPLING FOR THE DETECTION OF REACTOR OPERATIONS IN A COASTAL AREA.

The IAEA, in co-operation with Member States, is undertaking a series of field trials as part of its Programme 93 + 2 to evaluate environmental monitoring as a potential measure for the strengthening of safeguards. The first of these trials was conducted in Sweden in September 1993 near five nuclear facilities. Water, sediment and biota samples were collected, up to 25 km from the facilities, for radionuclide and trace metal measurements. Results from these tests show that nuclear reactors located at and discharging effluents into coastal water can be detected from the presence of activation products in waters and sediments near and up to 20 km from the outfalls, depending on local conditions. Mass spectrometric measurements could detect the presence of ultra-traces of high burnup plutonium near a facility. Tritium measurements, when used in the context of regional concentrations, showed anomalies around one facility. Trace element measurements indicate that detection of zirconium may indicate reactor operations. Additional work is under way to fully characterize the data obtained from this first field trial.
Environmental methods have been proposed [1] as one possibility to strengthen safeguards by the IAEA's Standing Advisory Group on Safeguards Implementation (SAGSI). In March 1993 the IAEA convened a Consultants Group Meeting (CGM) to evaluate the potential of such methods. The CGM identified the taking of water and sediment samples, combined with highly sensitive analytical techniques, as a means for detecting facility specific signatures; the measurement of certain radionuclide contaminants and their ratios could be used to determine the type of nuclear operations present. It further recommended to carry out field trials for the demonstration of the capability of the techniques in the long and short range detection of nuclear activities [2].

FIG. 1. Environmental monitoring in Sweden (Sep. 1993).
Following up on these recommendations, the IAEA started in August 1993 with the planning of a series of field trials. They will be carried out (a) to establish and document the major environmental signatures for reprocessing, enrichment and reactor operations; (b) to establish and document the procedures for sampling and analysis; and (c) to test these procedures for effectiveness, cost and viability under a variety of conditions.

With the assistance of the Swedish and the United States Support Programmes to the IAEA, an extensive sampling campaign was carried out in Sweden in September 1993. Sweden has a mature nuclear infrastructure that includes nuclear power plants, fuel fabrication, research facilities and waste operations (Fig. 1) and therefore provided a good test area for the first field trial.

2. SITE SELECTION AND SAMPLING PLAN

The vicinities of five Swedish nuclear facilities were selected for the sampling. The selection was based on nuclear operations and included Forsmark, Oskarshamn, Ringhals, Studsvik and ABB Atom in Västeras. Forsmark has three reactors and is the site of the permanent repository for low and intermediate level nuclear waste. Oskarshamn operates three reactors and is the site for the interim storage of spent fuel. Ringhals operates four reactors and is located on the North Sea, and the local waters may reflect influence from other nuclear operations in Europe. The research facilities at Studsvik include a 50 MW(th) test reactor, hot cells and waste operations. The ABB Atom fuel fabrication facility in Västeras was also sampled.

All of the Swedish nuclear facilities are located near the surrounding seas (the Baltic Sea and the North Sea). The Baltic Sea is a semi-enclosed basin that is connected with the North Sea by the straits of Denmark. Owing to the restricted exchange with ocean water from the North Sea, the Baltic Sea contains mostly fresh water (80%), which results in a renewal time of about 40 years for water in the Baltic Sea. Owing to this long renewal time, many elements supplied from the surrounding rivers, lands, discharges and atmospheric deposition are trapped. The North Sea has an open connection with the Atlantic Ocean and contains about 90% ocean water, and tides have a significantly greater range than in the Baltic Sea. The local waters may reflect influence from other nuclear operations in Europe. The ABB Atom facility discharges into a fresh water lake about 100 km west of Stockholm.

A total of 30 locations were selected for sampling, and the primary sampling locations were chosen to extend from the outfall of each facility to 20–30 km along the coast. In the field, secondary sampling locations had to be chosen to collect biota and sediment samples.
3. SAMPLING METHODS

Field operations consisted of taking environmental samples in the coastal waters adjacent to nuclear facilities, mainly from boats provided by the local authorities.

Grab water samples of 0.1–0.5 L were taken for tritium and trace element measurements. Water (300 L) was passed through a special filter to concentrate the gamma emitting radionuclides, uranium and plutonium (Fig. 2). Sediment samples were collected by a small dredge sampler or by hand if the water was less than 0.5 m deep. Depth recorders on the boats were used to locate flat sea basins where soft sediments tend to deposit. Other samples (algae or other vegetation) were collected because of their ability to concentrate radionuclides during their lifetime. The field data collected included the pertinent sample identification numbers, the temperature and conductivity of water, the dry and wet bulb air temperatures, the barometric pressure, the sediment sample description, and the latitude and longitude co-ordinates.
4. SAMPLE PREPARATION AND ANALYSIS

The samples were mainly processed at the Savannah River Technology Center (SRTC), using procedures developed for environmental measurements. Tritium measurements were also performed on parallel samples at the IAEA's laboratories. The samples were handled and processed in clean room facilities of Class 10 000 and Class 1000 to minimize the possibilities of contamination. As a sample progressed through chemical separation procedures, more stringent clean room conditions were used. Facilities of Class 1 and Class 10 were used in the final steps of sample preparation for mass spectrometer measurements.

A variety of chemical and analytical procedures were used in the analysis of the samples:

(a) Non-destructive gamma counting was performed first on filters and sediments to prioritize samples for Pu and U isotopic analysis.
(b) Gamma spectrometry measurements were made in the SRTC Underground Counting Facility [3], which is 15 m below ground level and is constructed with materials with low background radiation. Concentrations of $^{137}$Cs of 0.2 mBq/L can be measured when the low level counting method is combined with the high volume water filter.
(c) Plutonium isotopic ratios were measured by a single magnetic sector mass spectrometer whose design permits the measurement of isotopic signatures on as few as $10^8$ atoms of Pu [4]. The concentrations were determined by isotope dilution mass spectrometry (IDMS).
(d) Uranium isotopic ratios were determined by a three-stage mass spectrometer. Special chemical procedures were used to isolate the Pu and U from environmental matrices.
(e) Laser fluorescence was applied to determine the total uranium concentrations in the samples.
(f) Tritium measurements at SRTC were made using gas proportional counting on samples converted to hydrogen gas [5]. The samples analysed at the IAEA were enriched by electrolysis and measured by liquid scintillation counting.
(g) Trace element concentrations were measured using an inductively coupled plasma mass spectrometer.

5. RESULTS AND DISCUSSION

5.1 Gamma spectrometry

Gamma spectrometry showed the presence of man-made radionuclides in the water filters and sediments. The most readily detected radionuclides were activation
FIG. 3. Normalized $^{60}$Co activity in water samples.

FIG. 4. Normalized $^{60}$Co activity in sediment samples.
products, i.e. $^{54}\text{Mn}$, $^{58}\text{Co}$, $^{60}\text{Co}$, $^{65}\text{Zn}$, $^{110}\text{Ag}^{m}$ and $^{125}\text{Sb}$. The activities measured were at a maximum near the effluent points and rapidly decreased with distance. Figures 3 and 4 show that $^{60}\text{Co}$ could be detected at distances up to 20 km away from the effluent points. The $^{60}\text{Co}$ activities are normalized to the highest activity seen in a water sample from location U. They are shown for each facility location (coded as U, H, Q, T) and for the individual sampling points at various distances (the effluent points are marked by 0, and the sampling points to the left are indicated by a negative kilometre value). The activities measured in the sediment samples are by four orders of magnitude higher.

Radionuclide concentrations in the water filter and sediment samples can also provide insight into their sources. For example, the ratios of $^{134}\text{Cs}/^{137}\text{Cs}$ are slightly different for the two sample types (Fig. 5). The average ratios in the sediment samples observed at the four locations are around 0.05, which is close to that for residual Chernobyl fallout in the region and indicates that this is still present in the sediments. The $^{134}\text{Cs}/^{137}\text{Cs}$ ratios for the water samples are generally higher, and their averages range from 0.07 to 0.09, showing the presence of Cs radioisotopes from other, more recent, sources introduced into the water column. These results will be important reference data in the interpretation of results from other field trials, related to reactors, hot cells and reprocessing operations.

5.2. Plutonium and uranium

The measurement results available so far are only summarized. Measurements of plutonium in water filter samples, using IDMS, generally showed concentrations
at the fg/L level which were too low for an accurate isotopic analysis. Only one sample had sufficient plutonium to determine the isotope ratios. The results found are indicative of plutonium produced in light water power reactors and further indicate that operations associated with LWR fuel may be occurring in the vicinity.

Total uranium was measured using laser fluorescence. The concentrations found at the sampling locations were within the normal environmental concentration ranges expected in surface waters, i.e. a few parts per billion.

Uranium isotopic abundances, determined using thermal ionization mass spectrometry, showed slightly enriched $^{235}\text{U}$ at the same location where the LWR type plutonium was found. The combination of these two observations would indicate that some type of fuel characterization activities may be present in the vicinity. All other samples had uranium isotopic signatures that were within the normal environmental range.

5.3. Tritium

Grab water samples were analysed for their tritium content by two techniques. At only one sampling point, slightly elevated values were found. At all the other locations, the tritium concentrations were similar near and distant from the effluent points.

5.4. Trace metals

The concentrations of several trace metals, such as Fe, Ni, Co, Cr, Mn, Zr, Cd, La and Ce, were determined in the grab water samples by inductively coupled plasma mass spectrometry. Most results showed no correlation between concentration and location of the facility outfalls. Zirconium, however, did appear to be correlated with the sampling location. To better understand the elevation in the Zr concentration in waters near the facility outfalls, it is planned to determine the Zr/Hf ratio in these samples. If the elevated Zr is due to natural processes, the Zr/Hf ratio should be constant at all locations. If the elevated Zr is due to corrosion products in the effluents of the facilities, the Zr/Hf ratio should increase near the facilities.

6. CONCLUSIONS

Results from these tests show that nuclear operations in coastal areas can be detected up to 20 km from the facility, depending on releases and local transport conditions. Nuclear reactor operations can be detected by the presence of activation products in water and sediment samples. IDMS measurements show the presence of high burnup plutonium near one facility, which may indicate operations related to
spent fuel characterization studies. Tritium measurements, when used in the context of regional concentrations, showed the presence of local anomalies around one facility. Trace element measurements indicate that detection of zirconium may have the potential for indicating reactor operations. Work is continuing to fully characterize the data obtained from this first field trial.

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Rapporteur Presentation

ENVIRONMENTAL SAMPLING AND ANALYSIS

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1. INTRODUCTION

Environmental monitoring techniques, perhaps better termed environmental sampling and high sensitivity analysis, have been recommended by the Standing Advisory Group on Safeguards Implementation (SAGSI) as a technical means for IAEA efforts to detect undeclared nuclear activities.

A Consultants Group Meeting on this subject was held in 1993, and the IAEA is currently gaining practical experience in a variety of sampling and analysis techniques through field trials being conducted in its Programme 93 + 2, Task 3.

The following four technical papers, which are briefly reviewed here, are on the subject of environmental sampling and analysis. Three of the papers draw upon empirical data that have been acquired as a result of environmental sampling operations. These were conducted in the vicinity of facilities under the United States Department of Energy; these facilities were involved in the production and processing of nuclear materials for the US nuclear weapons programme.

One of the papers discusses the use of 'clean' chemical and instrumentation laboratories and of sensitive analytical techniques for the determination of actinide concentrations in complex sample matrices. Two of the papers contain extensive general discussions of the technical capabilities of environmental sampling and analysis for safeguards applications, and one paper describes this subject from the perspective of a private environmental engineering firm that has been associated with environmental compliance and waste remediation activities within the regulatory framework of the USA.

2. REVIEW OF THE PAPERS

2.1. Environmental sampling and analysis as a safeguards tool

In this paper it is pointed out that environmental sampling and radionuclide analysis can be utilized as a new approach in international safeguards, especially in
the detection of undeclared nuclear activities. Sample collection and analysis can indicate the fate of key signature materials from uranium enrichment or plutonium production within different environmental settings. The methodology used integrates information regarding the sources for individual signature materials within the primary pathway and alternative environmental pathways along which the materials migrate. Meteorological, geological and hydrological information is used to determine where, what and how often sampling should be performed to provide the greatest likelihood for detection if signature materials are present.

A proliferating country pursuing $^{239}$Pu production produces signature materials such as short lived and long lived radioisotopes from the fission process. Short lived isotopes would be found if the effluents had just been produced. Longer lived fission process radioisotopes would be measured after longer periods, with progression of the proliferation programme. The longest lived radioisotopes, of course, are also from past weapons test programmes; specific isotope ratios can be used to delineate the source term.

The signature radionuclides associated with reactor operations include those in both gaseous and liquid effluents. To conduct a survey, radionuclides with half-lives of several hours or more are collected at suspect sites and taken to a laboratory for analysis. The shorter lived radionuclides are analysed in situ.

The paper summarizes the detection of concealed nuclear reactor operations and reprocessing activities on the basis of environmental sample collection, including sampling of air particulates, water, sediments, biota, vegetation and soil.

2.2. Application of commercial environmental measurement and analysis techniques for the monitoring and detection of nuclear fuel cycle activities (IAEA-SM-333/96)

Many of the environmental monitoring techniques and technologies developed in the USA and internationally within the last decade could possibly be adapted for use in international safeguards inspections. These techniques have been designed to measure, with high accuracy and precision, organic, inorganic and radionuclide contaminants in air, soil, solids and water. The paper describes the initial effort to identify the various commercially available environmental measurement techniques and technologies and to catalogue them in a computer based international safeguards environmental measurements (ISEM) database. So far, 168 different instruments and 48 different techniques have been identified. ISEM currently includes data on sample mode, principles of operation, target analytes, size, weight, and many other parameters that are essential for optimal selection. Early results of assessing the potential vulnerabilities and countermeasures for environmental sampling and measurement techniques are also reported.

In the commercial environmental industry, techniques and technologies for monitoring and/or detecting organic and inorganic compounds and radionuclides in
the environment and measuring their chemical and physical properties have been developed and validated over many years.

One of the most important elements of environmental sampling and measurement is the planning process before any samples are acquired and analyses performed. This planning process determines the level of quality control required, the objectives of sample collection, the types of measurement to be performed, and the chosen target analytes. No single instrument or technique is capable of measuring all contaminants. During planning, the costs can be accurately established and signatures for comparative analysis designated. Also, the requirements for background determination, number of samples and confidence levels can be determined at this stage. Proven processes for planning result in lower costs, greater efficiency and high assurance that objectives can be met.

Finally, an initial examination of environmental sampling and measurement techniques has been performed. The authors conclude that the instruments and techniques designed for organic detection or analysis are more susceptible to compromise than those which are specially designed for radionuclide or inorganic measurements.

2.3. **Radionuclide analysis techniques for safeguards**

(IAEA-SM-333/123)

The paper considers the signatures of radionuclides that were released in reactor effluent water during the operation of the Hanford plutonium production reactors. The utility of these radionuclides for detecting covert nuclear reactor operations is evaluated. For reactors with a design similar to that of the Hanford reactors, which employed natural uranium and once-through cooling, a spectrum of radionuclides similar to those discharged from the Hanford reactors would be expected.

The concentrations of radionuclides in reactor discharge water, in various stretches of the Columbia River and in the Pacific Ocean are presented, and possible analytical techniques for their collection and measurement are discussed. The utility of the various reactor effluent radionuclides as signatures of reactor operation as a function of time since discharge is considered on the basis of the half-lives of the radionuclides and the relative ease of collection and analysis.

2.4. **Actinide determination and analytical support for the characterization of environmental samples**

(IAEA-SM-333/99)

The paper describes the development and operation at Los Alamos National Laboratory of a facility housing 'clean' chemical and instrumentation laboratories possessing very sensitive analytical techniques (utilizing thermal ionization mass spectrometry) for the determination of actinide elemental and isotopic concentrations in complex sample matrices. The facility and methodologies developed have been
used for a wide variety of applications, including very sensitive measurements on environmental actinides.

The integrated sample method used at the facility described in the paper extracts actinides from 2 L of water, 1-10 g of sediment and 1-20 g of soil. The paper describes the application of these techniques to environmental samples taken around the Rocky Flats Plant (RFP) of the United States Department of Energy near Denver, CO, which until 1989 conducted plutonium metallurgy activities related to the US nuclear weapons programme.

The almost overwhelming conclusion from these measurements is that 'clean' chemistry and mass spectrometry combined with a 'clean' sampling protocol are essential for maintaining field sample integrity. Continuous, performance based evaluation of individual laboratories is also critical for accurate data interpretation.

3. CONCLUSION

The empirical data given in the papers result from environmental sampling operations around very large nuclear material processing facilities associated with the US nuclear weapons programme. During most of the history of operations, these facilities were not required to comply with strict environmental control regulations, nor were they trying to mask or disguise their operations, which had the objective of producing thousands of nuclear weapons. The Hanford reactors were of a distinct design (natural uranium fuel, once-through cooling); this design may not be characteristic of what a potential proliferant would choose today.

Experience with both sampling and analysis procedures is discussed in the papers, as well as a variety of highly sensitive measurement tools which offer the possibility of detecting effluents from small facilities in the environment, although perhaps at more limited distances. While the awareness of potential proliferants that such tools may be used provides an incentive for undertaking countermeasures, it appears that the variety of effluent pathways, subsequent sampling opportunities, and the sensitivity of the available analytical measures create a significant potential for use of these measures in IAEA safeguards applications. Of course, cost and other factors must be addressed in determining the scope and frequency of using such measures.
ENVIRONMENTAL SAMPLING AND ANALYSIS AS A SAFEGUARDS TOOL

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Abstract

ENVIRONMENTAL SAMPLING AND ANALYSIS AS A SAFEGUARDS TOOL.

Environmental sampling and radionuclide analysis of the resulting material can be utilized as a supplemental approach in safeguarding practices and particularly for detection of undeclared nuclear activities. Production of nuclear weapons could be pursued by uranium enrichment processes to produce highly enriched $^{235}\text{U}$ or by nuclear reactor operations followed by chemical separations to produce $^{239}\text{Pu}$. The application of either of these processes results in the production of signature materials, some of which will be released to the environs. Results from the operations of the Hanford production facilities are discussed and indicate the type of signatures that may be expected from plutonium production facilities. These signatures include noble gas emissions from the reactors and chemical separation processes, the production of radionuclides in reactor cooling water, followed by their subsequent release to the Columbia River, and the release of mildly contaminated process water from the chemical processing facilities. These signature materials are carried by both gaseous and liquid effluents and enter various compartments of the environment. The types of signature materials that are most likely to be released and the environmental compartments where they are likely to be accumulated are discussed, and examples of the quantities that have been released during past separations are given. There are numerous processes by which natural uranium can be enriched to produce highly enriched $^{235}\text{U}$. The most definitive signature of such processes is always a modification in the uranium isotope ratios, and materials showing either enriched or depleted uranium in gaseous and liquid effluents provide the best indication that uranium enrichment processes are taking place. Therefore, techniques for sampling and analysis of airborne, waterborne or deposited uranium in environmental matrices provide a means of detecting uranium enrichment which may lead to products that can be used in proliferation activities.

1. INTRODUCTION

A safeguards programme for the detection of illicit nuclear weapons production could be enhanced by collection and analysis of environmental materials, including aerosols, water, soil and biota. For example, a proliferant State could pursue
nuclear weapons production based on uranium enrichment processes or via production of $^{239}$Pu, with their associated signatures [1]. Products and by-products from these activities would include modified uranium isotope ratios in the case of uranium enrichment, and numerous fission and neutron activation products in the case of plutonium production. A fraction of these signature radionuclides would enter the environment and would be concentrated by natural processes in the surface soils, in sediments, and in terrestrial and aquatic bioaccumulators.

The signature radionuclides associated with reactor operations include those in both gaseous and liquid effluents. The fission process results in the production of several noble gas radionuclides, as well as halogen radionuclides, which are relatively volatile and are more likely to be released. The spectra of relatively short lived radionuclides, such as $^{89}$Sr ($t_{1/2} = 50$ d), $^{140}$Ba ($t_{1/2} = 13$ d), $^{95}$Zr ($t_{1/2} = 64$ d), $^{95}$Nb ($t_{1/2} = 35$ d), $^{131}$I ($t_{1/2} = 8$ d) and $^{103}$Ru ($t_{1/2} = 40$ d), may be some of the first to be observed from plutonium production operation, and the longer lived fission process radioisotopes, $^{90}$Sr ($t_{1/2} = 29$ a), $^{144}$Ce ($t_{1/2} = 284$ d), $^{129}$I ($t_{1/2} = 10^6$ a), $^{106}$Ru ($t_{1/2} = 386$ d), and plutonium isotopes, may be more prevalent after longer periods.

In the case of the Hanford reactor operations, cooling water was discharged to the Columbia River, and several of the radionuclides became attached to particulate material and were deposited in sediments behind dams. Collection and analyses of these sediments provided an excellent means of detection of reactor operations.

Fuel reprocessing involves the removal of the fuel cladding, either by chemical or by physical means, followed by dissolution of the uranium fuel. During fuel dissolution, the volatile radionuclides are released from the dissolver tank together with oxides of nitrogen, and these gases pass through various types of filters and reactors to trap most of the fission and activation products. The radionuclides which are most readily emitted from reprocessing plants include $^{85}$Kr, $^3$H, $^{131}$I (if present), $^{129}$I, $^{99}$Tc, $^{14}$C, $^3$He, and, in some cases, particulate radionuclides such as $^{103}$Ru, $^{106}$Ru, $^{141}$Ce, $^{144}$Ce, $^{137}$Cs and $^{90}$Sr. A fraction of the radiiodine is released to the atmosphere during the dissolution process, and subsequently this fraction is eluted from sorption beds which are typically employed for its trapping. If production of high enriched $^{235}$U is the objective of the proliferator, the liquid and gaseous effluent may contain enriched and depleted uranium in chemical forms associated with the enrichment process. These signature materials may be detectable in air particulates, soils, groundwater and biota.

2. DISCUSSION

Effluent signatures are observable through sample collection and subsequent laboratory analyses, and in some cases they might be observed by real-time measurements. The fission process results in the production of several noble gas radio-
nuclides, as well as halogen radionuclides, which are relatively volatile, and these may be released during reactor operations. The noble gas radionuclides will always be released to some degree and, depending on the reactor confinement system, varying amounts of the halogen radionuclides will also be released. While sampling for the noble gas radionuclides is rather complicated, the collection of their daughter radionuclides can be accomplished easily by passing air through particulate filters. In conducting a survey at a suspect production site, radionuclides with half-lives of several hours or more could be collected for subsequent laboratory analysis; however, the very short lived radionuclides may require in situ analysis. For example, a large amount of $^{41}$Ar ($t_{1/2} = 1.82$ h) is produced from atmospheric argon ($^{40}$Ar), which may be dissolved in the cooling water or may simply be present as a component of the reactor cover gas. Plumes of $^{41}$Ar, and perhaps those of other radioactive gases, may be detectable with a high efficiency ground-based or airborne radiation sensor.

The major reactor effluent water radionuclides, together with their release rates from the world's first plutonium production reactor, the Hanford B reactor, during May 1960, are summarized in Refs [2-4]. With eight such reactors in operation, very large amounts of radioactivity entered the river, including up to 100 Ci/d $^1$ quantities of radionuclides such as $^{24}$Na, $^{32}$P, $^{45}$Ca, $^{46}$Sc, $^{56}$Mn, $^{64}$Cu, $^{76}$As, $^{51}$Cr, $^{60}$Co, $^{65}$Zn, $^{69}$Zn m, $^{72}$Ga, $^{89}$Sr, $^{90}$Sr, $^{131}$I, $^{140}$Ba, $^{140}$La, $^{152}$Eu m, $^{152}$Eu, $^{153}$Sm and $^{239}$Np. This spectrum of radionuclides was typical of that from the original eight plutonium production reactors at Hanford; however, the operation of the newer N reactor involved recirculation of the cooling water and discharging a few per cent to a trench that was located parallel to and approximately 200 m from the Columbia River. The radionuclide attenuation via soil adsorption, which occurred during the approximately six-day transit time from the trench to the river, ranged from two to five orders of magnitude. However, even with this approach, substantial quantities of several radionuclides did enter the Columbia River and could be observed throughout its entire downstream length.

When reactor effluent waters entered the Columbia River, several of the radionuclides became attached to particulate material and some were deposited in sediments behind the Columbia River dams. The physical forms of the reactor effluent water radionuclides and their concentrations throughout various reaches of the river over a one-year period are summarized in Ref. [2]. These data illustrate that, by separating radionuclides from a large volume of water, it is easy to detect reactor effluent discharges that have occurred at upstream locations.

Radionuclides which become associated with particulates within a river system may be deposited in the river bottom sediments, and collection and analysis of these sediments provide an excellent means of detecting reactor effluent discharges at upstream locations. There are several bioaccumulators of radionuclides which

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$^1$ 1 Ci = $3.7 \times 10^{10}$ Bq.
concentrate those present in river water. These bioaccumulators include terrestrial vegetation that is irrigated with river water and the associated food chain items, as well as aquatic biota within the river. The concentrations of radionuclides in river biota relative to those in river waters, and a summary of some of the radionuclide concentrations in four types of fish are presented in Refs [3, 4]. It is clear that sampling of river biota followed by laboratory analysis could provide an indication of upstream reactor operations.

After uranium fuel has been exposed in a nuclear reactor, the fuel is allowed to decay for a certain period of time prior to reprocessing for plutonium extraction. The decay interval depends on the period of time during which the country may be willing to wait for the extraction of plutonium. One of the major considerations in determining the decay period is the problem of release of volatile $^{131}$I ($t_{1/2} = 8$ d). If decay periods of a year or more are permitted, then essentially all of the $^{131}$I will have decayed. On the other hand, if decay periods of 3 months or less are permitted, then there will be a considerable amount of $^{131}$I present in the fuel and a substantial amount of this may be released to the environment.

As previously mentioned, when nitric acid is used in the dissolution of nuclear fuels, volatile radionuclides are released together with oxides of nitrogen. While these gases pass through various types of filters and traps, several radionuclides, including tritium, $^{131}$I (if present), $^{129}$I, $^{99}$Tc, $^{14}$C, $^{3}$He, and, in some cases, particulate radionuclides, such as $^{103}$Ru, $^{106}$Ru, $^{141}$Ce, $^{144}$Ce, $^{137}$Cs and $^{90}$Sr, are released. A fraction of the radiiodine is released to the atmosphere during the dissolution process, and subsequent to this process an additional release results from elution of the sorption beds which are employed for trapping of the radiiodine.

While $^{129}$I is a naturally occurring radionuclide, formed by spontaneous fission of $^{238}$U in the Earth's crust, its concentration at the Earth's surface, in vegetation and in natural waters is very low. Therefore, its measurement in environmental matrices provides an excellent indication of reprocessing activities. It can be measured with very high sensitivity by various types of laboratory techniques.

During the operation of the Hanford (Richland, WA) reprocessing plants, monitoring programmes [5–10] were carried out to measure the $^{129}$I in the atmosphere and in other compartments of the environment. It is clear from these studies that the concentrations of this radionuclide are some two orders of magnitude higher in the environs of Richland than on the Olympic Peninsula (500 km upwind), which is in the extreme north-west corner of the State of Washington. The concentrations of $^{129}$I in Spokane, WA (200 km downwind) are intermediate.

The observed concentrations of $^{129}$I in bovine thyroids, collected from meat packing firms throughout south-eastern Washington, indicated that by far the highest concentrations were from the area near the Hanford site. The concentrations in air, rain, grass and soil provided good indications of nuclear fuel reprocessing activities. Other types of vegetation, including perennial mosses and lichens, are good indicators of nuclear fuel reprocessing activities.
<table>
<thead>
<tr>
<th>Sampling</th>
<th>Signature</th>
<th>Collection technology</th>
<th>Limitations</th>
</tr>
</thead>
<tbody>
<tr>
<td>Air (from vehicles, stationary, aircraft)</td>
<td>Short lived noble gas daughters, radioiodines and tritium</td>
<td>LVAS(^a): using filter media and chemical sorption beds</td>
<td>Distance depends on release rates</td>
</tr>
<tr>
<td>Natural waters (from rivers, lakes and seawater)</td>
<td>Activation/fission products ((^{24})Na, (^{76})As, (^{32})P, (^{64})Cu, (^{65})Zn, (^{131})I, (^{140})Ba, (^{140})La, (^{152})Eu, (^{239})Np, etc.)</td>
<td>LVWS(^b): pumping water through ion exchange media</td>
<td>Effluents must enter natural waters</td>
</tr>
<tr>
<td>Sediments</td>
<td>Activation/fission products ((^{24})Na, (^{76})As, (^{32})P, (^{64})Cu, (^{65})Zn, (^{131})I, (^{140})Ba, (^{140})La, (^{152})Eu, (^{239})Np, etc.)</td>
<td>Sampling river, lake and ocean sediments</td>
<td>Effluents must enter natural waters</td>
</tr>
<tr>
<td>Biota (natural concentrations)</td>
<td>Activation/fission products ((^{24})Na, (^{76})As, (^{32})P, (^{64})Cu, (^{65})Zn, (^{131})I, (^{140})Ba, (^{140})La, (^{152})Eu, (^{239})Np, etc.)</td>
<td>Catching fish, oysters, etc.</td>
<td>Effluents must enter natural waters</td>
</tr>
<tr>
<td>Vegetation (terrestrial)</td>
<td>Activation/fission products ((^{24})Na, (^{76})As, (^{32})P, (^{64})Cu, (^{65})Zn (^{131})I, (^{140})Ba, (^{140})La, (^{152})Eu, (^{239})Np, etc.)</td>
<td>Cutting grass</td>
<td>Vegetation must be adjacent to facility</td>
</tr>
<tr>
<td>Soil samples</td>
<td>Activation/fission products ((^{24})Na, (^{76})As, (^{32})P, (^{64})Cu, (^{65})Zn (^{131})I, (^{140})Ba, (^{140})La, (^{152})Eu, (^{239})Np, etc.)</td>
<td>Collecting surface soils (top few millimetres)</td>
<td>Distance depends on source term</td>
</tr>
</tbody>
</table>

\(^a\) Large volume air particulate sampling.

\(^b\) Large volume water sampling.
## TABLE II. DETECTION OF REPROCESSING ACTIVITIES

Items collected for laboratory analysis

<table>
<thead>
<tr>
<th>Sampling</th>
<th>Signature</th>
<th>Collection technology</th>
<th>Limitations</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Air</strong>&lt;br&gt;(from vehicles, stationary)</td>
<td>Tritium, $^{131}$I, $^{129}$I, $^{99}$Tc, $^{103}$Ru/$^{106}$Ru, etc.</td>
<td>LVAS$^a$: using filters/sorption beds</td>
<td>Distance depends on release rates</td>
</tr>
<tr>
<td><strong>Natural waters</strong>&lt;br&gt;(from rivers, lakes and seawater)</td>
<td>Fission products&lt;br&gt;(tritium, $^{137}$Cs, $^{103}$Ru/$^{106}$Ru, $^{141}$Ce/$^{144}$Ce, etc.)</td>
<td>LVWS$^b$: using filters/sorption beds</td>
<td>Effluents must enter natural waters</td>
</tr>
<tr>
<td><strong>Subsurface waters</strong></td>
<td>Fission products&lt;br&gt;(tritium, $^{137}$Cs, $^{103}$Ru/$^{106}$Ru, $^{141}$Ce/$^{144}$Ce, etc.)</td>
<td>LVWS$^b$: using filters/sorption beds</td>
<td>Effluents must reach sampling wells</td>
</tr>
<tr>
<td><strong>Sediments</strong></td>
<td>Fission products&lt;br&gt;(tritium, $^{137}$Cs, $^{103}$Ru/$^{106}$Ru, $^{141}$Ce/$^{144}$Ce, etc.)</td>
<td>Collecting bottom sediments</td>
<td>Effluents must reach sampling wells</td>
</tr>
<tr>
<td><strong>Biota</strong>&lt;br&gt;(plankton/algae)</td>
<td>Fission products&lt;br&gt;(tritium, $^{137}$Cs, $^{103}$Ru/$^{106}$Ru, $^{141}$Ce/$^{144}$Ce, etc.)</td>
<td>Using plankton nets, algae growths</td>
<td>Biota must be available</td>
</tr>
<tr>
<td><strong>Vegetation</strong>&lt;br&gt;(terrestrial)</td>
<td>Fission products&lt;br&gt;(tritium, $^{137}$Cs, $^{103}$Ru/$^{106}$Ru, $^{141}$Ce/$^{144}$Ce, etc.)</td>
<td>Collecting grass, vegetation clippings</td>
<td>Vegetation must be adjacent to facility</td>
</tr>
<tr>
<td><strong>Soil samples</strong></td>
<td>Fission products&lt;br&gt;(tritium, $^{137}$Cs, $^{103}$Ru/$^{106}$Ru, $^{141}$Ce/$^{144}$Ce, etc.)</td>
<td>Collecting surface soils&lt;br&gt;(top millimetres over wide area)</td>
<td>Distance depends on release rates</td>
</tr>
</tbody>
</table>

$^a$ Large volume air particulate sampling.

$^b$ Large volume water sampling.
Nuclear fuel reprocessing at Hanford required huge amounts of water, which were used in the process itself for cleaning reprocessing equipment and washing down contaminated areas within the reprocessing cells. These waters, which contained relatively small amounts of fission and activation products, were discharged into the ground. The underground water plumes migrated down-gradient with the tritiated water in lead [11]. Collection and analysis of well water samples for tritium indicate its presence, as well as the presence of a wide range of fission and activation products, and could thus provide an indication of reprocessing activities at a suspect site.

On the basis of the properties and quantities of signature radionuclides that would be released to the environment as a result of reactor operations or fuel reprocessing, it is possible to select sampling strategies that should permit early detection of these activities. Tables I and II summarize the detection of concealed nuclear reactor operations and reprocessing activities on the basis of sample collections. These collections involve sampling of air, water, sediments, biota, vegetation and soil. Their analyses would involve returning the samples to the laboratory for the measurement of radionuclides of the types indicated. Most of these collections could be made in a relatively straightforward manner with a minimum of special equipment, so that costs could be minimal and confidence in detecting signature materials indicative of proliferation activities could be high. Examples of the release, distribution and accumulation of the radionuclides have been discussed, and sampling scenarios are summarized below.

3. SAMPLING SCENARIOS

3.1. Air sampling

In order to ensure a high sensitivity for the measurement of airborne radionuclides, it is important to employ large volume air sampling (LVAS). Sampling at high rates (tens of cubic metres per minute) improves the detection probability. However, sampling at lower rates could be satisfactory if one were near (within a few kilometres) the suspect proliferation site. Particulate radionuclides are readily captured on efficient air filters that have relatively low flow resistance, and the iodine radionuclides can be trapped on charcoal impregnated filters or other filter matrices. The subsequent analyses of these materials in the laboratory would indicate the presence of modified uranium isotope ratios or radionuclides associated with reactor operations and fuel reprocessing, as indicated in Tables I and II.

3.2. Water sampling

Aqueous effluents from nuclear activities are generally discharged to either surface or subsurface waters. They can be readily separated from the water, using
large volume water sampling techniques [12]. Large volume water samplers have been used for the past three decades by Pacific Northwest Laboratory for collection of radionuclides in the Hanford environs, through the length of the Columbia River, and for hundreds of miles out to sea. With this system, the water is pumped at a rate of 30–40 L/min through filters for collection of particulate radionuclides and through ion exchange or sorption beds for collection of dissolved radionuclides (see Tables I and II).

3.3. Sediments

As mentioned above, several of the radionuclides associated with uranium enrichment and reactor operations and reprocessing would be deposited in sediments of water bodies to which they are discharged. In order to provide a large amount of sediment for analysis, cores of 15 cm diameter are collected from slack water reaches of rivers; however, almost any kind of sediment collector would be satisfactory for collecting sediment materials for laboratory analysis.

3.4. Biota

Essentially all biota from fresh waters to which radionuclides are discharged will concentrate the radionuclides by a few orders of magnitude. Filter feeders, such as oysters, mussels and clams, are particularly good bioconcentrators and may concentrate radionuclides or other trace constituents by four to five orders of magnitude. Their collection and subsequent laboratory analysis provide a good indication of nuclear operations.

3.5. Vegetation

Where radionuclides are released to the atmosphere, vegetation is a particularly good collector for the detection of the processes that are responsible for these releases. Grass clippings are particularly good, since they may contain material from recent releases. Other types of vegetation which require longer growing periods may also be useful. Examples of vegetation commonly used for measurement of deposited radionuclides are given in Section 2.

3.6. Soils

Airborne radionuclides, except for the noble gases, are eventually deposited on the Earth’s surface, and concentrations are highest near the points of release. To achieve the highest sensitivity in the detection of radionuclides that may be deposited from the atmosphere, samples should be collected from a minimal soil thickness over a wide area for subsequent laboratory analysis.
3.7. Detection of radionuclides by aircraft flights

The operation of nuclear reactors and fuel reprocessing plants results in the release of radionuclides to the atmosphere. The radionuclides most likely to be released would be noble gas fission products and, in the case of reactor operations, $^{41}\text{Ar}$. An aircraft flying through areas downwind from places of suspected nuclear activities could possibly observe the increased radionuclide levels on a real-time basis. To maximize the detection sensitivity, large areas and/or high resolution sensors should be employed. While reconnaissance flights of this type would be rather expensive, they could be useful in surveying a number of sites in a relatively short period of time. Also, these flights could incorporate sampling procedures to collect both radioactive gases and particulates for laboratory analysis.

4. SUMMARY

It is possible to detect uranium enrichment processes, nuclear reactor operations and fuel reprocessing activities on the basis of environmental sampling and analysis. The signature materials that may be the best indicators of these processes and their collection techniques are summarized in Tables I and II.

REFERENCES


APPLICATION OF COMMERCIAL ENVIRONMENTAL MEASUREMENT AND ANALYSIS TECHNIQUES FOR THE MONITORING AND DETECTION OF NUCLEAR FUEL CYCLE ACTIVITIES*

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Presented by I.N. Goldman

Abstract
APPLICATION OF COMMERCIAL ENVIRONMENTAL MEASUREMENT AND ANALYSIS TECHNIQUES FOR THE MONITORING AND DETECTION OF NUCLEAR FUEL CYCLE ACTIVITIES.

Recent nuclear safeguards inspections by IAEA personnel have led to the conclusion that there is a need for proven environmental sampling and monitoring techniques that complement traditional IAEA safeguards inspection tools. These techniques and technologies can help verify operations as declared and monitor for undeclared nuclear fuel cycle/nuclear weapons related operations. Environmental monitoring is a relatively new mission of the IAEA. It carries broad support by the United Nations following the discovery in Iraq of undeclared nuclear weapons design and materials production activities. Many of the environmental monitoring techniques and technologies developed in the United States of America and internationally within the last decade could possibly be adapted for use in international safeguards inspections. Techniques designed to measure or monitor unique organic/inorganic signatures from nuclear materials production or weapons manufacturing, effluents in air, water and soil are just now being evaluated for use as complementary safeguards tools. These techniques are used by industry to locate and detect environmental contamination and the source of such contamination. These same commercially available techniques and methods, if properly adapted to the task of inspecting, monitoring and searching for nuclear fuel cycle/weapons production activities, can greatly increase the likelihood of success in concluding that operations are as declared and that clandestine operations do not exist or have not been resumed. Furthermore,

* This work was performed for the United States Department of Energy and the United States Department of State under Contract No. 9XY35327H-1 with Los Alamos National Laboratory.
in evaluating the applicability of commercial environmental techniques in international safeguards inspections, a thorough understanding of the potential vulnerabilities and susceptibility to countermeasures is required; otherwise, techniques that could be defeated might be employed and inaccurate conclusions reached. The paper discusses studies undertaken to determine the applicability of existing techniques to international nuclear safeguards inspections. The preparation of the international safeguards environmental measurements database is described, and the results of a preliminary assessment of vulnerabilities and countermeasures of environmental sampling and measurement techniques are discussed.

1. INTRODUCTION

Environmental protection has become an issue on every continent and in every country. As the scope of environmental issues has grown to cover all media — air, water and solids — and trace to ultra-trace constituents, the need for accurate and sensitive measurement techniques has also grown [1]. Data from an environmental measurements programme benefit industry in several ways. In some areas, these data are required as evidence of regulatory compliance and for the issue of permits. They are also needed for effective selection and design of pollution control or reduction strategies.

On behalf of the US Department of Energy and the US Department of State, Los Alamos National Laboratory contracted Radian Corporation to evaluate the application of commercially available environmental measurement techniques to detect and monitor nuclear fuel cycle activities. Radian Corporation is an international engineering and consulting firm with special expertise in environmental testing and monitoring. This paper summarizes the types of environmental measurements routinely performed in our industry and their potential applications to safeguards measurements.

2. ENVIRONMENTAL MEASUREMENTS

In the USA, the growth of environmental science mirrors that of environmental regulations. Major legislative acts that have resulted in the regulation of effluents and emissions are the Clean Air Act, the Clean Water Act and the Resource Conservation and Recovery Act. The following sections describe measurement techniques for air, water and solids. These environmental approaches to measuring physical and chemical properties have been developed and validated over several years in the environmental industry. Many of these principles and practices can be used to identify effluents from nuclear fuel cycle and nuclear weapons manufacturing facilities.
TABLE I. EXAMPLE PLANNING MATRIX FOR SELECTING THE MAJOR CATEGORY OF MONITORING

<table>
<thead>
<tr>
<th>Primary objective</th>
<th>Stream</th>
<th>Frequency</th>
<th>Location</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Source</td>
<td>Ambient</td>
<td>Intermittent</td>
</tr>
<tr>
<td>Identify emission sources</td>
<td>✓</td>
<td>×</td>
<td>≈</td>
</tr>
<tr>
<td>Detect transient emissions</td>
<td>≈</td>
<td>≈</td>
<td>×</td>
</tr>
<tr>
<td>Detect inadvertent releases</td>
<td>≈</td>
<td>✓</td>
<td>×</td>
</tr>
<tr>
<td>Track effluent composition</td>
<td>×</td>
<td>✓</td>
<td>≈</td>
</tr>
</tbody>
</table>

✓✓ — Required or highly preferred;
✓ — Preferred or likely to be used;
× × — Not permitted or not available;
× — Discouraged or unlikely to be used;
≈ — Neutral, either is acceptable.
2.1. Scope of environmental measurements

The major steps involved in any environmental testing and measurement project are illustrated in Fig. 1. Typically, scientific research tends to focus on the sampling and analytical aspects of the process. However, planning (design), data handling and quality control can greatly influence the results from sampling and analytical efforts. Therefore, the environmental industry has developed protocols and tools to aid in all aspects of environmental measurements, not just sampling and analysis.

2.2. Planning

Selecting the type of sampling and measurement technique that should be used is crucial to planning. Table I is an example planning matrix for selecting the major category of monitoring, either source or ambient, intermittent or continuous, or laboratory or field analysis. The choice of sampling and analytical techniques depends largely on the primary objective, but is also influenced by logistics and other practical issues. For example, if the objective is to "identify an emission source", then a monitoring station at the source is desirable. However, source sampling is often impossible (e.g. the source stream is intermittent, or the source is inaccessible). In this case, ambient sampling must be performed at some remote location. An ambient monitoring station would indicate whether an emission had occurred, but source identification would be less accurate. Also, with ambient sampling, accurate meteorological (for air monitoring) or hydrogeological (for surface or groundwater monitoring) data must be collected, together with the ambient stream data. These considerations and the primary objectives given in Table I can generally be applied to the case of nuclear safeguards as well.

Planning includes developing a list of target constituents, or analytes. In the environmental arena, industry often pools its test data to characterize a given process, thus reducing the amount of testing that any one company will be required to undertake. The pooled data provide an effluent profile for the industry that serves as a 'signature' for identifying a select group of effluent constituents. Similar information that would identify a select group of effluent constituents for the safeguards mission can be obtained from known nuclear weapons component or material production facilities.

Establishing background concentrations of constituents of concern must also be considered during the planning process. Contaminated samples can be distinguished from background samples by the level of the constituents present. This is not always straightforward, especially if residual concentrations of the species of concern are present in background samples. In this case, a greater number of samples may need to be taken to verify whether there is a difference in concentration between the sample of concern and background levels. The number of background and target
samples needed will depend on three variables: (i) the threshold or background value, (ii) an estimate of the range or variance in sample concentrations, and (iii) the level of confidence desired in the final answer. Given these variables, the number of samples required can be determined with a stated statistical confidence.

Another approach that can be used to distinguish background samples from contaminated samples is to compare constituent ratios. Often, the ratio of one constituent to another (or a set of other constituents) will be different in the contaminated samples. For example, the isotopic ratio of a natural chemical versus an anthropogenic chemical can differ greatly, especially if the chemical has undergone a great deal of processing. Isotopic ratio analysis is excellent for identifying samples affected by nuclear fuel cycle activities. In this case, the processes are often designed to alter the isotopic ratio of a radionuclide.

2.3. Measuring air emissions

The technologies for sampling and analysing air are many and varied. The choice of methods depends largely on the source. Stack emissions can be measured using standard sampling methods developed by the US Environmental Protection Agency (EPA) [2]. Fugitive emission tests are needed to detect emissions from equipment, such as pumps, pipes or valves. If the amounts and concentrations of emissions are needed from a large area, then wide-area or ambient monitoring techniques are preferred.

Source sampling techniques are well developed for criteria pollutants. Typically, the methods include capturing a slipstream of the stack gas with a probe and then selectively capturing constituents in the gas with filters, sorbents and impingers. If continuous stack monitoring is needed, then the use of continuous emission monitors (CEMs) is another option [3].

Plant equipment may leak or routinely emit chemical constituents. Emissions from equipment are measured by portable hydrocarbon analysers at potential leak points. A recent survey lists the types of analysers available for screening equipment leaks [4].

Measuring the concentrations of air pollutants in a large area presents a different set of objectives and problems. For direct measurements, ambient air is sampled at various points in the area of emissions. One method of direct measurement is the use of an emission flux chamber. Emissions are captured and mixed with a carrier gas for direct analysis in the field. If an overall emission flux rate is needed for an area, then several flux chamber measurements are taken, and a flux rate can be calculated.

Indirect measurement methods sample ambient air at points upwind and downwind of the source. Indirect measurements are advantageous for measuring emission rates from large, heterogeneous sources. Another advantage is that personnel and equipment do not come into contact with the source. Indirect, or wide-
<table>
<thead>
<tr>
<th>Method</th>
<th>Operating principle</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fourier transform infrared spectroscopy</td>
<td>A broadband source allows collection and analysis of a full IR spectrum. Standard spectra are compared with field readings to determine constituents and concentrations.</td>
</tr>
<tr>
<td>Ultraviolet (UV) spectroscopy</td>
<td>UV spectra are collected over a limited absorption spectral region. The differential absorptions of the compounds in the air are used to determine the identity and concentrations of contaminants.</td>
</tr>
<tr>
<td>Gas filter correlation</td>
<td>A sample of the gas to be detected is used as a reference. The correlation between the spectrum of the sample gas is compared with the gas in the measurement path to determine concentration.</td>
</tr>
<tr>
<td>Filtered band-pass absorption</td>
<td>Absorption of the gas is measured in certain bands to detect composition and concentration.</td>
</tr>
<tr>
<td>Laser absorption</td>
<td>One or more lasers are used to measure absorption at different wavelengths.</td>
</tr>
<tr>
<td>Photoacoustic spectroscopy</td>
<td>The pressure change from the deactivation of excited molecules is measured in a closed acoustic chamber.</td>
</tr>
<tr>
<td>Laser intensity direction and ranging</td>
<td>Molecular or aerosol backscatter are measured by either differential absorption or Raman scattering to identify gases and determine their concentrations. Unlike the other methods listed here, this method can provide ranging information on measurements.</td>
</tr>
<tr>
<td>Diode laser spectroscopy</td>
<td>A developing technology for open-air use; a line feature of the gas of interest is spectrally scanned to identify and quantify the compound.</td>
</tr>
</tbody>
</table>
area, monitoring relies heavily on meteorological data to define the dispersion of pollutants and potential population exposure. The fission process emits several radio-nuclides, which may be sampled using high volume air samplers.

Recent advances in instrumentation have increased the use of remote sensing techniques for indirect wide-area monitoring [5]. Table II describes several remote methods. Of the techniques listed, Fourier transform infrared spectroscopy, ultraviolet spectroscopy and gas filter correlation are the most common.

2.4. Water testing

Common collection points for water streams include post-treatment wastewater, process wastewater, stormwater, groundwater and spills. Water testing can consist of conventional sampling and laboratory analysis, sampling and analysis in the field, or continuous monitoring. Continuous methods can measure flow rate, temperature, pH, conductivity, dissolved oxygen, turbidity, total oxygen demand, cyanide, and total suspended solids. In measuring these parameters, the sensors are immersed directly into the water stream and the sensor signals are sent to a remote monitor. The monitor converts the analogue input to digital output signals proportional to the parameter or value. Water monitoring stations can have automatic samplers with rain detectors and depth sensors that will collect samples automatically at a given time interval. Aqueous effluents from nuclear activities have been monitored at various sites using high volume (30–40 L/min) water samplers [6].

2.5. Solid waste testing

Unlike air and water, solid wastes are not necessarily generated in a continuous stream. Consequently, sampling becomes critical in determining the concentration of pollutants. Several solid waste testing guides are available [7].

Solid wastes are most commonly sampled directly at the point of generation, such as a pipe or discharge point, or at a transfer or collection point, such as a silo, tank or thickener. It is often impossible to sample the waste as it is generated. In these cases, a sample must be obtained from a treatment, storage and disposal facility, which presents a different challenge. The simplest sampling devices are often all that are needed for solid sample collection: scoops, dippers and bottles. However, in many situations, the sample device must be able to extract a sample at a given depth or location without the sample becoming contaminated from other locations. Special sampling devices have been designed for many of these instances.

2.6. Analysis

Several instrumental methods are available for environmental analysis. The criteria for selecting a particular method include confidence level and sensitivity
required, potential chemical interferences, applicability of the method to the matrix and availability of the instrument.

Regardless of the method chosen, the most critical step before analysis is sample preparation. For the analysis, composing, subsampling, extracting and digesting the sample are often as critical as, or even more critical than, the measurement technique itself.

Separation methods are used for organic analysis (chromatography followed by detection). Capillary column gas chromatography has higher resolution than packed column gas chromatography, and the application will determine the required sensitivity of the detector (mass spectrometry, electron capture, flame ionization, or photo-ionization detectors).

Atomic absorption and emission spectrophotometry are most common for inorganic analyses. Plasma emission is especially useful in that it can detect multiple analytes and can be applied to both liquid and solid analyses.

2.7. Data handling

A comprehensive database will not only enhance the reporting capabilities from environmental measurements but can also assist in co-ordinating sample handling, data quality control and other aspects of data handling. In addition to databases, many other techniques and approaches used by the commercial environmental industry can be readily adopted to support the application of sampling and analysis systems for the IAEA. Some of these other techniques and approaches include:

(a) *Data quality objectives*, with precision and accuracy goals, and acceptance limits.
(b) *Detailed sampling design*, including number and location of samples, and selection of analyses.
(c) *Sample control*, including sample labels, preservation and holding times, routing, and sample disposal.
(d) *Analytical quality control*, in the form of system audits; control checks, replicates, spikes and blanks; corrective action; and mass balance or charge balance.
(e) *Data management*, including data acquisition, verification, data reduction (statistical, graphical, visualization), and reporting.
(f) *Process evaluation*.

There are also standard techniques used in the commercial industry to establish characteristic environmental signatures (e.g. effluent or emission characterizations). To set effluent guidelines, for example under the National Pollutant Discharge Elimination System, EPA first conducts an ‘effluent survey’ of representative facilities in the industry of interest. The effluent survey includes the following steps:
— Establishing process flow diagrams;
— Developing monitoring plans;
— Sampling and analysing effluents (5–15 facilities);
— Analysing and collating data; and
— Establishing the industry profile.

The profile, which is characteristic of the particular industrial process, is used to set effluent emission standards. These techniques would be useful in providing needed signature databases for assessing operations as declared or evaluating whether an undeclared facility was possibly being detected.

3. APPLICATIONS TO IAEA SAFEGUARDS

This paper focuses on identifying and evaluating commercially available environmental measurement techniques, technologies and instruments for possible use in international safeguards inspection, monitoring and search activities. Each technique and instrument identified must be compared against a list of target analytes to validate its usefulness. Table III presents a partial list of analytes. Using the identified techniques could help international inspectors verify operations as declared or detect the existence or historical presence of clandestine/undeclared nuclear materials production and/or nuclear weapons manufacturing and design activities.

3.1. Environmental measurement techniques database

As part of the effort to identify applications of environmental measurements to the safeguards arena, a database of measurement methods has been established. The database, known as the international safeguards environmental measurements (ISEM) database, catalogues information on the various commercial environmental measurement techniques and instruments by using several tables. Currently, the tables include sampling techniques, measurement techniques, instruments, manufacturers, references, and several relation definition tables [8]. The sampling techniques table includes fields such as sampling mode, sample matrix, target analyte, principle of operation, and intrusiveness. The measurement techniques table includes accuracy, precision and mitigating factors. The instrument table includes the manufacturer's name, and the size, weight, power requirements, etc. The current database will permit identification of potential sampling and measurement techniques and instruments for a specific requirement.

A goal for the database is to allow inspectors or safeguards planners to easily access information on applicable methods and measurement techniques. For example, the data can be searched by processing operations, which will lead to target analytes and appropriate sampling and analytical methods. The database provides
<table>
<thead>
<tr>
<th>Production step</th>
<th>Potential target analytes</th>
</tr>
</thead>
</table>
| Facilities construction and/or modification | High density concrete components: barite, ilmenite, magnetite, haematite, ferrophosphorus, iron shot  
Neutron shielding components: boron, cadmium  
High density glass components: barium oxide, lead oxide  
Shielded windows cavities: lead acetate, zinc bromide, barium bromide |
| Uranium enrichment | Gas diffusion: U fluoride compounds, hydrofluoric acid, isotopic uranium  
Laser photoionization: U fluoride compounds, hydrofluoric acid, isotopic uranium, laser dyes (Rhodamine 6G) |
| Fuel reprocessing | Purex process: tributylphosphate (TBP), kerosene, HTO, nitric acid, hydrofluoric acid  
Bismuth phosphate: nitric acid, hydrofluoric acid  
Other: ion exchange degradation products, I₂ gaseous effluents |
| Plutonium or uranium metal production | Calcium compounds, nitric acid, hydrofluoric acid |
| High explosives production | HMX, RDX, TNT, TATB |
| Component machining and assembly | Oils or cooling solutions containing uranium, plutonium or beryllium |
sufficient detail to allow the user to make informed decisions regarding the appropriateness of the methods to their use. References are included in the database for greater detail on the methods and instruments.

The database may be expanded to include detailed analyte descriptions, reference signatures, instrument inventory, expertise inventory, vendors, service providers, capital and operating cost data, instrument performance parameters, and geographic and environmental data. Perhaps most importantly, an expanded database could include information on inspection results. Artificial signature recognition intelligence could use the expanded database to help determine the likelihood of 'operations as declared' by comparing reference signatures, historical data, current data, and information on the regional environment.

In addition, mission planning could be aided by using the expanded database to help select appropriate sampling and measurement techniques, equipment and instruments, supplies and personnel. Sampling instructions, chain-of-custody forms, sample labels, sample logs and analysis result forms could all be easily generated. Inspectors and analysts would benefit from the availability of information and the high level of quality control and quality assurance.

Important to the use of the ISEM database and eventually to the use and choice of any instrument or technique is the potential vulnerability or susceptibility of a chosen technique to compromise on defeat. From preliminary data it is concluded that the instruments and techniques designed for organic detection and analyses are more susceptible to compromise than those optimized for radionuclide or inorganic measurements. The degree of susceptibility also depends on the distance from source detection objectives, the sensitivity to be achieved and the background concentration. More work is required in this area before environmental sampling is fully adopted as an inspection tool.

4. SUMMARY

A comprehensive review of commercial environmental monitoring and measurement techniques has been performed and their applicability to detecting effluents from nuclear fuel cycle and weapon production facilities evaluated. The information assembled has been organized into a relational database that allows a user to identify and select techniques and instruments capable of detecting an analyte of interest that could indicate fuel cycle activities or weapons production. Using commercial environmental monitoring and measurement techniques can help detect undeclared and nuclear fuel cycle or weapons production operations, and verify operations as declared. However, to use these techniques successfully, careful planning is required to determine where, how and when to take samples, define and follow quality control and quality assurance procedures, and to ensure that effluent characteristics are well known. Radian Corporation is continuing to evaluate the usefulness of these techniques to aid in detecting nuclear fuel cycle and weapons production activities.
REFERENCES


RADIONUCLIDE ANALYSIS TECHNIQUES FOR SAFEGUARDS

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Presented by I.N. Goldman

Abstract
RADIONUCLIDE ANALYSIS TECHNIQUES FOR SAFEGUARDS.

The signatures of radionuclides that were released in reactor effluent water during the operation of the Hanford reactors are considered and their utility for detecting covert nuclear reactor operations is discussed. For reactors with a design similar to that of the Hanford reactors, which employed natural uranium and once-through cooling, a spectrum of radionuclides similar to those discharged from the Hanford reactors would be expected. The concentrations of radionuclides in reactor discharge water, in various stretches of the Columbia River and in the Pacific Ocean are presented, and possible analytical techniques for their collection and measurement are discussed. The utility of the various reactor effluent radionuclides as signatures of reactor operation as a function of time since discharge is considered on the basis of the half-lives of the radionuclides and the relative ease of collection and analysis.

1. INTRODUCTION

Some of the most significant and easily observable indicators of plutonium production at the Hanford site were the neutron activation and fission products that were present in the effluent water. The eight original plutonium production reactors used purified river water as coolant in a once-through cycle, and this water was subsequently discharged to the Columbia River after a cooling period of a few hours. On entering the Columbia River, the water was diluted several hundredfold and interacted with suspended or bottom sediments.

About 600 km from the Hanford site the Columbia River discharges itself into the Pacific Ocean where its waters are transported, dispersed and diluted. A major concern in the operation of the Hanford site was the minimization of the amounts of radionuclides released and the understanding of their subsequent environmental pathways. In the case of reactor effluent water, it was important to understand the production mechanisms of the radionuclides in the reactors and their behaviour in the Columbia River and ocean water ecosystems. During the reactor operation
period, analytical methods were developed to measure the quantities and the physical and chemical properties of the radionuclides at discharge and in the environs of the Columbia River and the Pacific Ocean.

In this paper the emphasis is on reviewing the spectrum of radionuclides that were released from the Hanford reactors into the Columbia River, and the sampling and analytical techniques that were developed and employed, and evaluating these as possible proliferation detection methods. The following points are discussed: (a) the types of radionuclides in the reactor effluent water that were released, (b) the concentration of the physical forms of radionuclides present in effluent water and in the Columbia River, (c) the relevant analytical methods used for the collection and measurement of these radionuclides, and (d) the signature radionuclides that could be observed most easily in river waters and in the ocean.

2. DISCUSSION

One of the main environmental and health concerns in the operation of the Hanford reactors was the formation of certain radionuclides in the cooling water and their discharge into the Columbia River. The Hanford reactors employed once-through cooling with purified Columbia River water, and the effluent water was discharged into retention basins where it was held for 2–4 hours before it entered the Columbia River. This delay period allowed the very short lived radionuclides to decay to insignificant quantities, but the concentrations of the radionuclides with half-lives of several hours or more were little affected.

A disappointing observation, made during the early operation of the reactors, was that the radionuclide content in the effluent water was in all cases higher than that which had been predicted on the basis of the residence time of the cooling water and its elemental constituents within the flux region of the reactor; in some cases the actual radionuclide content was several orders of magnitude higher than predicted. Various attempts were made to understand this phenomenon, but a complete description of the process was not available until 1961, when one of the present authors wrote a report in which he summarized the source of radionuclides in reactor effluent water [1].

The discrepancy between the observed radionuclides in reactor effluent water and the values calculated on the basis of the content of target elements in the cooling water for one of the early Hanford reactors is summarized in Table I. The important point is that the actual concentrations of all radionuclides were substantially higher than the concentrations calculated on the basis of the parent element concentrations in the reactor cooling water and the neutron flux to which they were exposed. This point is further clarified in Table II, where the 'average' hold-up time for the various parent elements is shown. Since the cooling water was actually present in the neutron flux region of the reactor for only 0.83 s (see Table I), the average 'hold-up' time
<table>
<thead>
<tr>
<th>Radioisotope</th>
<th>From 5 ppm H₂SO₄</th>
<th>From 10 ppm Al₂(SO₄)₃ 18H₂O</th>
<th>From 2 ppm Na₂Cr₂O₇</th>
<th>Normal process water</th>
<th>Normal process water</th>
</tr>
</thead>
<tbody>
<tr>
<td>Na-24</td>
<td></td>
<td>88</td>
<td></td>
<td>320</td>
<td>1500</td>
</tr>
<tr>
<td>P-32</td>
<td></td>
<td>0.72</td>
<td>0.014</td>
<td>0.33</td>
<td>38</td>
</tr>
<tr>
<td>Sc-46</td>
<td>0.000014</td>
<td>2.20</td>
<td>0.00012</td>
<td>2.20</td>
<td>1700</td>
</tr>
<tr>
<td>Cr-51</td>
<td>1.6</td>
<td>19</td>
<td></td>
<td>10000</td>
<td></td>
</tr>
<tr>
<td>Mn-56</td>
<td>0.00015</td>
<td>19</td>
<td>0.00082</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>Co-60</td>
<td>0.0000016</td>
<td>0.0000018</td>
<td>0.0000073</td>
<td>0.30</td>
<td></td>
</tr>
<tr>
<td>Cu-64</td>
<td>0.085</td>
<td>3.35</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Zn-65</td>
<td>0.048</td>
<td>0.0017</td>
<td></td>
<td>37</td>
<td></td>
</tr>
<tr>
<td>Ga-72</td>
<td>0.0033</td>
<td>0.027</td>
<td></td>
<td>64</td>
<td></td>
</tr>
<tr>
<td>As-76</td>
<td>0.000019</td>
<td>0.000011</td>
<td></td>
<td>0.30</td>
<td></td>
</tr>
<tr>
<td>La-140</td>
<td>0.000095</td>
<td>0.031</td>
<td></td>
<td>3.3</td>
<td></td>
</tr>
<tr>
<td>Sm-153</td>
<td>0.0021</td>
<td>0.0019</td>
<td></td>
<td>13</td>
<td></td>
</tr>
<tr>
<td>Np-239</td>
<td>0.0010</td>
<td>0.041</td>
<td></td>
<td>550</td>
<td></td>
</tr>
</tbody>
</table>

a P-32 is from the P-31 parent.
b P-32 is from the S-32 parent.
### Table II. Minimum Percentage of Parent Element Adsorbed on the Process Tube Film to Produce the Observed Effluent Water Radioactivity* and Hold-up Time*<sup>b</sup>

<table>
<thead>
<tr>
<th>Parent isotope</th>
<th>Daughter radioisotope</th>
<th>Minimum percentage of uptake</th>
<th>Hold-up time (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Na-23</td>
<td>Na-24</td>
<td>0.0051</td>
<td>3.8</td>
</tr>
<tr>
<td>P-31&lt;sup&gt;c&lt;/sup&gt;</td>
<td>P-32</td>
<td>0.13</td>
<td>3 200</td>
</tr>
<tr>
<td>S-32&lt;sup&gt;c&lt;/sup&gt;</td>
<td>S-32</td>
<td>0.00054</td>
<td>9.6</td>
</tr>
<tr>
<td>Sc-45</td>
<td>Sc-46</td>
<td>0.24</td>
<td>26 000</td>
</tr>
<tr>
<td>Cr-50</td>
<td>Cr-51</td>
<td>0.018</td>
<td>640</td>
</tr>
<tr>
<td>Mn-55</td>
<td>Mn-56</td>
<td>3.3</td>
<td>440</td>
</tr>
<tr>
<td>Fe-58</td>
<td>Fe-59</td>
<td>0.018</td>
<td>1 000</td>
</tr>
<tr>
<td>Co-59</td>
<td>Co-60</td>
<td>0.0014</td>
<td>3 400</td>
</tr>
<tr>
<td>Cu-63</td>
<td>Cu-64</td>
<td>0.48</td>
<td>320</td>
</tr>
<tr>
<td>Zn-64</td>
<td>Zn-65</td>
<td>0.060</td>
<td>18 000</td>
</tr>
<tr>
<td>Ga-71</td>
<td>Ga-72</td>
<td>2.7</td>
<td>2 000</td>
</tr>
<tr>
<td>As-75</td>
<td>As-76</td>
<td>1.7</td>
<td>2 300</td>
</tr>
<tr>
<td>Sb-123</td>
<td>Sb-124</td>
<td>0.031</td>
<td>2 300</td>
</tr>
<tr>
<td>La-139</td>
<td>La-140</td>
<td>0.042</td>
<td>88</td>
</tr>
<tr>
<td>Sm-152</td>
<td>Sm-153</td>
<td>2.4</td>
<td>5 700</td>
</tr>
<tr>
<td>U-238</td>
<td>Np-239</td>
<td>4.6</td>
<td>13 000</td>
</tr>
</tbody>
</table>

* It is assumed that the aluminium corrosion products are not reflected from the film.

*<sup>b</sup> This is the irradiation time required to produce the observed radioactivity.

*<sup>c</sup> In each case, it is assumed that all P-32 in the effluent water is from the particular parent.

of a parent element such as <sup>45</sup>Sc, which on neutron capture produces <sup>46</sup>Sc (83.81 d half-life), was over 26 000 min. The shortest apparent hold-up period was that of <sup>23</sup>Na, a parent element of <sup>24</sup>Na (14.96 h half-life). While there is some additional information that should be considered for <sup>24</sup>Na and some other parent isotopes which contribute somewhat to the radionuclide concentration, as indicated in Ref. [1], the above values clearly illustrate the unexpected high concentrations of many radionuclides in reactor effluent water.

The relatively high concentrations of the various radionuclides as shown in Table I indicate that their measurement in effluent water would be relatively easy, and, on dilution in the Columbia River by a few hundred times, their concentrations could still be observed if large volumes of water (5-20 L) were collected, evaporated to a small volume and measured with a high efficiency γ ray spectrometer.
FIG. 1. Radionuclide concentration in the Columbia River at Pasco, in the Hood River and at Vancouver for the period January 1964 to January 1965.
A study of the transport and the behaviour of eleven of the longer lived reactor effluent water radionuclides in the Columbia River was conducted in 1964 and 1965 [2]. Figure 1 shows the concentrations observed in this study, which were based on monthly measurements, for three locations on the Columbia River. Over a period of one year, the concentrations fluctuated by approximately an order of magnitude, depending on the flow velocity of the Columbia River.

Figure 2 shows the physical forms of the radionuclides. Some radionuclides, such as $^{46}$Sc and $^{59}$Fe, were mainly in particulate form; other radionuclides, such
FIG. 3. Concentrations of $^{51}\text{Cr}$ collected at 11 sampling places off the west coast of the USA from 13 to 16 November 1968 (dis/min per 1000 L).

as $^{51}\text{Cr}$ and $^{124}\text{Sb}$, were mainly in dissolved form. Regarding the utility of the various radionuclides as indicators of an upstream reactor operation, it is evident that such an operation could be detected by simply filtering large volumes of river water and measuring the particulate radionuclides collected on the filters. Also, in fresh water it is relatively easy to capture dissolved cationic and anionic species on ion exchange resins. Therefore, a sampling system for large volumes of water [3, 4], such as that developed by Battelle Pacific Northwest Laboratory, which uses ion...
exchange beds through which water can be pumped at rates of up to 40 L/min, could provide large samples for subsequent laboratory analyses, and these could be measured by direct $\gamma$ ray spectrometry of the material from the ion exchange beds. The actual radionuclides that could be measured with greatest sensitivity would depend on the time when the water entered the river and on the delay time between collection and analysis. In general, shorter lived radionuclides are present in higher concentrations, but they would have to be measured shortly after collection so as to minimize loss due to decay. Conversely, longer lived radionuclides are generally present in much lower concentrations and, therefore, larger samples would be required for measurements.

It is apparent from Table I and Fig. 1 that the radionuclides $^{24}$Na, $^{64}$Cu, $^{72}$Ga, $^{76}$As and $^{239}$Np are those with the greatest sensitivity if they could be measured within 1–2 days after discharge of the effluent water into the river. If the analysis of the radionuclides would be made one month or more after discharge of the effluent water into the river, then the most sensitive radionuclides would probably be $^{46}$Sc, $^{51}$Cr, $^{54}$Mn, $^{59}$Fe, $^{65}$Zn and $^{124}$Sb. In studies performed during the early 1950s and during the whole operating period of the Hanford reactors, all of the radionuclides were measured directly on residues from the evaporation of water samples or on ion exchange resins.

After discharge of the Columbia River into the Pacific Ocean and mixing of the river water with the saline water, the solubility of many of the radionuclides increases, while other radionuclides, which were originally associated with particulate material, may become attached to sediments and eventually be deposited on the ocean floor. This, however, takes a considerable period of time, and some of the highest radionuclide concentrations in organisms in the Pacific Ocean were found in filter feeders such as oysters and clams [5]. By sampling and analysing these and other organisms, it was possible to detect radionuclides at several trophic levels in the food chain.

One of the more abundant long lived radionuclides entering the ocean was $^{51}$Cr (27.7 d half-life). As shown in Fig. 2, this radionuclide was very soluble in river water, which is due to the fact that $^{51}$Cr was present as a dichromate ion. Techniques for collecting some of the more abundant radionuclides in reactor effluent water and fallout were developed and used. Large area chromatographic-grade aluminium oxide sorption beds [3, 4] were employed for the collection of radionuclides present in ocean water. By pumping large volumes (1–10 m$^3$) of filtered seawater through such 6 mm thick sorption beds, it was possible to remove a substantial fraction of a wide spectrum of radionuclides. While anionic $^{51}$Cr could not be collected with this procedure, a modified procedure, which involved impregnation of the aluminium oxide with the reducing agent SnCl$_2$, permitted reduction and quantitative sorption of $^{51}$Cr [6].

By employing this technique of large volume water sampling, followed by $\gamma$ ray spectrometric analysis of the sorption beds, it was possible to observe $^{51}$Cr in
the Columbia River plume in the Pacific Ocean for hundreds of kilometres. Figure 3 shows the concentrations of $^{51}$Cr measured by direct $\gamma$ ray spectrometry of samples collected during an ocean cruise from San Francisco through the Strait of San Juan Fuca [7]. This indicates the very high sensitivity that could be achieved for measurements of reactor effluent waters discharged into the ocean.

3. SUMMARY

In the case of reactors producing plutonium by using natural uranium fuel and once-through cooling, the technique which was employed in the Hanford reactors, a wide spectrum of radionuclides would be released in the cooling water. Several of these radionuclides could serve as excellent indicators of reactor operation processes. Those radionuclides with the greatest sensitivity to measurement for detection purposes and for which collection/analysis is possible shortly after discharge of the effluent water or for which a longer delay time is necessary are summarized below.

Possible signature radionuclides

<table>
<thead>
<tr>
<th>Short lived radionuclides</th>
<th>Longer lived radionuclides</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{24}$Na (14.96 h)</td>
<td>$^{46}$Sc (83.81 d)</td>
</tr>
<tr>
<td>$^{64}$Cu (12.70 h)</td>
<td>$^{51}$Cr (27.70 d)</td>
</tr>
<tr>
<td>$^{72}$Ga (14.10 h)</td>
<td>$^{54}$Mn (312.2 d)</td>
</tr>
<tr>
<td>$^{76}$As (26.3 h)</td>
<td>$^{59}$Fe (44.51 d)</td>
</tr>
<tr>
<td>$^{239}$Np (2.35 d)</td>
<td>$^{65}$Zn (243.8 d)</td>
</tr>
<tr>
<td></td>
<td>$^{124}$Sb (60.20 d)</td>
</tr>
<tr>
<td></td>
<td>$^{140}$Ba (12.75 d)</td>
</tr>
</tbody>
</table>

All of these radionuclides, collected by large volume sampling techniques, can be measured easily by direct $\gamma$ ray spectrometric analysis of filters or sorption beds, and very high sensitivity measurements are possible.

REFERENCES


ACTINIDE DETERMINATION AND ANALYTICAL SUPPORT FOR THE CHARACTERIZATION OF ENVIRONMENTAL SAMPLES

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Presented by I.N. Goldman

Abstract

ACTINIDE DETERMINATION AND ANALYTICAL SUPPORT FOR THE CHARACTERIZATION OF ENVIRONMENTAL SAMPLES.

Clean chemical and thermal ionization mass spectrometry procedures have been developed to permit the determination of environmental actinide element concentrations and isotopic signatures. The isotopic signatures help to identify the origin of elements and to separate naturally occurring or background contributions from local anthropogenic sources. Typical sample sizes for processing are 2 L of water, 1–10 g of sediment and 1–20 g of soil. Measurement limits are \(<1 \times 10^8\) atoms for Pu, Am and Np, and \(<2.5 \times 10^{12}\) atoms for U. For isotopic signatures, \(<5 \times 10^8\) atoms of Pu, Am and Np are necessary, and \(8 \times 10^{12}\) atoms of U are required. Of potential interest to the IAEA is the incorporation of these techniques into its Safeguards Analytical Laboratory for environmental sampling. Studies of surface waters, sediments and soils from the Rocky Flats Plant in Colorado are examples of this methodology. These studies showed that, although the actinide concentrations at the plant boundaries approached, on the downstream side, natural or background levels, the isotopic signatures characteristic of plant operations were still discernible.

1. INTRODUCTION

The US underground nuclear weapons test programme promoted the development of 'clean' chemical and instrumental measurement techniques. These techniques permit the analysis of very small concentrations of actinides extracted from very complex matrices. In support of this mission, the Los Alamos National Laboratory built a special 1400 m² facility that houses 'clean' chemical and instrumentation laboratories. 'Clean' in this sense means that the work surfaces are washed with

* Now with Radian Corporation, White Rock, NM, USA.
laminar flow air with less than 3500 particles per cubic metre (class-100, < 100 particles/ft$^3$) of 0.5 μm diameter or greater.

This facility is also being utilized to examine a wide variety of compatible problems, which include: determining actinide concentrations and isotopic signatures in surface waters, sediments and soils at the Rocky Flats Plant (RFP) near Denver, CO; determining the absolute uranium isotope inventory of holding ponds at the West Valley, NY, reprocessing plant; measuring the secular equilibrium concentration of naturally produced plutonium and technetium in several ore bodies (including Cigar Lake, Canada, and Alligator River, Australia), in support of the International Natural Analogues programme; and many isotope geochemistry projects in which small (less than picogram or femtogram) amounts of various isotopes ($^{230}$Th, $^{99}$Tc, $^{239}$Pu, $^{226}$Ra, $^3$He, $^{237}$Np and noble gas isotopes) are determined. Other projects provide high accuracy determinations of isotopic composition for various elements that are ubiquitous in our environment (Fe, U, Pu, Pb, Nd, Sr and Ba). To illustrate the use of this facility and the accompanying techniques, results for U and Pu samples taken at the RFP will be discussed.

This 'clean' facility is unique because of the size of the integrated chemical and instrumental capability under one roof. The methods developed and utilized therein, however, are not restricted to this magnitude of operation; their scope can be reduced so that they can be applied to much smaller 'clean' environments and used to give high integrity results.

It should be kept in mind that 'clean' facilities are designed to deliver analytical results that represent exactly the condition of the 'field' sample. If the integrity of analytical results cannot be maintained, proper interpretation of the results is not possible. Therefore, making a detailed sampling protocol and performance based evaluation of an individual laboratory on samples of known composition is the only way to ensure that proper interpretations can be made.

2. MEASUREMENT SYSTEM EVALUATION

The methodology used for the chemistry [1, 2], instrumentation and mass spectrometry [3-5] is well documented.

In order to establish the system accuracy for measuring ratios similar to those expected for environmental samples, synthetic standard solutions were produced from weighed dilutions of well characterized stock solutions: NBS 949f, a plutonium metal reference standard; and a 99.999% pure $^{242}$Pu spike obtained from Lawrence Livermore National Laboratory that was assayed with NBS 949f by isotopic dilution. The sample sizes were varied from 0.1 to 0.5 to 1.0 ng (nominal) in order to determine the effect of chemical yield on the measurement results (see Table I). The $^{239}$Pu/$^{242}$Pu ratios covered the range from $\sim 2.4 \times 10^{-5}$ to $\sim 1 \times 10^{-3}$, for a content of $6.3 \times 10^7$ to $2.5 \times 10^9$ atoms $^{239}$Pu per sample. When obviously contami-
nated samples are eliminated (5/72 or 6.9% of the total samples processed), the results yield positive biases, ranging from 21% for the small-value samples to essentially statistical agreement for the more concentrated samples. These positive biases are attributed to the measurement limitations of the single-stage mass spectrometer being used for this experiment. This positive bias represents a 13 fg (3.3 × 10^7 atoms) equivalent isobar. It should be noted that isotope signatures at the level of 5 × 10^8 atoms ^{239}\text{Pu} are altered by this level of bias by about 2.5% — about twice the precision/accuracy reported for this measurement. The possibilities of either isobaric interference or contamination at these levels of measurement are very real, even with the 'clean' laboratory. Therefore, the necessity of performing duplicate analyses and frequent parallel blank tests is obvious.

System blanks are ~ 1 × 10^7 atoms for Pu and (0.8-13) × 10^{12} atoms for U. This is the total system blank, including chemistry, loading and mass spectrometry. The blank for U is much higher because of the ubiquity of uranium in the environment.

3. RESULTS AND DISCUSSION

The RFP project was initiated to characterize the radioactivity in surface waters and sediments collected at the plant. The study quantified the amount of radioactivity present and determined whether the radioactivity was naturally occurring, background, or a result of plant operations. In this study, the local source terms are well defined, i.e. isotope signatures for both uranium and plutonium processed at the RFP are a matter of record. The external variables are naturally occurring uranium and its decay product, and fallout from weapons testing. Another objective of the study was to identify locations that may contain radioactive sources that could increase the surface water inventories at the RFP. The data collected in this study serve as a baseline from which the impact of future remediation efforts can be evaluated.

The RFP is built on the eastern slopes of the Rocky Mountains. Prevailing winds blow from west to east and drainage is also from west to east.

The largest amount of anthropogenic activity detected was in the pond sediments. One gram of pond sediment contains about 50 times more plutonium than 1 L of water (see Table II). It is also apparent that both depleted uranium and plutonium are mobile through the drainage system. The largest source of activity, however, is naturally occurring uranium and its decay product radium, which provides 70-450 times more alpha activity than that produced by the plutonium in the terminal ponds. The largest source of anthropogenic activity was depleted uranium, which comprised 20-50% of the total alpha activity in the samples.

Samples taken upwind and upflow (outside the plant boundaries) showed only naturally occurring uranium. Plutonium concentration was found at fallout levels, with the ^{240}\text{Pu}/^{239}\text{Pu} ratio also reflecting fallout.
## TABLE I. PLUTONIUM BLEND DATA: MEASURED $^{239}$Pu/$^{242}$Pu RATIOS

<table>
<thead>
<tr>
<th>Plutonium loaded</th>
<th>Blend 1</th>
<th>Blend 2</th>
<th>Blend 3</th>
<th>Blend 4</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.1 ng</td>
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<td>0.000249</td>
<td>0.000496</td>
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<td>0.000143a</td>
<td>0.000249</td>
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<td>0.000982</td>
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<td>0.000507</td>
<td>0.000989</td>
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<tr>
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<tr>
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<td>0.000008</td>
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<td>0.000009</td>
<td>0.000005</td>
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</table>

**Overall mean**

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<tr>
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<th>Mean</th>
<th>SD</th>
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<td>(11.8%)</td>
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<td>0.000003</td>
</tr>
<tr>
<td>(2.1%)</td>
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<tr>
<td>(2.0%)</td>
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<td>(1.1%)</td>
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</table>

**Make-up value**

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</thead>
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<td>0.000247</td>
</tr>
<tr>
<td></td>
<td>0.000496</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.000995</td>
</tr>
</tbody>
</table>

**Bias from make-up value**

|                   | +21%      | +1.6%     | +1.3%     | -0.3%     |

*a* Contaminated in process, dropped from mean and standard deviation (SD).
### TABLE II. PLUTONIUM CONCENTRATIONS AND $^{240}\text{Pu}/^{239}\text{Pu}$ ATOM RATIOS FROM SURFACE WATERS AND SEDIMENTS AT RFP

#### WATERS

<table>
<thead>
<tr>
<th>Sampling date</th>
<th>Pond</th>
<th>Sample number</th>
<th>Atoms/L $(10^{-9})^a$</th>
<th>$^{240}\text{Pu}/^{239}\text{Pu} \pm 1 \sigma$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1993-02-24</td>
<td>A1</td>
<td>SW60312WC</td>
<td>2.36</td>
<td>0.054 0.004</td>
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<td>A1</td>
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<td>A2</td>
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<td>SW60016JE</td>
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<tr>
<td>1993-05-12</td>
<td>A3</td>
<td>SW60015JE</td>
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<td>1993-02-24</td>
<td>B4</td>
<td>SW60314WC</td>
<td>3.21</td>
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<td>1993-05-13</td>
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<td>SW60320WC</td>
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<td>0.069 0.006</td>
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<tr>
<td>1993-08-23</td>
<td>C1</td>
<td>SW60050JE</td>
<td>2.34</td>
<td>0.057 0.005</td>
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</tbody>
</table>

#### SEDIMENTS

<table>
<thead>
<tr>
<th>Pond</th>
<th>Sample number</th>
<th>Atoms/g $(10^{-9})$</th>
<th>$^{240}\text{Pu}/^{239}\text{Pu} \pm 1 \sigma$</th>
</tr>
</thead>
<tbody>
<tr>
<td>A1</td>
<td>SW60277WC</td>
<td>98.0</td>
<td>0.052 0.005</td>
</tr>
<tr>
<td>A2</td>
<td>SW60285WC</td>
<td>116.8</td>
<td>0.061 0.001</td>
</tr>
<tr>
<td>B4</td>
<td>SW60272WC</td>
<td>43.8</td>
<td>0.059 0.001</td>
</tr>
</tbody>
</table>

* Two litres of water processed per sample.

#### 3.1. Plutonium results

Table II shows a small sample of the results from the measurement of the concentration and isotopic signature of plutonium from several of the holding ponds and sediments at the RFP. The plutonium concentrations vary from 0.13 to $27.12 \times 10^9$ atoms/L — a factor of ~200, with the measured isotopic signature maintained within measurement error over the entire range. Smaller concentrations of plutonium were measured in water, but isotopic signatures were not obtainable.
The sediment samples in Table II show much higher (approximately 50 times) concentrations of plutonium, and the $^{240}$Pu/$^{239}$Pu ratios are high precision measurements, with the exception of those for pond A1, where low count rates yielded a poor result. With only four exceptions, the plutonium concentrations measured in the waters around the RFP were below the allowable discharge limits, but identification of the source term by isotopic signature was still possible. Concentrations were determined by mass spectrometric isotope dilution.

Three terminal holding ponds, A4, B5, C2, and the sewage treatment plant were monitored on a monthly basis (see Fig. 1). The plutonium concentration of pond C2 definitely increases during the warmer months, which may be caused by
increased biologic activity (algae growth), increased solubility due to higher temperatures, or simply physical disturbance by plant personnel. An experiment designed to identify the cause of this increased concentration would be of benefit for future sampling strategies.

If a more sensitive measure of local activity and/or a time record is desired, sediments, where the plutonium is more concentrated, are the samples of choice. These layers could provide a history, for an extended period, of declared/undeclared activity levels determined from plutonium concentrations with isotope signatures providing source term identification. Other elements found in waters and sediments, such as technetium [6], could provide additional information.

3.2. Uranium results

Uranium is quite different from plutonium in that it is relatively abundant in nature, as much as 3.5 ppm in most regions of the Earth’s crust, and even higher in areas where uraniferous ores are present. Thus, it is much more difficult to detect anthropogenic uranium contamination. While a change in ratio of the major isotopes, $^{235}\text{U}$ and $^{238}\text{U}$, whose relative natural abundance is $\sim 1/137.8$, provides the best chance of determining anthropogenic activity, high concentrations of these isotopes can mask the presence of other isotopic compositions of uranium. Uranium has two other long lived isotopes, $^{234}\text{U}$ and $^{236}\text{U}$. Uranium-236 is generated by neutron capture on $^{235}\text{U}$ and is present in most anthropogenic uranium associated with nuclear fuel or weapons materials. In addition, the composition of $^{234}\text{U}$, with $\sim 55$ ppm of natural uranium, is also changed when either depleted or enriched material is added to that which occurs naturally. These two isotopes present two additional opportunities for detecting the presence of anthropogenic insertions into the environment.

Table III shows an example of some of the uranium measurements made in several ponds and sediments, and in a ditch at the RFP. The range of concentrations is much smaller ( $\sim 30$ times versus $\sim 200$ times) than in the case of plutonium because of the considerable background of natural uranium. Pond C1 shows a near-natural composition of uranium, while pond A1 closely approaches depleted uranium values. The $^{236}\text{U}$ content of pond A1 also shows a considerable increase over zero natural content, giving additional evidence of the presence of anthropogenic uranium. Pond A3, while closely approaching a natural $^{235}\text{U}/^{238}\text{U}$ ratio, has a perturbed $^{234}\text{U}$ and $^{236}\text{U}$ content, showing an anthropogenic contaminant. The ponds on the downstream side of the RFP show very low natural concentrations of uranium, but the isotopic signatures indicate perturbed $^{234}\text{U}$ and $^{236}\text{U}$ contents. Measurements of this type would be useful for detecting possible proliferant activities.

In the course of making measurements at the RFP, it has become apparent that two different types of analyses are required: measurement of plutonium, the concentration of which in nature from fallout is very low, remains a high-sensitivity and
TABLE III. URANIUM CONCENTRATIONS AND ISOTOPIC COMPOSITIONS FROM SEDIMENTS, SOILS AND WATERS

<table>
<thead>
<tr>
<th>Sample number</th>
<th>Description</th>
<th>Uranium (at. %)</th>
<th>Atoms/g (10⁻¹⁵)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
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<td>U-235</td>
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<td>Natural U</td>
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WATERS

<table>
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<th>Atoms/L (10⁻¹⁵)</th>
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<td>0.7225</td>
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</tbody>
</table>

a Sediment; b South interceptor ditch.

relatively low-precision method, while measurement of uranium, which has a very high concentration in the Earth’s crust, has become a high-precision/high-accuracy method requiring accurate measurement of the low-abundance isotopes ²³⁴U and ²³⁶U.

4. DISCUSSION

The combination of 'clean' room chemistry and thermal ionization mass spectrometry is a very sensitive, specific and unambiguous method for the determination of very low levels of actinides in environmental samples. Source term identification
information provided by isotopic signatures also helps to separate anthropogenic components from natural ones. The ability to isolate and characterize even very small quantities of actinides from large volumes of complex matrices permits the detection of undeclared activity with a very high success rate. Once the activity has been discovered, a whole barrage of other, more specific analyses, including particle analysis, can be brought to bear on the problem. This integrated sample approach, which was developed at LANL for nuclear weapons testing, provides a unique and cost effective tool for the non-proliferation and counter-proliferation community.

REFERENCES


SAFEGUARDS INFORMATION MANAGEMENT SYSTEM

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Lawrence Livermore National Laboratory, Livermore, California

United States of America

Presented by I.N. Goldman

Abstract

SAFEGUARDS INFORMATION MANAGEMENT SYSTEM.

The requirements for the management of information at the IAEA and its Department of Safeguards are rapidly changing. Historically, the Department of Safeguards has had the requirement to process large volumes of conventional safeguards information. An information management system is currently in place that adequately handles the IAEA’s conventional safeguards data needs. In the post-Iraq environment, however, there is a growing need to expand the IAEA information management capability to include unconventional forms of information. These data include environmental sampling results, photographs, video film, lists of machine tools, and open-source materials such as unclassified publications. The United States Department of Energy (DOE) has responded to this information management need by implementing the safeguards information management system (SIMS) initiative. SIMS was created by the DOE to anticipate and respond to IAEA information management needs through a multilaboratory initiative that will utilize an integrated approach to develop and deploy technology in a timely and cost effective manner. The DOE will use the SIMS initiative to co-ordinate US information management activities that support the IAEA Department of Safeguards.
1. INTRODUCTION

Historically, the Department of Safeguards has processed large volumes of conventional safeguards information derived from the Member States and the IAEA inspectors. An information management system is currently in place that adequately handles conventional safeguards information requirements. While at some point in the future this system will need modernizing, its basic capabilities meet the needs of conventional verification activities.

In the post-Iraq environment, however, the IAEA is experiencing a growing need to expand its information management capabilities to include more diverse types of information. These new information sources include environmental monitoring sample results (e.g. water and sediment), sample results from inside a facility (e.g. smear samples), photographs, maps, line drawings, video films, lists of equipment (e.g. machine tools and flow forming equipment), and open-source materials (e.g. unclassified publications).

Non-traditional and open-source data are anticipated to play a significant role in the IAEA’s effort to carry out its new and expanded responsibilities in international safeguards. The IAEA has recognized the need for enhanced information management capabilities and has started several internal initiatives and discussions with Member States. Both the Standing Advisory Group on Safeguards Implementation (SAGSI) and the Board of Governors have identified the need for the IAEA to implement enhanced and more sophisticated information management techniques. The objective is to improve the overall effectiveness of the IAEA and to strengthen international safeguards.

2. INITIATIVE OF THE UNITED STATES DEPARTMENT OF ENERGY (DOE)

The DOE has responded to this IAEA information management need by implementing the safeguards information management system (SIMS) initiative. The DOE will use the SIMS initiative to co-ordinate US information management activities that are in support of IAEA safeguards activities.

SIMS is a US multilaboratory initiative that utilizes an integrated approach to develop and deploy technology in a timely and cost effective manner. It is comprised of participants from the Lawrence Livermore National Laboratory (LLNL)\(^1\), the Los Alamos National Laboratory (LANL)\(^2\), the Pacific Northwest Laboratory

\(^1\) The LLNL performed this work under the auspices of the DOE under Contract W-7405-Eng-48.

\(^2\) This work is supported by the DOE, Office of Arms Control and Nonproliferation, International Safeguards Division.
3. APPROACH

The SIMS initiative will anticipate and respond to IAEA information management needs. The intent is to target specific, high priority needs. The initial tasks will be modest, very near term needs for fiscal year (FY) 1994 and FY 1995. The initiative will identify long term needs and develop a longer range vision during FY 1994 for FY 1996 and beyond. Integrated standards relative to system architecture, user interface and data communications protocol will be developed.

A key ingredient in the process is to initiate and maintain a dialogue with the users, thereby establishing IAEA ownership in both the process and the outcome. The products of the SIMS initiative will be modular in nature, allowing the IAEA to assimilate products and activities of manageable size. The expectation is to evolve a family of systems over time that meet the high priority needs of the IAEA.

It is our intent that the SIMS initiative be viewed as a technical support activity to the IAEA. It is anticipated that the SIMS family of systems (Fig. 1) will be discrete and evolutionary in nature. These products must be user friendly tools that can be used by IAEA inspectors and/or analysts. Throughout this process, the development activities will require close co-ordination with the IAEA to ensure that functionality is based on what the IAEA needs.

The initial activities build on the international nuclear safeguards inspection support tool (INSIST) platform, but we anticipate that significant changes in hardware application and software architecture may occur in the process of identifying user needs. The INSIST platform, which is described in detail in Ref. [1], has two unique features that meet existing needs at the IAEA: (1) the ability to geo-reference data and (2) the capability to handle multimedia information. INSIST is based on a Sun SPARC 10 workstation.

4. ORGANIZATION

The SIMS initiative is organized as shown in Fig. 2. Operationally, management of the initiative is provided by the DOE/Multilaboratory Steering Committee comprised of representatives from the four national laboratories and chaired by the

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3 The PNL is operated for the DOE by the Battelle Memorial Institute under Contract DE-AC06-76RL01830.

4 The SNL is operated for the DOE under Contract DE-AC04-94AL85000.
FIG. 1. Approach to the SIMS initiative.

FIG. 2. Organization of the SIMS initiative.
DOE. Its purpose is to (1) provide the overall leadership for the various working groups; (2) set the schedules; (3) approve the architectural designs, establish project management criteria including configuration control; (4) interact with other US Government (USG) agencies and initiatives (e.g. POTAS\textsuperscript{5}) and the IAEA; and (5) help prioritize user requirements. The Steering Committee provides the broad perspective for the initiative and articulates the SIMS vision.

At present, two Working Groups report to the Steering Committee, but another Working Group may be added in the near future. The two Working Groups are (1) the User Requirements Group and (2) the Technology and Systems Group. Each group is comprised of representatives from the national laboratories.

The User Requirements Group is an ad hoc Working Group that is responsible for interfacing with the IAEA community and determining its information management needs. The process used to develop the IAEA user needs will be interactive: the User Requirements Group will meet with the IAEA staff and management to obtain their perspectives regarding their needs, prepare a User Requirements Document following these discussions, and then review the document with the IAEA to obtain feedback. Once the User Requirements Group is satisfied that it has prepared a document that reflects the IAEA's perspective, the document will be presented to the SIMS Steering Committee. This User Requirements Document will evolve as needs are further identified and the project progresses (i.e. it will be a 'living' document).

In summary, the User Requirements Group will define IAEA user needs in terms of IAEA needs, but it will not define the functional capabilities necessary to meet these needs.

The Technology and Systems Group is an ad hoc Working Group that is responsible for developing and defining the umbrella systems architecture, standards for compatibility, software configuration management, and integration requirements. This group provides the expertise for computer science, software engineering, and hardware compatibility to the SIMS initiative.

Once the User Requirements Group completes the User Requirements Document, the responsibility of the Technology and Systems Group will be to ensure the necessary technical overview of the products being developed under the SIMS initiative. One of the main issues will be to ensure that the proper functionality has been defined.

A third ad hoc Working Group may be utilized in the near future. A Safeguards Advisory Group would be responsible for advising the Steering Committee on implementation issues. This group will be made up of subject matter experts who understand how a SIMS product will be implemented and how the information will be utilized at the IAEA.

\textsuperscript{5} United States Programme of Technical Assistance to IAEA Safeguards.
5. PRELIMINARY USER NEEDS FEEDBACK

The User Requirements Group held its first meeting with members of the IAEA staff and management on 10–14 January 1994. This group learned a lot about existing databases for traditional safeguards accountancy and related purposes. Significant progress has already been made with respect to managing information on nuclear activities of States.

The preliminary results of this meeting in Vienna indicate several high priority, immediate needs:

— Enhancements to the existing INSIST workstation dedicated to the UNSC-687 Action Team;
— Support of the field trials environmental monitoring data with a database for the Programme 93 + 2 initiative workstation;
— Notebook based field portable systems such as the on-site inspection system (OSIS), which was demonstrated throughout the meeting.

A follow-up meeting is planned for April/May 1994, to review and revise the findings of the User Requirements Group in January 1994. An important purpose of the planned meeting will be endorsement of these findings by the IAEA.

6. NEAR TERM (FY 1994) PRODUCTS

The User Requirements Group will identify and define IAEA user needs for enhanced safeguards information management capabilities aimed at strengthening safeguards. A User Requirements Document will be prepared, reviewed, and revised by the IAEA, and presented to the Steering Committee.

The Technology and Systems Group will define the hardware, software and databases to expand and enhance IAEA information management capabilities to meet the needs described in the User Requirements Document. This group will provide consistent interfaces for analysis, storage, retrieval, reporting and browsing application systems. In addition, the systems architecture will be developed to afford ease of maintenance, extensibility, and modularity of function.

An IBM compatible field portable unit with a graphical user interface will be developed to visually select and manipulate site diagrams, equipment lists, site information and inspection reports. The user will also be able to install and run independent applications (such as text retrieval and word processing) from a Windows environment. The field portable unit developed in FY 1994 will be considered a prototype because it will not be completed for the intended uses of the unit as an on-site inspection tool until the following fiscal year. Additional capabilities will be developed during the following year, utilizing the expertise gained through the early deployment and use of the unit by the IAEA.
An INSIST Programme 93 + 2 initiative workstation will be enhanced to handle environmental monitoring data from the IAEA Field Trials in Sweden and Hungary. This new capability will be implemented using a geographical information system (GIS). The GIS capability will provide for the ability to query a sample database and display results on a map background.

The INSIST UNSC-687 Action Team workstation will be enhanced to (1) search and retrieve text and graphics, (2) analyse data, (3) access their network for information, and (4) download selected information to portable computer systems. Additional database capabilities that are developed by the IAEA Action Team will be implemented in the INSIST workstation as appropriate.

In addition, TOPIC, which is a full-text search and retrieval software, will be integrated with the current INSIST code for rapid deployment to the IAEA.

Finally, an existing geographic reference capability (Gazetteer) developed at LLNL will be deployed. The Gazetteer is an index of names and co-ordinates that is accessed to implement text-to-location and location-to-text searches.

7. CONCLUSION

In conclusion, the demands on the international safeguards world is changing, and the information management systems must adapt to new sources and types of information. In response to these changes, the DOE has created the SIMS initiative that is intended to anticipate and respond to the future IAEA information management needs. This DOE multilaboratory initiative will target specific, high priority IAEA information management needs. Even though the SIMS initiative is still in its infancy, it has the potential for developing and deploying significant information management systems for the IAEA.

REFERENCE

CAN INTERNATIONAL SAFEGUARDS BE EXPANDED TO COVER TRITIUM?

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Abstract

CAN INTERNATIONAL SAFEGUARDS BE EXPANDED TO COVER TRITIUM?

Effective international control of tritium does not yet exist although there are several arguments in favour of it. A precedent is given by an agreement between Euratom and Canada. The paper discusses whether the necessary political and technical preconditions for such control are fulfilled. For this purpose an international tritium control system (ITCS) has been outlined. The goal is to avoid any undeclared tritium uses originating from civilian facilities. The rules and verification procedures of the ITCS are comparable to those agreed upon for plutonium and highly enriched uranium in the NPT. Therefore, the paper discusses the possibility of tying an international tritium control to the NPT, as an integral part or additional protocol. The required verification measures as obtained by a comprehensive diversion path analysis entail the verification of non-production and of non-removal from existing and declared inventories of tritium in civilian facilities. The technical feasibility of verification is assessed and compared to the standards agreed upon for nuclear safeguards. For the assessment the significant quantity of tritium is assumed to be one gram; a detection time of one year should suffice.

1. POLITICAL BACKGROUND OF INTERNATIONAL TRITIUM CONTROL

Although tritium is not essential for the production of nuclear weapons, all five recognized nuclear weapon States (NWS) make extensive use of tritium within their weapons programmes, and other States are also thought to have followed this path or to have attempted to do so. In turning from simple fission devices to fusion boosted or thermonuclear weapons, tritium plays a key role. Tritium has strategic significance, because it enables smaller and lighter warheads to be built, while retaining the same yield. Because of its decay rate of 5.5% per year, a continuous supply of tritium is necessary to maintain those nuclear arsenals which depend on it. In the long run, this might pose a problem to Ukraine. Tritium from civilian sources is becoming increasingly available in excess of civilian demand, which raises questions about its non-proliferation.
Therefore, the control of tritium is of relevance to fulfil the norm of horizontal non-proliferation and has consequently to be dealt with within the nuclear non-proliferation regime. An effective international control of tritium would strengthen this regime.

The control of tritium, which used to be regulated solely by national export controls, has gradually been expanded to the international level. The inadequacy of these controls became known to the public in 1989, after illegal exports of tritium and associated technology by German companies to Pakistan had been uncovered. Most initiatives at the international level aim at tightening the export controls of tritium (Co-Com; 4th NPT Review Conference; ' dual-use list' of the Nuclear Suppliers Group) by denying proscribed countries the transfer of tritium and associated technology. The notable exception is an extension of a co-operation agreement between Euratom and Canada [1] which was finalized in 1991. It has given Euratom the mandate to control tritium imports for fusion research from Canada to Euratom Member States. In this agreement, Euratom will act as a supervising agency authorized to establish control procedures for tritium shipments and to verify the declared end use.

However, to date international tritium control has only been incoherent and insufficient, since only some areas of concern have been regulated, while most proliferation paths remain open [2].

2. INTERNATIONAL TRITIUM CONTROL SYSTEM

To correct the identified deficits, an international tritium control system (ITCS) has been suggested [3, 4]. It represents a comprehensive and systematic approach to dealing with the problem of curbing the horizontal proliferation of tritium, while allowing its civilian use.

The goal is to avoid any undeclared tritium use originating from civilian facilities. In this scenario, any non-nuclear-weapon State has to abide by four rules:

(a) No use of tritium for nuclear weapon purposes
(b) No undeclared tritium imports
(c) No undeclared tritium production
(d) No removal of tritium for undeclared uses.

For NWS these obligations are only applied to all civilian facilities and materials. Nuclear weapon uses of tritium coming from military production facilities in NWS are permitted and remain uncontrolled1.

1 However, if a cut-off of fissile materials were to be negotiated, due consideration should be given to including tritium in such an agreement. It could complement the ITCS, because it would operate within the norm of vertical non-proliferation [3–5].
2.1. Verification requirements of the ITCS and possible IAEA involvement

Verification procedures are an important feature in assuring compliance with both the legal obligations and the political commitments. The required verification measures are in many ways similar to the safeguards carried out by the IAEA for fissile materials (see below). The IAEA could technically be involved in verifying obligations (c) and (d) and in detecting non-compliance. The verification procedures are largely dependent on the shaping of the rules of the envisaged agreement, but no amendment to the Statute would be necessary. According to its Statute the IAEA concentrates its efforts on 'special fissionable materials'. However, tritium could be comprised within the term 'other materials' (Article III, A5). Such a 'tritium mandate' would also be consistent with the IAEA safeguards philosophy of concentrating on the material itself.

The IAEA can be requested to apply safeguards (Article XII, A) by the parties to any agreement, if it is in accordance with the Statute.

Following on from the rules, the verification procedures entail the following tasks:

Type I: Verify compliance with rule (c) by verification of non-production,
Type II: Verify compliance with rule (d) by verification of non-removal from existing and declared inventories.

The problem of clandestine facilities is the same as the one that the IAEA faces in connection with the NPT. The IAEA has only recently begun to gain experience with clandestine facilities. 'Special inspections' (INFCIRC/153,73) have not, however, been carried out under the NPT, but under United Nations Security Council resolutions.

2.2. Organizational framework of the ITCS

The rules of the ITCS are comparable to those agreed upon for plutonium and highly enriched uranium (HEU) in the NPT. Therefore, international tritium control could also be tied to the NPT. There are two ways to achieve this. Firstly, tritium control could be integrated into the NPT. In this case, the NPT would have to be amended. Decision making procedures for such an amendment are included in the NPT (Article VIII). In addition to the NPT, the model agreement (INFCIRC/153) would have to be renegotiated and altered to include tritium. Under current political circumstances this option is not likely. The NPT is considered by many scientists and political actors as unamendable, because attempts at amendments would lead to divisive debates which might endanger the NPT and the entire non-proliferation regime. However, the circumstances might change and the possibility of this option may improve, if the NPT can be stabilized after a possible indefinite extension in 1995.
Secondly, the NPT could be supplemented with a protocol. Such a protocol would also be an integral part of the NPT, but would not create the need for an amendment and a reopening of the NPT. Moreover, membership in the NPT is not tied to membership in the protocol, allowing a gradual development of the scope of the protocol in terms of membership. INFCIRC/153 could remain unchanged and a separate verification agreement between the IAEA and the respective Member States of the protocol would then have to be negotiated. Since both possibilities are tied to the NPT, the approach towards civilian facilities in NWS would be in a like manner. Under the NPT, NWS have offered the inspection of some civilian facilities. However, according to this approach taken in an ITCS, all civilian facilities of the NWS would have to be controlled. The implementation of the latter approach is basically dependent on their political will to open all their facilities for inspection and on the financial contributions that all Member States are prepared to make towards the verification procedures of an ITCS.

The alternative would be a separately negotiated ITCS. Although it would not be tied to the NPT, its character would be similar. This option would still allow the IAEA to perform the verification of the NPT and ITCS obligations during the same inspection.

3. VERIFICATION

3.1. Significant quantity (SQ)

The average amount of tritium in warheads is 2–3 g. Specific information concerning the minimum quantity of tritium necessary for a boosted or thermonuclear warhead is classified. A State that intends to use tritium for weapons can, however, make use of smaller amounts for research purposes and to gain experience in handling.

By using tritium the yield of a warhead will be boosted by a factor of between two and ten. Therefore a SQ must be established that is aimed against the horizontal proliferation of one boosted nuclear weapon. Although with respect to vertical non-proliferation within NWS only larger amounts are significant, the diversion of one SQ in a NWS is equally relevant, since it could end up in a foreign clandestine programme. A rather conservative approach would be to put the SQ at one gram.

3.2. Conversion time and efficiency

Conversion time depends on the availability and also on the chemical and physical form of the diverted tritium. It is by IAEA standards a determining factor

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2 The ITCS encompasses several variations [4].
regarding the frequency of inspections. Since the minimum period for putting diverted tritium to use in warheads could be as low as a few days, the inspection frequency would need to be higher than 52 times per year. Such an inspection frequency would not be practical in terms of required manpower and is out of proportion, because of the limited significance of tritium for the horizontal non-proliferation of nuclear weapons in relation to fissile materials. Moreover, the limitations set by the IAEA verification budget have to be taken into account. Nevertheless, the inspecting agency needs to establish the truth regarding the received data.

As far as reactors are concerned, a possible solution would be to tie tritium inspections to the routine IAEA inspection procedures for plutonium and HEU. However, all facilities with an inventory or production capacity of one gram or more per year should be inspected at least once per year. Thereby, usual reporting periods of inventories would be observed. But IAEA inspectors would need additional training and instruments which could be provided by the national support programmes. However, some tritium handling facilities in industry and research have to be inspected solely for tritium.

The restrictions with regard to inspection frequency pose the question of 'efficiency'. This requirement is still satisfied. Firstly, the inspections would still be a deterrent. Secondly, although a diversion of tritium might have taken place one year earlier, such a verification method would provide sufficient time to allow for political reaction. If a diversion of one SQ has been detected, it can be assumed that the proliferating State already has a first generation weapon or the capability and fissile materials for its production. If a proliferating State intended to use a nuclear weapon militarily, it would not do so immediately after the acquisition of tritium, launching an untested boosted weapon. Besides, verification is also a political and not only a technical matter and the attitude towards verification is determined primarily on the basis of political interests. In this context, efficiency cannot only be measured in technical terms. If the verification procedures are politically satisfying and acceptable, there will be no need to stretch technical verification goals beyond this required level.

3.3. Diversion path analysis and derived verification tasks

In order to find out what kind of verification activities would be needed, a comprehensive diversion (i.e. production and removal) path analysis was performed [6].

Possible paths to divert pure tritium in significant quantities are: breeding of tritium (\(^6\)Li path and \(^3\)He path), extraction of inadvertently produced tritium (heavy water path and ternary fission path) and removal of stored pure or recoverable tritium. For each path the maximum divertable quantity has been estimated. Only diversion paths by which more than one gram per year can be acquired were considered in this verification concept.
As a result, two different control tasks have to be solved. Verification of non-production (type I) relies mainly on the non-destructive analysis of possible irradiation targets at nuclear reactors to look for the absence of $^6\text{Li}$, the main raw material for tritium breeding. Verification of non-removal (type II) is achieved by accountability complemented by containment and surveillance at tritium handling facilities.

### 3.4. Verification of non-production

Most tritium control activities of type I are already covered by routine nuclear safeguards procedures. In general, safeguards designed to detect clandestine production of plutonium would detect production of tritium as well, making only a few additional measures necessary.

For example, lithium target detection in power reactor fuel can be achieved by non-destructive safeguards applied by the IAEA at the fuel fabrication facility. After inspection the fuel assemblies are sealed. Further safeguards are carried out by item counting, seal inspection, and containment and surveillance measures. The main IAEA safeguards instruments for physical inventory verification of fresh fuel are neutron coincidence counters. They provide a semi-quantitative measurement of enrichment and total uranium. They can detect non-uranium rods in the fuel assembly. The production scheme most difficult to detect would be the covering up of lithium in the fuel by declaring it to be gadolinium. Monte Carlo calculations indicate that even under such unfavourable circumstances routine measurements with the neutron coincidence collar would result in anomalies [7]. If these cannot be resolved, further, possibly destructive, investigation would be triggered.

For non-destructive identification of $^6\text{Li}$, active interrogation with neutrons or gamma rays is necessary. A dozen different measurement principles have been investigated [6]. At least one of them appears promising: nuclear resonance absorption (NRA). The 3.56 MeV level of $^6\text{Li}$ nearly satisfies the conditions defined in Ref. [8]. The ratio of the partial width of direct gamma ray transition to the ground state and the total width of this level is $\Gamma_{\gamma 0}/\Gamma_T = 0.999$ (and should be $> 0.01$), the ratio of the Doppler width to the total width is $\Delta/\Gamma_T = 1.38$ (but should be $< 1$).

Another method, which appears less promising, is gamma ray spectroscopy, used to identify peaks from the $^6\text{Li}(n,\gamma)^7\text{Li}$ reaction. Experiments have been undertaken to investigate the feasibility of identifying peaks at 0.478, 6.78 and 7.26 MeV using a Ge(Li) detector [6]. The low energy peak could be seen but could have indicated $^{10}\text{B}$ as well via $^{10}\text{B}(n,\alpha)^7\text{Li}$. The high energy peaks have too low a yield of gamma rays compared to the background. Long integration times (some ten hours) with a comparatively strong neutron source (about $10^8$ neutrons/s) would be required.
3.5. Verification of non-removal

The control tasks of Type II would have to be introduced at civilian tritium handling and production facilities with inventories or annual throughput greater than one gram. Worldwide there are fewer SQs of tritium (25 000) than of plutonium (111 000; Table I) in civilian stocks.

Accounting procedures are already established on the facility level, mainly for radiation protection purposes. The data obtained from tritium accountancy would have to be made available to the IAEA. The IAEA would have to evaluate and verify them with on-site inspections.

The United States Department of Energy (DOE) has a system of accounting and control. All quantities of tritium equal to or greater than 0.0005 g (0.18 TBq) are considered accountable within one material balance area (MBA)\(^3\). For DOE reporting purposes the accountable quantity of tritium for each facility is one order of magnitude larger [11].

3.5.1. Material unaccounted for

The book inventory \(B\) at the end of an inventory period is compared with the new physical inventory \(I_1\). The difference is the material unaccounted for (MUF). Apart from possible undeclared removals \(D\), there are three kinds of contributions to MUF:

1. The actual measurement error occurring in the inventory period in question, \(E_1\), causes random and systematic contributions to MUF.
2. An unaccounted process loss \(L\) causes a systematic contribution to MUF.
3. A hidden inventory change \(\Delta H\) causes a temporary contribution to MUF.

\[
\text{MUF} = B - I_1 = D + E_1 + L + \Delta H
\]

3.5.2. Inventory measurement of available tritium

An extensive analysis of accuracy in tritium inventory measurements in different modes (gaseous and aqueous storages, operating systems) has been undertaken to estimate typical values of MUF when closing a material balance [6].

The technical problems in accountancy can be aggregated and compared by defining the expected accountancy capability \(E\) which is the minimum loss of nuclear material that can be expected to be detected by material accountancy:

\[
E = 3.29 \times \delta_E \times A
\]

\(^3\) The same is envisaged for two large European tritium handling facilities [9,10].
TABLE I. COMPARISON OF DEMANDS AND EXPECTED CAPABILITIES FOR PLUTONIUM AND TRITIUM ACCOUNTANCY

<table>
<thead>
<tr>
<th></th>
<th>Plutonium</th>
<th>Tritium</th>
</tr>
</thead>
<tbody>
<tr>
<td>Significant quantity SQ(^a)</td>
<td>8 kg</td>
<td>1 g</td>
</tr>
<tr>
<td>(\delta_{\text{MUF}}) at bulk handling facilities</td>
<td>0.005-0.010</td>
<td>0.0025-0.0500</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Quantities</th>
<th>t</th>
<th>SQ(_{Pu}) kg</th>
<th>SQ(_T)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total world civilian inventory, end of 1993</td>
<td>890</td>
<td>111 000</td>
<td>25</td>
</tr>
<tr>
<td>Subject to IAEA safeguards, end of 1992</td>
<td>404</td>
<td>50 500</td>
<td>0</td>
</tr>
<tr>
<td>Separated world civilian inventory, end of 1993</td>
<td>180</td>
<td>22 500</td>
<td>8</td>
</tr>
<tr>
<td>Total world military inventory in 1993</td>
<td>250</td>
<td>31 250</td>
<td>130</td>
</tr>
<tr>
<td>Expected accountancy capability (E) (see text) for separated world civilian inventory</td>
<td>2.96-5.92</td>
<td>370-740</td>
<td>0.066-1.32</td>
</tr>
</tbody>
</table>

\(^a\) It should be noted that the definition of SQ for tritium is more conservative than it is for plutonium. SQ\(_{Pu}\) is larger by a factor of about two than the actual average amount of plutonium assumed to be used in nuclear warheads, whereas SQ\(_T\) is smaller by a factor of two to three than the average quantity of tritium believed to be inserted in weapons. Therefore, the assessment for tritium is more conservative than it is for plutonium by a factor of about four to six.

where \(A\) is the amount of material in the material balance expressed as the larger of the inventory or throughput, and the factor 3.29 corresponds to a detection probability of 0.95 and a false alarm probability of 0.05; \(E\) is the expected measurement accuracy for closing a material balance (expected accuracy of MUF). Expected measurement accuracies in nuclear safeguards based on international standards of accountancy, i.e. considered achievable in practice at bulk nuclear facilities, range from 0.002 for uranium enrichment plants and 0.01 for plutonium reprocessing plants to 0.25 for separate waste storages facilities [12].

Most measurement accuracies of MUF which are actually achieved or expected in various tritium handling facilities are between 0.0025 and 0.05, and about 0.2 for tritiated waste [6]. Studies undertaken for two large European tritium handling facilities with inventories of up to 100 g conclude that inventory measurements will have an accuracy of 0.03-0.04 (PVT-c), and with current technical constraints the MUF will have a standard deviation of about 0.05 [9, 10]. Progress can be expected through the application of calorimetry and more accurate determination of tritium in solid waste.
The contribution of waste to the unaccounted tritium can be expected to be small because the total waste per year is typically of the order of 1% of the inventory. Even for a large facility the expected accountancy capability $E$ associated with closing the tritium balance in the waste stream will be less than 0.1 g/a. Table I compares expected accountancy capabilities $E$ for plutonium and tritium. The expected detection capability in an imaginary closing of the material balance of the whole world inventory of all separated, civilian tritium stocks would amount to some 66 to 1300 SQs of tritium. For plutonium the respective amount is 370 to 740 SQs.

As a result, with current state-of-the-art technology and under normal operating conditions, tritium accountancy is possible on a routine basis with a capability which compares well with that required for nuclear safeguards.

### 3.5.3. Undetected losses of tritium

By definition, all tritium leaving the MBA without being accounted for belongs to this category. There are a variety of material loss mechanisms. Tritium can be lost, unnoticed, to the environment (in the production chain, handling and shipping, due to venting of equipment for maintenance) and as residues in waste which leaves the MBA without the tritium content being assessed. Tritium lost on account of radioactive decay does not belong to this category, because it can be calculated with a high degree of accuracy.

During normal operation the tritium concentration in vented air and discharged effluents is monitored. Emitted quantities can be calculated and treated as output in the material balance. Therefore, the amount of tritium lost to the environment without being counted can be expected to be much smaller than the total amount of tritium lost. A recent review showed that a containment performance can be achieved in which not more than 1% of the inventory is lost, or even less than 0.01% per year [13]. Even if as much as 10% of tritium lost to the environment were undetected this would not amount to more than 0.1 g/a in a large tritium handling facility with an inventory of 100 g, i.e. $L < 0.1$ g.

### 3.5.4. Hidden inventory

The hidden inventory is defined by unaccounted hold-ups in various process elements, most of it in uranium getter beds. This is nothing unfamiliar to nuclear materials accountancy. However, because of its high mobility and reactivity, it is more difficult to locate tritium than solid nuclear materials.

Studies for two European facilities conclude that a few grams of tritium will be bound; this is not linearly dependent on the total inventory$^4$. Most of it can be

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$^4$ See Refs [9, 10]. The latter estimates 3.5 g for the European Tritium Handling Experimental Laboratory, ETHEL, which includes 1.3 g annual accumulation of solid waste. This should in fact be accounted for as output. It can be measured with an error contributing to $E_1$. 

regained by special heat treatment without opening the equipment, and annual changes of bound inventory represent only a fraction of it, i.e. $|\Delta H| < 1 \text{ g}$.

4. CONCLUSION

A comprehensive and coherent international control system for tritium does not currently exist, but it is desirable and would represent an important instrument against the proliferation of advanced weapons designs into States with secret nuclear weapons programmes. The non-proliferation regime would be strengthened by expanding its scope to tritium.

The outlined ITCS could provide such a systematic control of tritium at an international level. It would be politically acceptable and technically feasible. It could be implemented without creating excessive bureaucracy or adding a disproportionate financial burden. Existing structures within the non-proliferation regime can be used to close this loophole. The ITCS could be tied to the NPT, and verification procedures could rely on IAEA safeguards comparable to those for fissile materials.

Tritium control can be achieved with state-of-the-art technology. Most tritium control activities of type I (verification of non-production) are already covered by routine nuclear safeguards procedures. Only a few additional measures are necessary.

Control tasks of Type II (verification of non-removal) can be based on technologies and experiences from tritium accountancy already established for radiation protection purposes. The precedent for this type will be set by an agreement between Euratom and Canada.

ACKNOWLEDGEMENT

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1. DECLARED ACTIVITY

We shall examine a class of inspection resource allocation problems in which it is supposed that only a subjective ranking of violation risks is feasible and in which only an equally subjective yes/no answer can be given to the likelihood of detection in any given instance. We have in mind inspection in which the objective is to verify the absence of undeclared activity, rather than confirmation of declarations. As a starting point, we consider first verification of declared activity.

Figure 1 is an analysis of a prototype data verification problem, showing, in the form of a game tree, the decision choices available to the two potential protagonists, inspector and inspectee. Specifically, a so-called attributes sampling situation is modelled in which false alarms, i.e. false accusations of violation, do not play a role.

In Fig. 1, $H_1$ and $H_0$ indicate illegal and legal behaviour by the inspected party, and $A$ and $\bar{A}$ indicate alarm or no alarm as a result of inspection. The doublets $(-a, -b)$, $(-c, +d)$ and $(0, 0)$ are the pay-offs (inspector, inspected party) to the players for the three possible outcomes of inspection: detected violation, undetected violation and legal behaviour, respectively. The symbol $\phi$ represents a strategy for the inspector chosen from some perhaps infinite but well defined set $\Phi$ of available inspection strategies. The inspected party decides with probability $t$ to act illegally, then choosing the data falsification strategy $\psi$, again from some known set $\Psi$ of possibilities. The detection probability $1 - \beta(\phi, \psi)$ for violation is a function of the two strategies chosen. It is assumed that it can be calculated and that it is a decreasing function of an implicit external parameter representing the amount of inspection.
FIG. 1. Extensive form of a simultaneous attributes sampling inspection game involving inspector, inspected party and chance. The shaded areas denote information sets of the inspected party. They reflect the assumption that at its decision points the inspected party is not aware of the strategy \( \varphi \) of the inspector.

effort invested. Note that the game model, although it is non-co-operative, is not zero-sum.

The pay-off utilities \( a, b, c \) and \( d \) reflect the subjective preferences of both players, and are assumed to satisfy \( 0 < a < c \) and \( 0 < b, 0 < d \). The assumption \( a > 0 \) means that the highest priority of the inspector is not detection of violation but rather inducement to legal behaviour, i.e. deterrence of violation. Of course the worst outcome for the inspector is non-detection, so that \( a < c \). Utility \( b \) is the inspected party’s perception of the consequences of being detected, and \( d \) is its incentive to violate nevertheless.
This inspection problem is solved by seeking an equilibrium of the game, i.e. a combination of strategies \((\varphi^*; t^*, \psi^*)\) that satisfy the following simple but fundamental property:

*Unilateral deviation of one of the players from equilibrium does not improve, but may diminish, his pay-off.*

Given the preferences of the players, an equilibrium strategy is the only recommendation that we can defensibly make to them. In particular, \(\varphi^*\) is the optimal inspection strategy.

It is easy to prove an important result for the situation in Fig. 1, namely that the sought-after equilibrium strategies must satisfy (see Ref. [1])

\[
\beta(\varphi^*, \psi) \leq \beta^* \leq \beta(\varphi, \psi^*)
\]  

(1)

for all \(\varphi \in \Phi\) and \(\psi \in \Psi\)

where \(\beta^* = \beta(\varphi^*, \psi^*)\), and, furthermore,

\[
t^* = \begin{cases} 
0 & \text{for } \beta^* < 1/(1 + dB) \\
1 & \text{for } \beta^* \geq 1/(1 + dB)
\end{cases}
\]

(2)

Inequalities (1) define the saddlepoint solution of a zero-sum game which does not involve the subjective utilities at all, but which nevertheless determines completely the inspector’s optimal strategy \(\varphi^*\). The pay-off function in this zero-sum subgame is the non-detection probability alone — a purely technical, in principle calculable, quantity. Provided the zero-sum game can be solved, \(\varphi^*\) can be determined objectively and recommended unequivocally. Subjectivity only enters into the inspected party’s decision \(t^*\) as to whether or not to behave legally, and there only as the ratio \(dB\) in Eq. (2). If the inspector is prepared to ascribe preferences to the inspected party, Eq. (2) also tells him that he should invest sufficient effort to ensure that the saddlepoint of the non-detection probability satisfies

\[
\beta^* < 1/(1 + dB)
\]

(3)

and thus fulfil his primary objective of deterring illegal behaviour.

2. THE IAEA SYSTEM

The utility ratio \(dB\) might be referred to as a *reciprocal index of deterrence*. Can we say anything sensible about it for international verification regimes without being political? This is clearly not easy. However, we can, at least in one context, ‘observe’ it empirically.

The IAEA sets its own safeguards effectiveness criteria, and it does so in part by prescribing for its inspectors in the field minimum required detection probabilities
TABLE I. SOME IAEA DETECTION PROBABILITY CRITERIA

<table>
<thead>
<tr>
<th>Material category</th>
<th>Quantity</th>
<th>$1 - \beta^*$</th>
<th>$d/b$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Direct use material (plutonium, highly enriched uranium) in readily accessible form</td>
<td>Pu: 8 kg U-235: 25 kg</td>
<td>90%</td>
<td>9</td>
</tr>
<tr>
<td>Irradiated direct use material (e.g. reactor spent fuel) not under C/S</td>
<td>Pu: 8 kg</td>
<td>50%</td>
<td>1</td>
</tr>
<tr>
<td>Natural uranium at fuel fabrication plants not under C/S</td>
<td>U-235: 75 kg</td>
<td>20%</td>
<td>1/4</td>
</tr>
<tr>
<td>Material that is very difficult to access (e.g. long term stored spent fuel and waste) under C/S</td>
<td>Pu: 8 kg</td>
<td>10%</td>
<td>1/10</td>
</tr>
</tbody>
</table>

for diversion from classes of declared material considered to have varying degrees of misuse potential.

In deciding on these criteria, some of which are presented in Table I [2], the inspectorate is certainly not motivated by the rather abstract analysis given above, but nevertheless the intention is the same: Different classes of materials are believed to have a different subjective attractiveness for short term misuse by the inspected party and, therefore, purely intuitively, different probabilities of detection are required.

The criteria in Table I are applied uniformly in all Member States so that, although these criteria are sometimes controversial and difficult to achieve, they are certainly impartial. The last column of Table I shows the inspectorate’s implied value judgement regarding the inspected party’s utility ratio according to Eq. (2); for simplicity, we have ignored the effect of measurement error for variables sampling an equilibrium very similar to Eqs (1) and (2), although it involves the false alarm probability.

Thus, the policy of inspectorates, in the case of the IAEA arrived at in a rather painful consensus finding process, can be related to subjective perceptions of the stakes involved. This gives us a basis for proceeding into more difficult territory.
3. UNDECLARED ACTIVITY

It will generally not be possible to determine detection probabilities for inspection procedures designed to detect undeclared activities within States that are violating their commitments. The scenarios are simply too diverse and the procedures too ill defined. We will therefore formulate the problem in a somewhat different way, in the knowledge that the satisfactory result arrived at in the previous section will no longer obtain: We will not be able to extricate ourselves from the mire of subjective utilities in deriving optimal inspection strategies. Nevertheless, we shall continue to be precise and see where that takes us.

To begin with, let us divide the violation possibilities or paths of concern into classes, $K$ of them, say, according to the subjective preferences or incentives of the potential violator, $d_i$, with $i = 1 \ldots K$. We might then order the classes according to decreasing incentive, i.e. $d_1 \geq d_2 \geq \ldots \geq d_K$. We expect that the number of paths in each class, $N_i$, with $i = 1 \ldots K$, may be large and hence require random sampling. If, however, illegal activity is occurring in one of these paths, and if, by chance, it is inspected, we will assume that violation will be detected.

In order to formulate the problem correctly, we must introduce additional utilities analogous to those of Section 1, even though we had only the subjective preferences $d_i$ in mind. We will see soon that this does not matter. For legal behaviour the pay-off doublet (inspector, inspected party) is again $(0,0)$, for undetected violation in the $i$-th class it is $(-c_i, d_i)$, and for detected violation it is $(-a_i, -b_i)$. As before, $0 < a_i < c_i, b_i > 0$ and $d_i > 0$, with $i = 1 \ldots K$.

If the inspector makes $n$ inspections in all, then his set of inspection strategies is

$$\Phi = \left\{(n_1, \ldots, n_K) \mid n_i \geq 0, i = 1 \ldots K, \sum_{i=1}^{K} n_i = n \right\}$$

$n_i$ being the number of items chosen for inspection in the $i$-th class. The set of illegal strategies is

$$\Psi = \left\{(q_1, \ldots, q_K) \mid 0 \leq q_i \leq 1, \sum_{i=1}^{K} q_i = 1 \right\}$$

assuming violation in just one class. Here, $q_i$ is the probability that the inspected party chooses the $i$-th class for its illegal activity.

An illegal strategy $\psi$ will consist at worst (from the inspector's viewpoint) of a violation at just one location, so that the probability of detecting it is $n_i/N_i$. The pay-off to the inspector is therefore

$$E_1(n, q) = \sum_{i=1}^{K} -c_i q_i \left(1 - \frac{n_i}{N_i}\right) - a_i q_i \frac{n_i}{N_i}$$
and the pay-off to the inspected party is

\[ E_2(n, q) = \sum_{i=1}^{K} d_i q_i \left( 1 - \frac{n_i}{N_i} \right) - b_i q_i \frac{n_i}{N_i} \]

The game whose equilibrium we are now seeking is the quadrupole

\[ \langle \Phi, \Psi, E_1(n, q), E_2(n, q) \rangle \]

We will present only part of the solution here, without proof and in the form of a prescription for the inspector. The complete solution is given in Ref. [1].

Having ordered the inspected party's attributed preferences \( d_i \) as suggested, the inspector chooses a value of \( k, 1 \leq k \leq K \), such that the following inequalities are satisfied:

\[
\sum_{j=1}^{k-1} \frac{d_j - d_k}{b_j + d_j} \frac{N_j}{n} < 1 \tag{4a}
\]

\[
\sum_{j=1}^{k} \frac{d_j - d_{k+1}}{b_j + d_j} \frac{N_j}{n} > 1 \tag{4b}
\]

If inequality (4b) is already satisfied for \( k = 1 \), then inequality (4a) is omitted; if inequality (4a) is still met with \( k = K \), then inequality (4b) is omitted. Having obtained \( k \) in this way, the inspector proceeds to solve the equation

\[
\sum_{j=1}^{k} \frac{d_j - E^*_2}{b_j + d_j} \frac{N_j}{n} = 1 \tag{5}
\]

for the equilibrium pay-off \( E^*_2 \) to the inspected party. The inspector's optimal strategy is then given by

\[
n^*_i = \begin{cases} 
N_i \frac{d_i - E^*_2}{b_i + d_i} & \text{for } i = 1 \ldots k \\
0 & \text{for } i = k + 1 \ldots K
\end{cases} \tag{6}
\]

We see that the inspector's solution is expressed entirely in terms of the utilities of the inspected party. This is a general property of equilibria in two-person games. Moreover, if the inspector is willing to assume that the inspected party's perception of the consequences of detection, \( b_i \), are independent of the class, i.e. \( b_i = b \), \( i = 1 \ldots K \), he can follow the above prescription on the basis of the attributed ratios \( d_i/b \) alone. This is in effect a task which is not more formidable than that faced by the inspector (Fig. 1) in choosing \( 1 - \beta^* \) to deter illegal behaviour across classes of material. The difference is, of course, that the optimal strategy for the game illustrated in Fig. 1 is independent of all subjective utilities. In the above example this is not the case.
Subnational violations would not be co-ordinated, and hence they represent non-optimal strategies in the sense of the two-person game we have been considering. On the other hand, violations planned at the national level could represent an optimal strategy, so there is no reason to deviate from the equilibrium inspection strategy.

In conclusion, the choice of inspection strategies for detecting undeclared activity is, and will remain, a subjective one. However, the subjective elements can be pinpointed and related in a precise and consistent way to the quantitative sampling plans that must ultimately be defined. This, we feel, provides a transparent and honest approach to a difficult but very important problem in verification.

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IAEA-SM-333/38P

CAPABILITY OF TRACE ANALYSIS TO REVEAL UNKNOWN NUCLEAR ACTIVITIES

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In a recent IAEA Consultants Group Meeting, the capability to detect fissile material and/or fission products in environmental samples for detecting clandestine nuclear activities was discussed [1].

At the Institute for Transuranium Elements, analytical techniques have been set up to reveal the origin and history of unknown nuclear samples, e.g. vagabond nuclear materials.
The methodology is analogous to that employed in archaeology: to determine the origin by the determination of the characteristic impurities of the material and to measure the age of the sample from the radioactive decay or buildup of nuclides.

The inherent consistency of the isotopic concentrations that can be observed after a nuclear process has taken place provides another possibility to reveal the history of a sample [2]. A prominent example of this was the description of the Precambrian Oklo event [3].

To perform this kind of investigation, different analytical tools have been set up in our laboratories. In particular, on account of their high sensitivity and multi-isotopic capabilities, the techniques of inductively coupled plasma mass spectrometry (ICPMS) and glow discharge mass spectrometry (GDMS) have been employed for trace element analysis.

For many years, plasma sources have been employed principally as sources for optical emission spectroscopy [4–6]. More recently these sources have been coupled with mass spectrometry [7].

Compared with conventional optical techniques, mass spectrometry has the advantages of increased sensitivity, obtained by direct sampling of the ions, and a much simpler spectrum, which makes a wider range of elements available to the analyst. This is also accompanied by much simpler quantification. Plasma sources coupled with mass spectrometers have thus dramatically extended both elemental coverage and detection limits in quantitative trace element analysis. This is true both for GDMS and ICPMS techniques.

An ICPMS instrument has been coupled on-line to a chromatographic system and installed in a glove box [8]. By means of this system, trace analysis of fissile material and fission products can be performed [9, 10] and isobars can be separated by the chromatographic facility, preceding mass spectrometry detection [11].

A GDMS instrument, after adaptation in a glove box, has been employed for the direct analysis of solid samples [12]. By this technique, the sample preparation is very quick and straightforward. Only cutting to rods or discs or fracturing of pins is required before analysis. Pre-sputtering can be used to remove surface contaminants to a large extent, thus minimizing the problem of contamination. The wide dynamic range of the detectors allows the determination of both major components and trace constituents within the same analytical cycle. The decoupling of the atomization and ionization processes results in uniform sensitivities for many elements and also minimizes matrix effects.

As an example, a 'hot particle' could be characterized as an LWR MOX specimen by the ICPMS determination of fission products and actinides. An unknown specimen confiscated in illicit trade was recognized as a known fuel, using the results for impurities obtained by GDMS. This finding was supported by other characteristics of the material.

Using the above two techniques concentration levels of ppt (ng/kg) can be detected. Thus they have been applied, in active samples, for very low level detection of fissile materials and fission products of safeguards interest.
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IAEA-SM-333/114P

A CONCEPT FOR COLLABORATIVE HUMAN/MACHINE PROBLEM SOLVING IN NUCLEAR NON-PROLIFERATION ANALYSIS

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The IAEA is attempting to strengthen safeguards and increase cost effectiveness. Among the measures that have been proposed are special inspections, early
provision and use of design information, universal reporting of export/import of certain equipment and materials, environmental monitoring, and enhanced analysis of the nuclear activities of States. By enhancing the analysis of safeguards relevant information, the IAEA seeks to provide early warning of possible nuclear related activities that are inconsistent with a State's obligations to use nuclear energy only for peaceful purposes. The goal of this research was to explore the application of automation technology to aid the IAEA Country Officer.

Considerable attention has been devoted to the identification of software tools to preprocess, analyse and synthesize large volumes of information to perform effective safeguards assessment. The trend toward increased automation means that the role of the user is shifting from perceptual tasks (e.g. finding keywords in a document, comparing photographs taken over time) to more cognitive tasks (e.g. recognizing patterns in data, drawing conclusions). Unfortunately, our system designs often fail to support the new roles of the user, perhaps because the automated feature is expected (unrealistically) to obviate human intervention. Thus, the full potential of new technology may not be realized because of poor interfaces with the operator turned evaluator.

Our approach was to examine user–system relationships that exploit the capabilities of the human as well as that of the machine. This approach has the goal of creating ‘collaborative human–machine systems’ that will perform better than either man alone or machine alone might perform. To guide the design of such systems, emphasis is placed on knowledge of the user's goals and intentions, available tools, procedures and contextual information. This is a balanced exercise in designing appropriate, intelligent user interfaces and in defining and identifying appropriate software tools and resources.

The following opportunities for intelligent support for the IAEA Country Officer were identified:

— **Information filtering and retrieval**: Automated systems for filtering or retrieving information based on keyword searches or more sophisticated semantic features (in the case of textual documents); and systems for identifying and categorizing non-textual information (such as geographical information systems).

— **Analysis support**: Tools that help the Country Officer to recognize patterns in safeguards data and to identify safeguards relevant features and trends. One example is a computer assisted system for bringing likely data to the attention of the Country Officer; this is based on rules of thumb and established criteria (such as matching import/export information on the nuclear supplier's group list). Another example is automated support for 'what-if' analyses (through computer simulation and sensitivity analysis) to test hypotheses about clandestine weaponization activities based on material flow data or material unaccounted for (MUF).
— Decision making support: Tools that help the user/decision maker to organize evidence collected from a variety of sources over extended periods of time. One example is the establishment of a note card system to annotate and associate bits of related documents using a note card filing system metaphor. The note card database would provide links back to source documents as well as links forward to the 'arguments' or 'cases' that the Country Officer is in the process of developing. Each such case would also be organized in a database that presents the logical process in a consistent fashion, together with annotation by the user and links back to the note card database and the original source information.

— Portfolio management support: Development of tools that will help to organize the various types of information, reports, critical requirements and information needs, in support of the day to day administration and maintenance of the Country Officer's file.

IAEA-SM-333/125P

PAIR:
PLANNING AND ANALYSIS OF INSPECTION RESOURCES*

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1. INTRODUCTION

The safeguards inspection effort of the IAEA consists of physical inventory verification (PIV) to close the annual material balance, interim inventory verification (IIV), conducted mainly to satisfy the timeliness criteria, flow verification (FV) to

* Work performed under the auspices of the United States Programme of Technical Assistance to IAEA Safeguards (POTAS).

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verify the transfer of material, and containment and surveillance (C/S) activities, helping to preserve the continuity of knowledge concerning the material. Estimating the required overall future inspection effort under a variety of conditions is an important part of safeguards planning of the IAEA.

During the past few years, the Safeguards, Safety and Nonproliferation Division at Brookhaven National Laboratory has been developing the computer program PAIR (planning and analysis of inspection resources) to facilitate the resource estimation process by providing a user friendly, computerized version of the review, planning and estimation procedures [1, 2]. PAIR is designed primarily for use by managers and systems analysts in the IAEA Department of Safeguards.

2. STRUCTURE OF PAIR

For each facility under safeguards, PAIR accumulates the effort for all types of routine inspections and provides graphical representations of the results. Using as input the list of facilities under safeguards, their characteristics and the inspections to which they are subject, the program calculates the total annual inspection effort (total AIE) and disaggregates it by

- Country (State)
- Facility category, subcategory and type
- Safeguards Agreement (NPT, NNPT or other)
- IAEA operations division, section and country.

The main underlying databases, supplied by the IAEA for actual analysis, are:

(a) A *facility database*, listing all the facilities under safeguards or anticipated to come under safeguards, with their relevant safeguards characteristics;
(b) An associated *inspection effort database*, containing a reference level inspection effort (currently based on the previous year's person-days of inspection (PDI) for the inspections specified above;
(c) An *inspection practices database*, containing information on the number of inspectors used for each type of inspection, for each country and each plant type (or subcategory);
(d) Various *criteria parameter databases*, dealing with safeguards criteria [3] parameters, such as detection goals and timeliness requirements for various kinds of material, and modelling parameters needed to evaluate new safeguards approaches such as random inspections and a zone approach;
(e) A *facility parameter database*, containing statistics based model facility parameters for fuel fabrication plants handling natural uranium or low enriched uranium (LEU) and for enrichment plants. The parameters are used to estimate the inspection requirements for new plants of these types.
3. PAIR ALGORITHM

The main algorithm in PAIR is the total AIE (for the year chosen by the user), which is given by:

\[ \text{Total AIE} = \sum_{\text{all facilities } j} \text{individual AIE}_j \]

where

\[ \text{AIE}_j = \text{PIV}_j + \text{IIV}_j \text{NIV}_j + \text{FV}_j \text{NFV}_j + \text{CS}_j \text{NCS}_j \]

Here, \( \text{PIV}_j, \text{IIV}_j, \text{FV}_j \) and \( \text{CS}_j \) are the inspection efforts (in PDI) for different inspection types, \( \text{PIV}, \text{IIV}, \text{FV} \) and \( \text{C/S} \), at the \( j \)-th facility, while \( \text{NIV}_j, \text{NFV}_j, \text{NCS}_j \), etc., are the corresponding numbers of such inspections. These quantities are all listed in the inspection effort database or are modelled through the facility parameter database. \( \text{AIE}_j \) is the total annual inspection effort at the \( j \)-th facility.

This formulation corresponds to a division of the inspection effort among these four types of inspection. Although such a strict division may not reflect the actual practice at all facilities, it is reasonable for an approximation of the PAIR program and generally reflects IAEA data.

The following points should be noted in regard to this main algorithm:

(a) \( \text{AIE}_j \) is zero if the plant has not yet begun operation or is not yet under safeguards.

(b) Only an aggregate figure \( \text{AIE}_j \) is used:
- for those facilities for which detailed inspection or modelling information is unavailable; and
- for some bulk handling facilities for which experiential information has shown substantial fluctuations in the required effort, or for which the safeguards approach is highly plant specific, so that any statistically based algorithmic formulas are questionable.

4. ANALYSIS OF CHANGES

After the relatively straightforward calculation of a reference case inspection effort, PAIR permits studies of the effects of additions and changes, and the use of certain algorithms:

(a) Additions to and changes in the facilities and their characteristics and in the associated inspection efforts;
(b) Changes in the inspection practices of the operations divisions, e.g. the number of inspectors used in various countries and for different facility subcategories;

(c) Changes in the timeliness parameters for unirradiated direct use material, irradiated direct use material, and indirect use material — by changes in the frequency of IIV;

(d) Changes in the detection goals, i.e. significant quantities of special nuclear materials and the associated detection probabilities through a linear approximation to the usual sampling formulas;

(e) Algorithms based on the statistical analysis to allow the estimation of the anticipated effort for new natural U and LEU fuel fabrication plants [4] and similarly for new enrichment plants [5].

(f) An algorithm to deal with the randomization of the IIV at specified facilities in a country, based on an elegant model [6];

(g) Application of the zone approach to facilities in a given country.

In all cases, the user may vary the magnitude of the changes by altering the parameters; such alterations can be made on a trial basis or on a permanent basis.

In effect, PAIR uses the information described above to determine the required AIE for these 'what if' situations. The same representations as mentioned in Section 2 are also available for these situations.

FIG. 1. Total AIE for the year 2000, broken down by country (base case, total annual PDI = 11 260).
FIG. 2. TOTAL AIE for the year 2000, broken down by facility categories (base case, total annual PDI = 11 260).

FIG. 3. Total AIE for the year 2000, broken down by facility categories (total annual PDI = 9562).

The spent fuel timeliness parameter for category A is changed from 13 to 26 weeks, and for facilities in categories D and E there are random flow verifications with an execution probability of 0.8.
5. SAMPLE RESULTS

Figures 1–3 show sample results based on data from a test database constructed at Brookhaven National Laboratory for about 500 facilities, as mentioned in the IAEA Annual Report [7] and in the World Nuclear Industry Handbook [8]. Figure 1 shows the total AIE for the year 2000, broken down by country (for the 25 countries with the largest safeguards inspection requirements). Figure 2 shows the same data, broken down by facility category. Figure 3, which is also a breakdown of the total AIE by facility category, shows the effect of changes in safeguards criteria parameters, i.e. the effect of a change in the timeliness goal from 13 weeks to 26 weeks for spent fuel at power reactors (Category A), as well as the independent effect of the randomization of flow verifications at fuel fabrication plants (Category D) and reprocessing plants (Category E).

6. DISCUSSION

As exemplified by the sample results shown in Figs 1–3, PAIR provides a straightforward means to analyse 'what if' situations in safeguards implementation. Thus, it permits managers and analysts to study future scenarios and their effect on human resources. It is planned to introduce into PAIR a direct capability for studying costs associated with these hypothetical changes in safeguards implementation. In this way, it would be easier to use PAIR in assisting the IAEA Safeguards Department in its current programme of investigating new safeguards approaches.

The most difficult challenge to the use of PAIR is the formulation of a comprehensive set of data for facility characteristics and facility inspection effort by inspection type. When this is done, each run of PAIR requires about one minute on a 386 personal computer. PAIR uses CLIPPER 5.0 software and dBASE IV database files. The graphical output generated by dGE 4.1 software is an integral part of the program operation.

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SENSITIVE FIELD ALPHA CONTAMINATION MONITORING FOR SPECIAL INSPECTIONS AND NON-PROLIFERATION VERIFICATION

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1. INTRODUCTION

Sensitive alpha detectors that can be used in the field offer important advantages in detecting the spread of nuclear material and identifying the existence of clandestine nuclear operations. The alpha decays from small amounts of uranium, plutonium or other nuclear material may be undetectable by the creator or carrier of the material, but the alpha signature (if it can be detected) could identify nuclear activities. Currently, the most sensitive method for monitoring suspected locations is to sample suspicious material for later analysis. However, the delay involved in getting results back from a laboratory makes it difficult to follow up on a clue or to know if it is necessary to take more samples. An accurate and reliable method of alpha detection in the field offers an investigator the opportunity to make these decisions as the inspection progresses rather than after the event.

2. THE LONG RANGE ALPHA DETECTOR

All traditional alpha detectors detect the alpha particle directly. Thus, they are limited by the short range of alpha particles in air. In contrast, the long range alpha
detector (LRAD) system detects the ionized molecules created by the interaction of the alpha particle with the air. The LRAD, therefore, is limited by the distance that the ion can travel (1–100 m) rather than the shorter distance (2–3 cm) that the alpha particle travels. Two basic types of LRAD are illustrated in Fig. 1 and described in detail in Refs [1] and [2].

In an object monitor (Fig. 1(a)) the contaminated object is placed in a closed chamber. Alpha particles emitted from its surface create a cloud of ions around the object. These ions are transported into an ion detector by an air current created by several small fans. The ion current (which is directly proportional to the amount of contamination) is measured with a sensitive electrometer. In a surface monitor (Fig. 1(b)) the contaminated surface forms one side of the enclosed volume. The alpha particles emitted from a surface also form a cloud of ions in a surface monitor, but in this case the ions are transported by an electric field rather than an air current. Again, the ion current is measured with a sensitive electrometer. The ions created by all of the surface contamination within the enclosure are measured simultaneously, generating a single measurement of all of the contamination in the chamber.

3. RESULTS

Although the LRAD has not been tested in the field as an inspection instrument, it has been used extensively in the field as an environmental surveillance moni-
FIG. 2. Response of LRAD to blasting-pad contamination.

tor. An example of a field monitoring result is illustrated in Fig. 2. The surface investigated was a blasting pad used for uranium testing nearly 50 years ago. As is common with such locations, very little was known about the quantity or extent of contamination.

The small cross-hatched rectangle near the bottom centre of the figure represents a concrete lined catch basin in the pad and is included for location reference only. Data were acquired for about 15 min at each location indicated by a dot in the figure. The contour lines and shades of gray were interpolated between these points by a computer. Note the two ‘hot spots’ to each side of the catch basin. These results are particularly significant since the uranium in these two spots had been exposed to the weather for 50 years. Most of the remaining data represent variations in naturally occurring uranium and thorium background radiation.

4. CONCLUSIONS

The LRAD has four significant advantages over traditional alpha detectors for field operations:

— It is 10–100 times more sensitive than traditional field instrumentation. Thus, the LRAD can be used to detect very small quantities of contamination that might have been missed by any previous inspection.
— It is completely automated and mobile. It can easily be operated in remote locations by personnel who are not highly trained.
— It is very rugged. Unlike traditional alpha detectors, it has no fragile windows and no fine wires. The detector is very hard to damage and easy to repair if damaged.
— It is sensitive to all alpha contamination on a large surface or a complex object. Large deposits of low level contamination that would be missed by traditional detectors are easily observable with a LRAD.

The LRAD offers laboratory quality alpha contamination measurements from a rugged and reliable field monitor. This combination of sensitivity and portability is not available from any other alpha detection system.

**REFERENCES**


**IAEA-SM-333/134P**

**HUMAN FACTORS, HUMAN RELIABILITY IN SAFEGUARDS AND THE POSSIBLE APPLICATION OF FUZZY SET THEORY**

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1. INTRODUCTION

The International Workshop on the application of fuzzy set theory in the field of nuclear engineering and safeguards (chairman: Y. Nishiwaki, scientific secretaries: A. Fattah, IAEA, and T. Onisawa, Japan) was held on the occasion of the 5th World Congress of the International Fuzzy Systems Association (IFSA) in Seoul, Republic of Korea, in July 1993. In the light of some of the discussions and papers presented at the Workshop, the possible application of fuzzy set theory in the field of safeguards will be discussed.
To design an effective verification system of safeguards, one must identify possible ways and means by which nuclear material could be diverted from peaceful uses, including means to conceal such diversions. These theoretical ways and means, which have become known as diversion strategies, are used as one of the basic inputs for the development of safeguards procedures, equipment and instrumentation. As a result of diversion and its concealment or other actions, anomalies will occur. All reasonable diversion routes, scenarios/strategies and concealment methods have to be taken into account in designing safeguards implementation strategies to identify anomalies. Precise information vital for such assessments and analyses is normally not available or, if it is, difficult and expensive collection of information might be necessary. Above all, realistic appraisal of truth needs sound human judgement and human reliability under such fuzzy environment. However, human factors or human reliability may be affected by various conditions — physical, psychological, physiological, environmental, etc. — fatigue, stress, learning, experience, personality, preference, level of consciousness, degree of responsibility, type and quality of work, degree of comfort, chance phenomena, etc. All these numerous factors may directly or indirectly influence human judgement, human reliability and performance. The reliability of detection of an anomaly or abnormal events is low when the amount of information is too little because too much guesswork is involved. With an increase in information, the reliability may increase to a certain level, but, with excess information, confusion may arise and the reliability may drop. It is doubtful whether these complex factors affecting human reliability and human judgement could be adequately represented by probability theory.

Probability theory deals with the definition and description of models involving the probability concept. Probability judgements are concerned with repetitive events which have basic similarity. Most applications of probability theory may be interpreted as special cases of random processes. If, as in many games of chance, equal probabilities are assigned to each of the simple events of a given finite fundamental probability set, then the probability of realizing a compound event (success) defined as the union of specified simple events (‘favourable’ simple events) can be computed as the ratio of the number of favourable simple events to the total number of simple events. It is on the basis of these considerations that probability is deduced by resolving the various outcomes into a number of equipossible alternatives. However, when we apply the term probability to a non-repetitive event or an isolated case, for example the probability of ‘Julius Caesar’s visit to Great Britain’ or the probability of a certain country secretly producing nuclear weapons, it is impossible, at any rate in any obvious way, to generate a sequence of trials and thus measure the probability of its occurrence by means of a frequency ratio. We have, therefore, to estimate the probability by a more or less subjective intuitive appraisal of such evidence as we may have. Since in such cases a universally acceptable quantitative estimate of the degree of our confidence in the statement cannot be given, probability, when used in this sense, cannot form a part of a scientific assertion in the conventional sense.
To avoid confusion, it is, therefore, better to restrict the use of probability in a scientific sense to repetitive events only and to use the term ‘possibility’ when we wish to speak of our expectations of non-repetitive events, as suggested in fuzzy set theory. In addition to the probability and possibility measures, belief, credibility, certainty, necessity, plausibility measures, etc. are also introduced as special cases of fuzzy measures.

2. FUZZY MEASURES AND FUZZY INTEGRAL [3, 4]

2.1. Fuzzy measures

A fuzzy measure of Sugeno is an extended probability measure in one sense, which assumes, in general, only monotonicity without additivity.

Let $X$ be a universal set and $\mathcal{B}$ be a Borel field. Then a set function $g$ defined on $\mathcal{B}$ with the following properties is called a fuzzy measure:

(a) $g(\emptyset) = 0$, $g(X) = 1$ (boundedness);
(b) If $A, B \in \mathcal{B}$ and $A \subseteq B$, then $g(A) \leq g(B)$ (monotonicity);
(c) If $F_n \in \mathcal{B}$ for $1 \leq n < \infty$ and a sequence $\{F_n\}$ is monotone (in the sense of inclusion), then $\lim_{n \to \infty} g(F_n) = g(\lim_{n \to \infty} F_n)$ (continuity).

A triplet $(X, \mathcal{B}, g)$ is called a fuzzy measure space, and $g$ is called a fuzzy measure of $(X, \mathcal{B})$.

For applications, it is enough to consider a finite case. Let $K$ be a finite set $K = \{s_1, s_2, \ldots, s_n\}$ and $P(K)$ be a class of all the subsets of $K$. Then a fuzzy measure $g$ of $(K, P(K))$ is characterized by the first two properties since the third one implies continuity.

In particular, $g(\{s_i\})$ for a subset with a single element $s_i$ is called a fuzzy density like a probability density. We denote $g^i = g(\{s_i\})$. As a special form, $g_\lambda$ fuzzy measures have the following characteristics:

$$A \cap B = \emptyset \Rightarrow g_\lambda(A \cup B) = g_\lambda(A) + g_\lambda(B) + \lambda g_\lambda(A) g_\lambda(B)$$

where $-1 < \lambda < \infty$.

When the fuzziness coefficient $\lambda = 0$, $\lambda$-fuzzy measures are probability measures ($\lambda > 0$, belief measure; $-1 < \lambda < 0$, plausibility measure).

2.2. Fuzzy integral

Let $h$ be a measurable function from $X$ to $[0, 1]$. Then the fuzzy integral of $h$ over $A$ with respect to $g$ is defined as
\[
\int_A h(x) \circ g = \sup_{\alpha \in [0,1]} \{ \alpha \land g(A \cap F_\alpha) \}
\]

where \( F_\alpha = \{ x, h(x) \geq \alpha \} \) and \( \land \) stands for minimum.

In the above definition, \( A \) is the domain of a fuzzy integral which is omitted if \( A \) is \( X \).

Now let us see how to calculate a fuzzy integral. For simplicity, consider a fuzzy measure \( g \) of \((K, P(K))\) where \( K \) is a finite set previously defined.

Let \( h: K \to [0,1] \) and assume without loss of generality that \( h(s_1) \geq h(s_2) \geq \ldots \geq h(s_m) \). Renummer the elements of \( K \), if they are not in descending order. Then we have

\[
\int h(s) \circ g = \bigvee_{i=1}^{n} [h(s_i) \land g(K_i)]
\]

where \( K_i = \{s_1, s_2, \ldots, s_i\} \) and \( \bigvee \) stands for maximum.

A fuzzy integral can be used as a model of subjective evaluation of fuzzy objects where the attributes of an object are measured by a fuzzy measure and the evaluation function of an object is integrated with respect to a fuzzy measure. In the example relating to the appropriateness of a safeguards methodology let \( h: K \to [0,1] \) be the evaluation function of the methodology, i.e. the function expressing the characteristics of the methodology. For example, we set \( h (\text{"sensitivity"}) = 0.9 \) and \( h (\text{"price"}) = 0.3 \), etc. Then the overall evaluation of the methodology is given by the fuzzy integral of \( h \) with respect to \( g \), i.e. the grade of subjective importance of each attribute. As is clear from the definition of a fuzzy measure, a fuzzy measure is not only a subjective scale for guessing whether an a priori non-located element in \( X \), a universe, belongs to a subset \( A \) of \( X \), but is also concerned

\[ FIG. 1. Example of a fuzzy integral. \]
with such cases as the grade of subjective importance of an attribute referred to in
the foregoing from a practical point of view. The fuzzy integral model is applicable
to non-linear cases, where one does not have to assume independence of one attribute
from another. In the example given it is highly possible that there is a certain depen­
dence between sensitivity and price. If this is the case, we cannot use a linear model
inasfar as we regard sensitivity and price as the attributes of the methodology.

We have to consider the dependence of attributes from two points of view. One
is objective dependence such as between sensitivity and price, and the other is subjec­
tive dependence. Even if an attribute seems physically independent of another, one
may consider that they are subjectively dependent in some cases.

The uncertainties involved in human judgement are usually non-additive and
fuzzy, and therefore could be treated more adequately with fuzzy set theory. An
example of a fuzzy integral is shown in Fig. 1.

3. OPTIMIZATION IN A FUZZY ENVIRONMENT [3-7]

According to Bellman and Zadeh, if one assumes $X$ to be a set of alternatives
that contain solutions of a given multicriteria optimization problem, then, if the goals
and constraints in a fuzzy environment formally have the same nature, they may be
represented by fuzzy sets on $X$.

The fuzzy goals $G$ and fuzzy constraints $C$ may be defined as fuzzy sets with
membership functions $\mu_G(x)$ and $\mu_C(x)$ respectively. When goals and constraints
have the same importance, the following fuzzy set $D$ on $X$ may be called a fuzzy
decision:

$$D = \left( \bigcap_{i=1,m} C_i \right) \cap \left( \bigcap_{j=1,n} G_j \right)$$

The membership function for $D$ may be expressed as:

$$\mu_D(x) = \min \left[ \min_{i=1,m} \mu_C(x), \min_{j=1,n} \mu_G(x) \right]$$

In other words, a fuzzy decision $D$ is considered as an intersection of given goals
and constraints.

$$\mu_D(x) = \mu_C(x) \land \mu_G(x) \quad (\land \text{ means minimum})$$

In this case an optimal decision $x^*$ may be defined as any alternative in $X$ which
maximizes $\mu_D(x)$.

$$\mu_D(x^*) = \sup_x \mu_C(x) \land \mu_G(x)$$
This definition of optimal decision means that the grade of membership of the optimal decision $x^*$ in $C$ is the same as the grade of membership of $x^*$ in $G$. A more detailed discussion of fuzzy optimization will be found in Ref. [4]. An example of an optimal decision is shown in Fig. 2.

4. DISCUSSION AND CONCLUSION [1, 4, 7-12]

In safeguards, sometimes, cost-benefit or cost effectiveness analysis could be based upon optimization of allocation of resources in different fields. Optimization models in engineering design usually assume that the data are precisely known, that constraints delimit a non-fuzzy set of feasible decisions, and that criteria are well defined and easy to formulate. However, in the real world such assumptions usually hold only approximately, especially when safeguards problems are involved. Optimization and decision making in a fuzzy environment may have special advantages when dealing with problems concerning safeguards system design. It is becoming increasingly clear that many real-world problems faced by implementation of safeguards concern fuzziness rather than randomness.

REFERENCES


IAEA-SM-333/136P

AN OBJECT ORIENTED APPROACH TO VERY LARGE SYSTEMS FOR SAFEGUARDS APPLICATIONS

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1. BACKGROUND

The IAEA has had much experience over the years with software development, both internally and externally. Such software has generally been associated with
instruments provided for analysis or used in connection with office automation. In the past several years we have tried to improve the effectiveness and to reduce the cost of software by examining development efforts, short lived products, poor user acceptance and some possible causes for redundant design. The modularity and flexibility of the products are increasingly discussed because they are good indicators of the usefulness of the product [1]. It is clear that we need new approaches to provide software that is more useful to the IAEA.

1.1. Modularity

A modular program is said to possess two very important characteristics — high cohesion and low coupling. These characteristics refer to the extent to which a program does a well defined task, and the connections provided to other systems. A system that does many tasks which are all interrelated has low cohesiveness and high coupling. Obviously, a system that does one task and has limited coupling has high cohesiveness and low coupling. UNIX operating system tools are an example of modularity with high cohesiveness and low coupling.

1.2. Flexibility

Programs are often required to perform tasks for which they were not originally designed. In fact, the utility of a program can be described by the generality of its use. Thus, a program that can only be used in one instance of a problem has low flexibility, while a program that can be applied to many different problems has high flexibility. A computer operating system can exhibit extreme flexibility, allowing multiple diverse programs to be run concurrently. High coupling in the form of a command language provides the most flexibility.

1.3. Usefulness

The way in which modularity and flexibility interact determines to a large extent the long term usefulness of the program [2]. Flexibility means that the program can be applied to other instances of similar problems as well as when the existing problem changes, while modularity means that the program can be easily expanded to increase its capability and therefore its applicability to other problems. It follows that programs with high modularity and high flexibility are needed to decrease the costs associated with software development, but it is evident that modularity and flexibility are trade-offs. With conventional methods, it is not possible to increase one of them without decreasing the other. We start out by creating modules that perform specific tasks. Then we join these modules together by creating connections between them to provide flexibility. However, the more connections we create, the less cohesiveness we have for each module and the more coupling we
have between modules. Again we are faced with the fact that a change in one module affects several other connected modules, requiring larger development efforts to accomplish smaller results and decreasing flexibility.

2. DETRACTORS FROM MODULARITY AND FLEXIBILITY

The first effort at building large systems usually begins with a batch program. Given that there is a program to compute payroll and another one to print cheques, the combination of the two programmes (and their associated databases) into a system to pay people is an example of a large system. Batch programs can execute batch level commands and therefore execute programs in a predetermined sequence. There is no inter-program communication, and programs cannot control the execution of other programs (only the batch level program can). Extensions of the batch processing capabilities provide for program status codes, symbolic macro substitution and some limited branching capabilities. These batch level programs are evident in the major existing systems. Their drawbacks are that the processing is hierarchical (programs are called by batch programs) and serial (batch programs run each command in sequence). Associated with these programs are also data files. Unfortunately, these files are all too often structured hierarchically. In the IAEA this is evident in binary files created by instrumentation systems as well as ADABAS files used for errors and state accountancy. These two characteristics, hierarchical structure and serial operation, account for much of the lack of modularity and flexibility in existing systems.

2.1. Hierarchical approach

A hierarchical database system cannot be easily expanded because the data structure must be known by the programs in the system [3]. This is in contrast to a relational database where the data structure is unknown by the programs in the system. Many large systems (and most existing safeguards systems) are really collections of programs and data that are used together to accomplish a specific task. The functions within programs are tightly coupled (cohesive) and represent the building blocks of the system. The interaction between programs is simply through sharing of data files or via a user interaction. Thus, a restriction is put on the use of each program; the programs must be used in their entirety in a predetermined and relatively inflexible manner. The data used by these systems are separated from the functions. The data also generally exhibit the inflexibility of a hierarchical (instead of relational) model. Many of these systems were designed when the approach of collecting large amounts of data that could be organized in a specific way was thought to be capable of providing long term satisfactory results. It has been shown that such systems fail miserably in an environment which forces change. They simply cannot
provide the flexibility required, are not expandable, and in a changing environment cannot even be implemented [4].

2.2. Serial approach

Early systems also exhibited an inclination toward serialization of the process [5], i.e. they expected the process to proceed from an initial stage through several intermediate stages to the final output. Such systems provide little ability to backtrack, start from the middle, or otherwise rearrange the sequence of operations. Again, a simplistic model that worked for small problems cannot provide the interoperability required of larger complex systems. Large complex systems require flexibility in task management if they are to approach any level of efficiency, or else they are constantly hampered by the lack of intermediate results or errors in intermediate operations (the failure analysis problem for complex systems).

3. VERY LARGE SYSTEMS

Building very large systems requires the concepts of co-operating systems, object oriented approach and expert systems. These attributes are indicators of high modularity and high flexibility. Co-operating systems are systems which contain multiple concurrent programs and databases that co-operate to perform a larger task [6]. The programs have little restriction on their functional combinations to create solutions to new problems. Any combination of functions from any of the multiple programs can be combined to create a new solution. Each program module represents a set of capabilities — its functions, which it makes available to other programs in the system via a messaging construct. This is the basis of object oriented programming, extended to programs as objects.

3.1. Object oriented approach

In the past several years, computer science has become obsessed with the object oriented approach [4]. This approach has found its way into systems analysis (Booch), languages (SmallTalk, C++) and databases. The reason is simple: object orientation provides the highest combination of modularity and flexibility. By combining data with functions, supporting inheritance and polymorphism, and providing a messaging connection that supports a language, one can create a very modular system which has a high degree of flexibility. It has even been proposed that any system comprised of over 100 000 lines of source code must take an object oriented approach to be successful (most popular PC applications now exceed 100 000 lines of source).
3.2. Expert systems approach

Another approach to delivering systems that has gained wide acceptance is the rule based expert system [7]. This system is designed around ‘if–then’ rules, which are grouped to provide knowledge banks. These knowledge banks are then used by an inference engine to solve problems in complex paradigms. Their success lies in the fact that they provide one of the highest degrees of flexibility attainable, in the form of rules, and are modular in that rules are grouped. Most of these knowledge banks are also interactive and provide immediate feedback and opportunity for alteration by the users, thus increasing flexibility even more. The visibility and comprehensibility of rules also facilitate the capture of complex information from experts and the verification and validation of the system.

4. LARGE SYSTEM APPLICATIONS

Given that one uses an object oriented approach and joins information from programs together with an expert system, what then are the guiding rules for building the system? Can one simply begin creating a system from rules and objects, to be expanded at will in any direction that seems appropriate? There seem to be two important guiding factors in building very large systems, using an evolutionary approach [8], which need to be followed. These factors are the concepts of user paradigms and combining outputs.

4.1. User paradigms

Some early efforts regarding large systems made the mistake of trying to combine paradigms. A paradigm is the conceptual model used to operate the system. For instance, a spreadsheet, a word processor and a database are user paradigms. Each of these three program types has a model which it follows. Essentially all spreadsheets follow the same model, as do word processors, databases, and others. There exist several well defined paradigms which users expect to see when operating a software program that is a member of it. Any attempt to combine these paradigms may lead to disastrous results for the user. The user has a method of dealing with a problem which has been developed in the field. A software program must respect this paradigm if it is to be successful. Other useful paradigms include user windowing interfaces, computer aided design (CAD), geographical information systems (GIS), relational database systems (RDBMS) and expert systems. These are not only classifications of the software programs, they are descriptions of paradigms with which the user expects to operate. The early systems referred to above tried to switch between paradigms using the application programs themselves. The idea was that this would make it easier for the user (and more lucrative for the developer). The user
community reacted quickly and strongly to this approach of 'all things to all people' and rejected such approaches in favour of the now prevalent open systems philosophy. The message was to use simple, operating systems to control paradigm switching — not applications.

4.2. Combining outputs

Another aspect of systems is their result or output. In the case of homogenous outputs, these outputs can simply be combined. For quantitative output, the output of one module can be used as input to another module. This provides the required coupling between modules. It is in the case of diverse, non-quantifiable output that problems arise. The expert system approach shows a great advantage in this respect, in that it can combine the outputs of several diverse modules logically to present an overall picture of the results [9]. In addition, expert systems can be used to reason with uncertainty when they are provided with the relevant implementation (this is a topic of other IAEA papers regarding Bayesian inference, Dempster–Shafer theory and defeasible logics).

5. PLATFORM REQUIREMENTS

Creating large co-operating systems also puts demands on the platform (computer operating system and hardware). On systems equipped with multitasking capabilities the operations of programs need not be sequential or hierarchical. The capabilities of inter-process communication and remote procedure call (RPC) are integral to allowing programs to co-operate. Such systems are able to share computing resources between concurrently running programs and to provide standards for applications that are equipped with an inter-process communications capability. Thus, programs can be viewed as a set of functional resources which can be combined to solve specific problems. Most large systems have this capability. But writing programs that support inter-process communication requires additional efforts. These programs must be able to interface with the operating system and to support the protocols necessary for another program to start and execute its command functions. In addition, the program must be well behaved and conform to operating system standards so that it can coexist with other programs that may need to share resources. For a single computer system, supporting inter-process communication for all functions of a program is a prerequisite for large systems.

5.1. Scaling to networked large systems

Systems which consist of multiple computers are usually connected via a network. The analogous co-operation capability is called RPC. This capability enables
a program to execute another program on another computer via the network. Database systems which include network servers are an example of RPC implementation. Extensions to the RPC protocols have led to network level inter-process communication capabilities which closely parallel those of single computer systems. For a networked multiple computer system, a supporting RPC is a prerequisite for large systems.

6. A NEW ARCHITECTURE

Using program functions as building blocks, one can develop an architecture for producing large systems. First, the existing paradigms for those tasks which are required must be identified. No attempt is made to alter or mix these paradigms; rather, a leading implementation of each paradigm is simply installed. Second, the results of these paradigms are combined in an expert system that has the ability to reason with uncertainty. Such a system is able to accept a diversity of input and to provide an explanatory output. The ability of the expert system to evaluate alternative reasoning paths and input data is an essential part of the system operation and is not necessarily required for its explanatory output. The capability of the expert system is used to provide coupling of each of the modules in a way that is visible, flexible and expandable. Third, the system is implemented on a platform which provides multitasking, inter-process communications and RPC. Such a platform would provide an operating system which standardizes these capabilities for all applications. Such a system would be flexible in that additional modules could be added in order to increase the system's capability. This system would be modular in that each paradigm would exist on its own, with interaction between paradigms provided by the expert system.

6.1. Implementation issues

The flexibility of an object oriented/expert system approach such as the one advocated here provides insurance against making the wrong initial choices for data structures, elements, etc. The modularity of this approach provides the ability to benefit from the system almost immediately, since one can begin with a single module. There is no need for a large, long development or integration effort, with the pot of gold promised at the end of the rainbow. In fact, implementing such a system would be similar to operating it, in that a co-operative, multitasking effort would be used. There would be no need for the hierarchical waterfall approach to developing large systems, which has so often failed in the past.
7. CONCLUSIONS

Recent applications of expert systems and object oriented techniques in the nuclear industry are encouraging [10-12]. Through the use of programs as objects, a modern operating system, and an expert system capable of reasoning with uncertainty, the prospect of building very large systems is bright. Future programs should be designed with these capabilities so that they can be integrated into such a system in order to add an ever increasing informatics capability which can help improve the efficiency and effectiveness of safeguards.

REFERENCES

SAFEGUARDS EFFECTIVENESS ASSESSMENT SYSTEM (SEAS): MORE THAN A DOCUMENT RETRIEVAL SYSTEM

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1. INTRODUCTION

1.1. Purpose

In order to increase the transparency and to improve the uniformity of safeguards activities at nuclear facilities, a variety of documents have been produced in the last few years. One consequence of this expansion of safeguards related reference material is the difficulty to collate this disparate information. The primary goal of the development of the safeguards effectiveness assessment system (SEAS) — a collaborative work of the Safeguards Effectiveness Evaluation (SEE) Section and the Data Processing Development (IDD) Section — is to provide the possibility of linking safeguards related information sources and to enable safeguards staff to access quickly information related to the description of IAEA verification activities.

1.2. Motivation

The need for a document information system became apparent several years ago. There are a number of safeguards manuals that inspectors may have to consult during their inspections. Finding the required information is sometimes difficult and time consuming. It is desirable to have a system that allows the user to find this information quickly, rather than having to search for it in various manuals. Also, there are computers in the facilities that are inspected, or the inspectors have portable computers. Safeguards manuals are seldom taken along, since they would increase the volume and weight of the equipment that has to be transported. A computer based system allows inspectors to have the documentary information available during their missions. The computer based electronic reference documents are available either on individual PCs or on local area networks (LANs). Furthermore, many of the reference documents may contain information on subjects that are not explicitly stated. For example, there may be information on physical inventory verification (PIV) activities in documents without the label PIV. Finally, information on safeguards
subjects is contained in various documents, and SEAS makes it possible to link these disparate information sources.

For the use of SEAS, inspectors will be provided with lists of the most obvious search criteria and will be able to formulate their queries by selecting keywords from each of these lists. This simple approach will make it possible for even the most inexperienced inspectors to find most of the information needed for their work.

1.3. Scope

We describe here the SEAS data and the SEAS application program, i.e. the mechanism serving as intermediary between the user and the data. Section 2 describes the source of the subject specific SEAS data and the process of transforming hard copy documents to computer data records, and discusses the logical view of SEAS data. Section 3 discusses design and implementation issues of SEAS as well as the user interface. Section 4 discusses experience with SEAS and the future direction of the project.

2. SEAS DATA

2.1. Source

The SEAS data provide information on the verification activities of the IAEA. The SEAS documents currently include the IAEA Safeguards Criteria for 1991–1995 [1], the evaluation procedures used by the SEE Section to evaluate the inspection goal attainment in accordance with the Criteria, relevant notes of the SEE Section for clarification of certain requirements of the Criteria, and selected parts of the IAEA Safeguards Manual (e.g. SMO 6.12) [2]. Verification methods and recommended instruments that will satisfy the requirements of the Criteria are expressed by 'mathematical formulas'.

2.2. Categorization

The information is stored in records; normally, one paragraph of a document corresponds to one record. A set of attributes is assigned to each record. The attributes serve as search criteria for the user interface. For example, paragraph 2.4 of Section 1 of the IAEA Safeguards Criteria [1] is identified by CRITERIA, LWR, PIV and CF. There are five attributes to categorize the information being retrieved:

- Source document
- Facility type
- Verification (type)
- Stratum (type of nuclear material)
- Goal attainment (level).
### Source Document

<table>
<thead>
<tr>
<th>CRITERIA</th>
<th>LWR</th>
<th>PIV</th>
<th>FH</th>
<th>Goal</th>
<th>Attainment</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>☐ Partial</td>
</tr>
</tbody>
</table>

#### FIG. 1. Example of SEAS query formulation: Source = CRITERIA AND Facility Type = LWR, Verification = PIV and Stratum = FH.

#### FIG. 2. Summary information of documents retrieved by the formulation of queries as shown in Fig. 1.
2. Physical inventory verification (PIV)

2.1 There is one PIV of a physical inventory taken by the operator (PIT) each calendar year during which the activities in 2.2-2.6 are carried out. The period between PIV's does not normally exceed 14 months [See Note (2) to 2.1]. The PIV is scheduled as in (a), (b) or (c):

(a) If the core is refuelled (or opened for another reason) during the year, the PIV is performed during the period the core is open;
(b) For those reactors which are not refuelled during the year, the PIV is done without opening the core;
(c) For multi-core facilities, a PIV is performed once a year, in connection with refuelling (or opening) of one core if it occurs, provided that the other cores are maintained under Acceptable C/S.

References: 2.2 2.6 Annex B

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FIG. 3. Another view of information of documents retrieved by the formulation of queries as shown in Fig. 1.

The user can choose any combination of these attributes as search criteria or he can leave blank any of the attributes; in the latter case, more than one record will be retrieved.

It was easy to develop the lists of attributes. The first attribute — source document — is a list of documents to be searched. The user can limit the search to one document or he can search all documents for relevant information. The second attribute — facility type — corresponds to the sections of the Safeguards Criteria. Usually, an inspector is interested in the requirements for a particular type of facility, e.g. a light water reactor (LWR). The inspector can limit his search to that type of facility, or he can call up all types of facilities to compare the different requirements for them. The third attribute — verification (type) — corresponds to the major verification activities (e.g. PIV, transfer verification) in each section of the Safeguards Criteria. Each verification activity is identified by a set of standardized paragraph numbers in the relevant section. For instance, the first set of paragraphs describes a comparison of records and reports; the second set describes PIV activities and the third set describes transfer verification activities. In SEAS, words are more meaningful than paragraph numbers. The fourth attribute — stratum (type of nuclear material) — is the main stratum code that describes the relevant material in the
FIG. 4. A condensed view of the entity–relationship diagram of SEAS (@ denotes the primary key).

Safeguards Criteria and is used by the SEE Section in evaluation procedures. The final search attribute — goal attainment (level) — qualifies whether the information relates to goal attainment or not. For example, a user may request information from the Safeguards Criteria regarding the requirements on verification of fresh fuel containing unirradiated direct-use nuclear material (FH)\(^1\) during a PIV at an LWR (see Figs 1–3).

Besides the search criteria described above, there is a list of keywords that will help a user in his search. Many of the keywords can be used to retrieve the definitions given in Annex A of the Safeguards Criteria. Other keywords can be used to find the sections related to the individual keywords.

### 2.3. Data model

Since the SEAS data are stored as structured query language (SQL) tables and since their implementation is procedure oriented, we choose Codd’s relational model

\(^1\) FH denotes fresh high fuel and refers to a stratum containing fresh fuel assemblies with direct-use nuclear material.
and outline the interactions with the aid of Chen’s entity-relationship diagram [4]. The database of SEAS consists mainly of these data structures implemented as SQL tables (see Fig. 4); identical data element or field names are used to relate the information stored in different SQL tables. The Cartesian product of the attribute domains, together with the keywords, forms a set of sixtuples (Source, Facility Type, VerifAct, Stratum, Goal_Attainment, Keyword). The set of sixtuples provides the means for retrieving the document contents, the verification methods, represented as ‘formulas’ (e.g. \( I + A + H\langle RH\rangle + F\langle RH\rangle \)), and the recommended instruments that will satisfy the verification requirements. The constraints imposed on the assigned attributes of the documents further partition the set of sixtuples into two disjoint subsets: (Source, Facility Type, VerifAct, Stratum, Goal_Attainment, —) and (Source, —, —, —, —, Keyword).² Query formulation is checked, in order to minimize the incidence of retrieval of an empty resultant set.

A SEAS document uses SEAS_DOC_REF to implement ‘hyper-links’ to its referenced documents. A referenced document can either be a part of this document or it can be another SEAS document; multilevels of references are possible.

3. SEAS SOFTWARE

3.1. Development tool selection

Two important events have contributed to the development of SEAS: the introduction of LAN into the safeguards computing environment and the use of Microsoft Windows 3.1³ as the standard user interface. Installation of LANs for safeguards provides a distributive data processing platform which makes it possible to develop a basis for client/server application. The popularity of Windows 3.1 has standardized a highly attractive, intuitive and easy to use graphical user interface (GUI), and has made available many Windows based development tools. In addition, SEAS is also intended for use in the field as an on-line reference tool — an alternative to (or perhaps a replacement of) the hard copy Safeguards Criteria documents. These factors have made the selection of development tools more critical.

3.2. Design issues

The productivity, performance and maintainability issues of the SEAS software should be emphasized. The selected tool must support easy access of data stored in the SQL server and of data stored locally without the necessity of changing the

² The symbol — indicates that the positional value cannot be given.
³ Microsoft Windows 3.1, Visual Basic 1.0, Microsoft Access, Visual Basic 3.0 and ODBC are trademarks of the Microsoft Corporation.
implementation code. The first version of SEAS was written in Visual Basic 1.0. This required rewriting of the database retrievals if SEAS was moved to a laptop computer. (This problem has been resolved in Visual Basic 3.0 by the use of ODBC.) Later, Microsoft Access was used as a prototype of an edition for in-field use. However, Access performs poorly when data reside in the SQL server.

The difficulty in maintaining the original version of SEAS (which was essentially a prototype) warranted a new design of SEAS. This new design of SEAS has enhanced its functionality. One of the various enhancements is the ability to associate more than one keyword with a SEAS document. Other enhancements include the integration of two components, SEASUTIL and SEASMOD\(^4\), the ability to delete a document reference, a print facility and a system development documentation.

Altogether, there are three editions of the new SEAS: a read–write edition for staff of the SEE Section who maintain the SEAS data, a read-only edition whose data are based on the SQL server, and a read-only edition for use in the field.

### 3.3. User interface

The application of SEAS is based on Windows. Its GUI is intuitive and easy to learn. A user requests SEAS functions by clicking/selecting on lists, text boxes, buttons and menus. As a Windows application, an on-line hypertext help feature specific to SEAS application is available. SEAS responds either by presenting the user's requested information or by issuing feedback messages. For example, after a user has obtained access to the SQL server, SEAS presents several lists of search criteria, such as source, facility type, verification activity, stratum, keyword and goal attainment. A query is formulated by using these lists. Since not all combinations of choices are appropriate, SEAS provides guidance through feedback messages.

The parameters that are chosen for the document search are implicitly connected by the logical **AND** operator, i.e.

\[
\text{Source} = \text{CRITERIA AND Verification} = \text{OIC (other inventory change)}.\]

In this example, 11 records containing the corresponding criteria for the 11 types of facilities specified in the Safeguards Criteria are retrieved. Interested readers can find a more complete description of the Windows based interface in the SEAS User’s Guide [5].

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\(^4\) SEASUTIL is the utility that maintains search lists; SEASMOD is the main program of the early prototype.
4. CONCLUSION

Pre-release versions of SEAS prototypes have been received favourably. The read–write edition of the new design of SEAS was delivered to the SEE Section at the end of 1993. SEE staff is updating the contents of documents, inserting new documents and generating references. As soon as the SEAS data are completed, SEE staff will release the read-only editions to safeguards staff and will conduct training.

Several lessons have been learned in the course of the development of the SEAS project. Both the SEE Section and the IDD Section are satisfied with the product. Future plans may include a more intelligent user interface. A drawback is that the SEE staff must devote some effort to preparing the presentation of the contents of documents. The word processing format of the original document is not preserved when the content of a document is transformed to a SEAS data record. If reuse of data without reformatting is a concern, then another method or tool which does not distort the original format should be considered for future implementation.

In addition, staff in the Department of Safeguards are trying to convert printed documentation to electronic reference documents. The Section for Equipment Management is converting instrumentation manuals, and the Section for Standardization, in co-operation with the Division of Information Treatment, is converting other Safeguards Manuals (e.g. SMO 7.1). Deciding on and requiring common user access to these documents, as well as providing guidance for their use and establishing all the links between them, are important and challenging tasks.

REFERENCES


1. INTRODUCTION

All chemical elements that are not monoisotopic may show small variations in their isotopic compositions. These variations are usually very small because they mostly arise from the natural fractionation of the isotopes. Processes leading to isotopic fractionation in natural materials are based on small differences in chemical reaction kinetics or transport rates, both of which arise from the mass differences of the species (isotopes) involved in these mechanisms. Differences in the isotopic composition of chemical elements may be related to the origin of the material. Such differences have been observed for a number of elements. The larger the relative mass difference of the isotopes, the larger is the fractionation effect; consequently, the lighter elements show more distinct differences in their isotopic composition than the heavier ones. In natural uranium the variations in isotopic composition are expected to be small and, hence, high precision, high accuracy isotopic measurements are required to identify significant differences in materials of different origins.

During IAEA inspections in Iraq, carried out under the United Nations Security Council (UNSC) Resolution 687, a number of natural uranium samples were
collected. The samples were classified according to their visual appearance. They consisted of rather pure products (uranium oxide powder), intermediate products (yellow cake), filter powders of poor purity and slurry samples. The sample size varied between a few milligrams and several grams, according to the amount of material available and the sampling procedure.

The Institute for Reference Materials and Measurements (IRMM) was requested by the IAEA to perform highly accurate isotopic measurements on 24 of these samples to identify samples of common origin and to distinguish between samples of different origin. This information was required by the IAEA’s nuclear inspection teams in Iraq.

2. MEASUREMENT TECHNIQUES

2.1. Measurement problems

In most text books and tables, natural uranium is described as having an isotopic composition of 0.0055 mole% $^{234}$U, 0.72 mole% $^{235}$U and 99.2745 mole% $^{238}$U [1]. Consequently, the molar isotope abundance ratios (for simplicity henceforth called ‘ratios’) to be measured by mass spectrometry are:

\[
(n \, ^{234}\text{U})/(n \, ^{238}\text{U}) \approx 55 \times 10^{-6}
\]

\[
(n \, ^{235}\text{U})/(n \, ^{238}\text{U}) \approx 7.2 \times 10^{-3}
\]

The measurement problems can be formulated as follows: accurate determination of the isotopic ratios of the individual samples, detection of (potential) small variations in these ratios, and quantification of these differences.

Since the differences were expected to be very small, the experimental set-up was designed to minimize the risk of sample (cross-) contamination during chemical preparation. The measurement method and instrumentation were selected for maximum reliability of the results.

2.2. Instrumentation

The $^{235}$U/$^{238}$U ratio was measured by gas source mass spectrometry using a Varian MAT 511 instrument equipped with a permanent magnet and two Faraday cups adjusted to the mass positions 330 and 333 ($^{235}$UF$_5^+$ and $^{238}$UF$_5^+$, respectively). A Finnigan MAT 262 RPQ was used for measuring the isotopes of minor abundance. This instrument is equipped with a retarding potential quadrupole filter for enhanced abundance sensitivity and with ion counting for precise measurement of the ion beam intensities of the isotopes of minor abundance.
2.3. Sample treatment

Upon arrival in the laboratory the samples were divided into two subsamples, say G and T. Sample G was used for measurement in the gas source mass spectrometer, after conversion to UF₆ and purification (distillation and degassing). A minimum amount of 2 mmol uranium (500 mg) is required to obtain sufficient UF₆ for the actual measurement. According to earlier studies [2], this process does not introduce additional (systematic) errors. The measurements were performed using the double standard method of the IRMM, i.e. the measurements were calibrated against two isotopic reference materials.

Subsample T was used for thermionic measurement. This was performed after chemical purification, calcination and dissolution. Figure 1 shows schematically the

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FIG. 2. Ratio of $^{235}\text{U}/^{238}\text{U}$ in natural uranium samples of different origin.
FIG. 3. Ratio of $^{234}\text{U}/^{235}\text{U}$ in natural uranium samples of different origin.
FIG. 4. Ratio of $^{236}U/^{235}U$ in natural uranium samples of different origin.
sample treatment. A minimum of 20 mg U was required for this procedure, allowing two independent chemical treatments on each sample and thus minimizing the risk of false results by cross-contamination.

2.4. Data treatment

The absolute $^{235}$U/$^{238}$U ratio as obtained by gas source mass spectrometry was used as internal standard for the thermionic measurements. Hence, these results were individually corrected for known systematic errors (e.g. isotopic fractionation). For calculation of the isotopic composition, the results of the gas mass spectrometric measurements and the thermionic measurements were combined. The uncertainties of the final results were calculated by combining the repeatability of the sample measurement, the repeatability of the measurement on the reference material and the uncertainty of the reference material used. All uncertainties were calculated on a 2 SD basis.

3. EXPERIMENTAL RESULTS

The results of the $^{235}$U/$^{238}$U ratio determination by gas source mass spectrometry are shown in Fig. 2. In two cases there was not enough material to perform a conversion to UF$_6$. Consequently, the $^{235}$U/$^{238}$U ratio measurement had to be done by thermal ionization mass spectrometry, which introduces a larger uncertainty ($<0.1\%$) than is usually obtained with UF$_6$ measurements ($\leq 0.04\%$).

The $^{234}$U/$^{235}$U ratios were measured by ion counting. Additionally, the absence of $^{236}$U was checked by measuring the $^{236}$U/$^{235}$U ratio. In some cases, traces of $^{236}$U — significantly higher than background — could be detected. The major error component in these cases was the counting statistics; this was due to the low count rate. However, accuracies of 0.4–0.6\% ($^{234}$U/$^{235}$U) and 2–5\% ($^{236}$U/$^{235}$U) were achieved. Figures 3 and 4 show the results of these measurements. The presence of $^{236}$U in samples A to C seems to originate from cross-contamination of the natural uranium source material in the conversion facility. This facility was known to process also low enriched uranium in cascades that recycled uranium from spent fuel. Sample P was taken from process residues which may contain various source materials.

4. DISCUSSION

From the results of the $^{235}$U/$^{238}$U ratio measurements no conclusions regarding common or different origin of samples can be drawn because there is no significant difference between the individual results, even though the accuracy is better
This is rather surprising, because the natural uranium reference materials that are commercially available show a much larger spread. The $^{234}\text{U}/^{235}\text{U}$ ratios indicate clearly that several samples may be grouped together. Samples A to C are apparently of the same origin (group I) and are different from the rest of the series. Samples D to P (group II) show higher values; however, they form a rather homogeneous group, with the relative SD being only 0.8%, while the first group shows a relative SD of 0.06%. The difference between the two averages is 2.2%. This significant difference between the samples from groups I and II was an important finding, which was used in the verification of the correctness of a source material declaration.

Regarding samples Q and R, the question was whether these are of the same origin as samples A to C. The $^{234}\text{U}/^{235}\text{U}$ results show that this is not the case and indicate that samples Q and R correspond to the samples of group II. Samples S to X were taken from a population of uranium oxide items which, according to their physical appearance, could be divided into three subgroups: samples S and T and samples W and X are visibly different, but samples U and V appear to be a mixture of the two above mentioned subgroups. The $^{234}\text{U}/^{235}\text{U}$ ratios show that samples S and T belong to group II, whereas samples W and X do not belong to this group.

When we compared the spreads observed in the $^{235}\text{U}/^{238}\text{U}$ ratios of the samples being investigated with the results for natural uranium reference materials obtained earlier [3], we observed a surprising 'homogeneity' in the samples collected in Iraq. Furthermore, the $^{234}\text{U}/^{235}\text{U}$ ratios measured for the natural uranium reference materials showed a much larger spread than the $^{234}\text{U}/^{235}\text{U}$ ratios of the samples from Iraq. Consequently, we had to draw the conclusion that the samples under investigation originate from only a few sources, whereas the commercially available natural uranium reference materials represent a much wider variety of origin.

5. CONCLUSION

The measurement results obtained at the IRMM for natural uranium samples collected by IAEA inspectors in Iraq under the UNSC Resolution 687 had a direct influence on the further steps to be taken by the IAEA.

For the experimental techniques, especially the measurement of isotopes of minor abundance, there is still a potential for improvement, i.e. a reduction of the uncertainty. Investigations in this field are being performed at the IRMM, and more information on natural uranium samples from various origins is being collected.

Measurement procedures for ultra-accurate determination of isotope ratios (uncertainty of $\sim 10^{-5}$) in gaseous compounds have been developed. They have been applied for redetermination of the Avogadro constant. At present, they are being tested for applicability to UF$_6$ measurements.
REFERENCES


IAEA-SM-333/236P

AUTOMATIC CONTROL SYSTEM FOR THE DETECTION OF UNAUTHORIZED TRAFFIC OF NUCLEAR MATERIALS ON THE BORDER CHECKPOINTS OF THE REPUBLIC OF BELARUS

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Byelorussian–Polish Joint Venture ‘Polimaster’, Minsk, Belarus

1. INTRODUCTION

Promoting the policy of non-proliferation of nuclear weapons, automatic control systems to detect unauthorized traffic of radioactive materials and special nuclear materials (SNM) have been installed at the border checkpoints of Belarus since mid-1993. Within the bounds of the national export control system it is intended to supply all the border checkpoints of Belarus (covering roads, railways, air terminals and pedestrian crossing points) with this equipment. Six months of experience and the results of the checks carried out by four such systems on the border between Belarus and Poland have proved that it is necessary to supplement legal norms for export activity with means of technical control. This problem is typical for all countries which have nuclear facilities on their territories. Side by side with the problem of non-proliferation of nuclear weapons the systems described solve a number of ecological control problems related to the export of polluted goods connected with the Chernobyl accident. Also, these systems prevent unauthorized import of wastes containing radioactive materials.

The basic element of the automatic control system is a vehicle portal monitor (VPM). Existing VPM systems at nuclear power stations have low sensitivity.

The automatic VPM for vehicle border checkpoints is described below. It is totally adopted to the special demands for controlling radioactive materials and SNM traffic.
2. APPLICATIONS

The monitor is especially intended for application at the entrances to nuclear facilities and at border checkpoints. It can also be used at weapons manufacturing plants and waste storage sites. The unit is microprocessor controlled and should it become contaminated, a warning signal is generated. Also, if the detector becomes defective, the malfunction is signalled. An autotest is performed every hour.

3. OPERATION

The monitor is installed near the road and, depending on local conditions, up to four columns can be installed. A detector is installed beneath the road surface. If the threshold is exceeded, an alarm is set off, with both audible and visual signals.

4. OPERATING PRINCIPLE

A NaI(Tl) probe is used for measurements of ambient gamma background, including possible external contamination, as well as for detecting any abnormal increase of gamma radiation from radioactive materials passing the monitor. The autotest at each full hour executes the following checklist: power status, probe status, correct operation, external contamination indication. The results are stored in the unit memory and can be printed out. The detector ensures that during the background measurement there is no obstacle inside the measurement field. The unit parameters can be set using any notebook PC.

5. SYSTEM COMPONENTS

The monitor consists of:

- scintillation probe NaI(Tl), $\phi = 2.5 \text{ in} \times h = 6.5 \text{ in}$
- processing unit
- alarm unit
- power supply unit
- road column
- detector
- printer (optional)
- notebook PC (optional)
- standard program diskette.
6. SPECIFICATIONS

*Mechanical:*
- column: $\phi = 125 \text{ mm}$, $h = 1750 \text{ mm}$
  weight = 60 kg

*Electrical:*
- power: 220 V, 50 Hz
- backup batteries: 12 V, 36 A·h
- backup duration: 48 h at $-20^\circ\text{C}$, 175 h at $+20^\circ\text{C}$

*Climatic:*
- temperature: $-30^\circ\text{C}$ to $+50^\circ\text{C}$
- relative humidity: 85% at $+50^\circ\text{C}$

7. SENSITIVITY
(for unshielded radioactive sources) at 2 m distance from the detector:

<table>
<thead>
<tr>
<th>Vehicle speed (v) (km/h)</th>
<th>Radionuclide</th>
<th>Threshold activity (\mu\text{Ci})^a</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>$^{60}\text{Co}$</td>
<td>4.5</td>
</tr>
<tr>
<td>(static sensitivity)</td>
<td>$^{137}\text{Cs}$</td>
<td>6.2</td>
</tr>
<tr>
<td>7.5</td>
<td>$^{60}\text{Co}$</td>
<td>9.8</td>
</tr>
<tr>
<td></td>
<td>$^{137}\text{Cs}$</td>
<td>13.5</td>
</tr>
<tr>
<td>30</td>
<td>$^{60}\text{Co}$</td>
<td>21.4</td>
</tr>
<tr>
<td></td>
<td>$^{137}\text{Cs}$</td>
<td>29.4</td>
</tr>
</tbody>
</table>

Static sensitivity for other radionuclides:

- $^{241}\text{Am}$: 614$^b$
- $^{239}\text{Pu}$: 99.1$^b$
- $^{235}\text{U (36\%)}$: 106.2$^c$
- $^{235}\text{U (93\%)}$: 47.2$^d$

---

\(^a\) 1 Ci = $3.7 \times 10^{10}$ Bq.

\(^b\) Point source.

\(^c\) Reactor fuel element WWER-SM type in horizontal position.

\(^d\) Reactor fuel element MR-6 type in horizontal position.
1. INTRODUCTION

We describe briefly the process whereby a comprehensive, performance based training programme was created for the IAEA Department of Safeguards to prepare inspectors for their routine safeguards related activities at CANDU facilities worldwide. This was the first use of such a methodology by the IAEA as part of a major training programme, and the approach was designed and documented to serve as a model for future training requirements. The process was divided into five discrete phases — analysis, design, development, implementation and evaluation — which are described in the following section.

2. PROJECT PHASES

2.1. Analysis

The preliminary task was to determine the requirements of the end-users of the training programme by performing a needs analysis. This was conducted via questionnaires and interviews with subject matter experts (SMEs) from various sections of the Department of Safeguards. A report on the preliminary findings, including a draft training course proposal with terminal and enabling learning objectives relating to the subject matter, was then prepared and distributed to the same individuals for comment. This was done as part of the analysis to accelerate the process by combining portions of the analysis and design phases, and receiving, in essence, a simultaneous feedback. The same approach is recommended for future endeavours of this type.

The results clearly indicated that an expanded scope of training was required and that the limited focus of the preliminary findings proved to be inadequate. Nevertheless, one proposal that was endorsed by all participants was the concept of a modular approach, whereby specific sessions of the training material could be prepared and implemented on a priority basis and later integrated into the overall
package. This was felt to be particularly appropriate in the current dynamic situation when major upgrades in safeguards equipment were already under way.

In addition to the needs analysis, a task analysis was also performed by reviewing the various inspection related activities and, again, by interviewing SMEs. Activities were subdivided until the appropriate enabling learning objectives were determined, and a report showing this breakdown in the chronological order of a typical safeguards inspection was prepared. A task pyramid was also created to illustrate the hierarchy of cognitive and skills requirements and their interrelationships during routine inspection work. Both of these documents were then distributed to project staff for ultimate feedback on the analysis phase.

2.2. Design

With the groundwork complete and the end-users well informed about, and in agreement with, the training approach, more specific design work could proceed. The training package was envisaged to comprise a number of sessions addressing specific topics and, after numerous proposals, it was decided that the training manual would have two primary documents for each session, namely:

— A lesson plan containing:
   (a) The session objective,
   (b) Action steps on how to proceed,
   (c) A self-check segment to identify the major areas of interest,
   (d) A list of study resources, and
   (e) The actual study material.

— A control sheet (see Fig. 1) containing:
   (a) The session objective,
   (b) The trainer interface to summarize the instructional requirements, and
   (c) The performance criteria to be demonstrated by the trainee as the final prerequisite to formal certification.

It was felt that these complementary documents would provide the trainee with all the necessary guidance to use the available study material in the most efficient way possible and at the same time to maintain a record of the training progress as part of the certification scheme. Both of these aspects were deemed to be essential in establishing a training programme incorporating both self-paced and conventional instruction. Of course, it was understood that the success of the approach would also depend on the creation and upkeep of a library containing current versions of all the study resources given in the training plan, including documents, video tapes, interactive computer programs and equipment mock-ups.
Session Objective:
Given a MUX CCTV System and a procedure, be able to service and operate the system in order to establish surveillance and prepare a tape for safeguards review.

<table>
<thead>
<tr>
<th>TASK</th>
<th>TRAINER INTERFACE</th>
<th>PERFORMANCE CRITERIA</th>
<th>DATE</th>
<th>INITIALS</th>
</tr>
</thead>
<tbody>
<tr>
<td>9.1</td>
<td>Oversee routine inspection activities as they are performed by the Trainee on a mockup MUX recording subsystem.</td>
<td>Using a mockup MUX recording subsystem, stop surveillance on one channel, replace the tape and re-establish surveillance.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>9.2</td>
<td>Oversee routine inspection activities as they are performed by the Trainee on a mockup MUX download subsystem.</td>
<td>Using a mockup MUX download subsystem, perform a scene count and download a sample MUX tape provided by the Trainer.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>9.3</td>
<td>Oversee routine inspection activities as they are performed by the Trainee on a MUX CCTV System during a safeguards inspection at a CANDU facility.</td>
<td>Service and operate a MUX CCTV System during an actual safeguards inspection at a CANDU facility.</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

NAME OF TRAINEE

FIG. 1. Control sheet for The CANDU Course.

Lesson plans were designed primarily to guide the trainee toward the appropriate study resources and to provide any necessary study material in order to realize the stated session objective. Use of control sheets was intended to be the mechanism whereby a standardized approach could be established and maintained by clearly defining the scope and depth of specified training tasks for both the trainer and the trainee. In this way a consistent end-product could be ensured, regardless of the personnel involved in the process; in other words, it would no longer be necessary to have a resident expert in performance based training to maintain the momentum of the programme for future iterations. This would allow the IAEA to proceed with
training of inspectors for activities at CANDU facilities without the need for external assistance. These control sheets were also designed to serve as a record of achievement in the certification process.

To appreciate the concept of certification as it is applied here, one must understand that it does not encompass the overall expertise and abilities embodied in the work of the inspector but that it focuses only on those topics that are defined by the stated objectives. In other words, certification is strictly limited to clearly defined subject areas, such as the servicing and operating of specific pieces of safeguards equipment, and is in no way intended (or able) to evaluate a higher order of inspection skills or knowledge.

Another point of interest regarding the certification process is the fact that full qualification for all course subjects cannot be achieved during the limited period of designated instruction in Vienna. Skills and/or knowledge in certain subjects cannot be adequately acquired without actually performing tasks in the true working environment or, in some instances, without the use of full-scope simulators. This particular problem was solved by introducing the notion that some of the performance criteria would be designated for completion in the field under normal working conditions while the trainee is under the supervision of a senior staff member.

2.3. Development

Once the design of the programme had been formulated, the next step was to enlist the appropriate SMEs who would provide the requisite technical input for the preparation of training material for each session. A set of guidelines was issued to each SME to ensure that a standardized approach would be used for development of the individual sessions and to simplify the job of assembling training material.

Each session was to incorporate:

(a) A clearly defined session objective to establish the scope and depth of training;
(b) Suggested action steps to instruct the trainee on how to proceed;
(c) Performance criteria to define the intended outcome of the training, i.e. what the trainee should be able to do as a result of the training;
(d) Single-source study resources to be used as reference material, i.e. the use of existing material for training unless updating or further clarification or expansion of a topic is required; and
(e) Specific reference, wherever possible, to the exact location of pertinent material in the study resources.

Emphasis was placed on the concept of using existing documentation as much as practicable rather than creating redundant material. Also, it was stressed that references should be as specific and precise as possible in order to guide the trainee directly to the intended source of information.
With these points in mind, lesson plans and control sheets were prepared and forwarded, on an iterative basis, to the SMEs for technical evaluation and modifications. In parallel with these activities, interactive computer programs and video tapes were produced by the Canadian Safeguards Support Programme (CSSP) for the purpose of orientation on facility types and safety procedures, and a room with mock-up safeguards equipment and computer terminals, also provided by the CSSP, was set up as a study centre for CANDU related safeguards installations.

2.4. Implementation

Implementation of The CANDU Course for groups of 12 trainees has now been carried out on three separate occasions. The initial course was held in 1991 and consisted of two weeks of instruction — the first week in a classroom environment in Vienna and the second week at two facilities in Canada. Because of the high cost of the field exercise, as well as the fact that new inspectors are closely supervised on their initial visits to a facility, it was felt that, by placing the emphasis on the practical application of safeguards activities through interactive training, the field exercise could be discontinued.

Each of the succeeding courses was held at the IAEA Headquarters in Vienna for a period of one week only. With the emphasis on hands-on activities, it was frequently necessary to subdivide the trainees into small working groups in order to accommodate the available equipment and instructors. This resulted in a situation in which trainees were occasionally scheduled for periods of free time or for activities different from those of their colleagues — a type of staggered approach. However, the duration of practical training tends to be a function of the overall experience of each group and can vary dramatically; therefore, unless the entire target population is at approximately the same level of proficiency, some flexibility must be factored into the scheduling process.

Certification of trainees was implemented on a trial basis for the first time as part of a major training course at the IAEA. Headquarters related performance criteria were completed voluntarily, the major difficulty being insufficient time for some group training sessions.

2.5. Evaluation

Formal evaluation of the course was carried out by the trainees via evaluation sheets presenting a number of specific questions regarding individual sessions, overall course content and administration. Of particular interest were the answers associated with sessions incorporating new approaches and technologies, such as a computer based, multimedia presentation on the operation of CANDU facilities that was introduced in 1992. Formal and informal feedback from all participants is used to improve the process during the next cycle.
3. CONCLUSIONS

This performance based instructional approach offers a high degree of flexibility within the structured format of the IAEA safeguards training framework and has proven to be a very successful addition to the curriculum of the Section for Safeguards Training. Also, it is felt that, by demonstrating the feasibility of trainee certification through the use of control sheets, a positive step has been taken in response to the ever increasing demands for accountability and consistency, and that this will serve both the IAEA and the CSSP well in future training endeavours.

ACKNOWLEDGEMENTS

In general terms, any type of training is only as good as the subject matter experts who are available to disseminate the information. We have been fortunate in this project in dealing with highly competent and motivated personnel at the IAEA, whose combined efforts have helped to establish the current training programme. Special thanks are also due to O.G. Bates and J. Vidaurre-Henry of the Section for Safeguards Training for their support and advice, particularly during the design and development activities.

This work was sponsored by the CSSP of the Atomic Energy Control Board of Canada, in affiliation with the Section for Safeguards Training of the IAEA. The training programme and methodology described in this paper were developed by T. Ellacott while he was under contract as a CFE to the IAEA in the Section for Safeguards Training.

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A COMPUTER BASED, INTERACTIVE, MULTIMEDIA TRAINING MODULE FOR SELF-PACED LEARNING

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1. INTRODUCTION

The Canadian Safeguards Support Programme (CSSP) of the Atomic Energy Control Board (AECB) has been involved in a number of training presentations for
IAEA inspectors, including seminars on on-load reactor (OLR) safeguards, and courses on the operation and maintenance of equipment developed by the CSSP, such as the Cerenkov viewing device, bundle counters and closed circuit television (CCTV) systems. Training materials have historically included lecture notes, copies of transparencies and slides, workbooks, reference manuals, etc., and implementation has incorporated classroom lectures, in-field training, equipment demonstrations and hands-on sessions. Although many of these features are still considered to be essential components of a successful training programme, certain disadvantages of the traditional approach are obvious. Recent improvements in computer technology have created potential alternatives to overcome the disadvantages of the traditional training approach.

2. DISADVANTAGES OF TRADITIONAL TRAINING METHODS

Traditional training is dependent on the simultaneous availability of instructors, trainees and training materials, as well as the setting in which the presentations can take place, such as classrooms, laboratories and nuclear facilities. This necessitates long term advance scheduling for co-ordination of these fundamental prerequisites. The preparation time for a first-time training session is usually orders of magnitude greater than that required for a session which has already been implemented and evaluated on previous occasions; nevertheless, each time a new instructor is assigned to the task, the preparation time can increase dramatically. This is a particular problem faced by the IAEA owing to turnover of staff.

Other significant factors that further complicate the equation are the instructor’s previous overall experience in the field of training, his general suitability as a trainer and his technical expertise on the subject matter. In fact, the search for an alternative training approach was driven, to some extent, by ongoing difficulties in recruiting IAEA staff having these fundamental instructional attributes plus the available time to perform the task.

In a worst-case scenario, adherence to the traditional training approach may impose an extensive waiting period prior to the delivery of the requisite scheduled instruction. Such a delay can have serious negative repercussions both for new safeguards inspectors and for experienced ones who have been assigned new responsibilities or who have identified a personal need for refresher training in a specific subject area. Fortunately, in many cases this problem can be solved through the availability of stand-alone, computer based training modules.

Finally, the traditional classroom lecture approach, although still commonly used, is known to be one of the least effective methods for assimilation of information by the trainee.
3. DEVELOPMENT OF A COMPUTER BASED TRAINING MODULE

As part of a comprehensive course on CANDU reactor safeguards for IAEA inspectors, the CSSP developed a computer based, interactive, multimedia, self-paced training program, called the CANDU power reactor training module (CPRTM), which addresses the general physical layout and operation of CANDU power reactor facilities. This decision was influenced, to a considerable extent, by the positive reviews expressed for a much less sophisticated program, called the bundle counter interactive training module (BCITM), which focuses specifically on the spent fuel bundle counter [1]. The BCITM had been introduced in the IAEA several years earlier, and since its creation the techniques for multimedia presentations using computers have greatly improved, making it now possible to include colour photographs, video, animations and sound at a reasonable cost.

The CSSP assembled source material, which included an outline of the course, diagrams, slides, charts, photographs, video tapes, etc., to be used by a company specializing in the production of multimedia programming to enable construction of the CPRTM. At the time the project was initiated, the Macintosh computer platform was the only one able to provide multimedia capability at the cost specified by the CSSP. The developers, through an iterative process with the CSSP, created the final product: a multimedia, interactive, computer based training module on the operation of CANDU reactors, with major emphasis placed on the flow of nuclear fuel through the facility. Numerous reviews by subject matter experts were conducted to confirm the technical accuracy of the material before it was used as a training resource.

The final cost to develop the CPRTM was kept at approximately Can $60,000 by providing the contractor with all visual resources and technical assistance. Fortunately, no new visual source material had to be produced in addition to what the AECB and the utilities already had. For graphical representations and animations, and because of the precise wording for text boxes and narration clips that were used in the program, a higher degree of communication between the contractor and the CSSP than was originally anticipated was required to ensure that a suitable product was emerging in the most efficient manner.

4. OPERATION OF THE CPRTM

The CPRTM is designed to be accessed in two distinct ways, by selecting either the 'Guided Tour' mode or the 'Reference' mode. All selections are made via the computer mouse, thereby eliminating the need to remember keyboard command sequences, and manipulation of the various program components is largely self-evident. Nevertheless, a 'Help' section, which explains all the navigational options, is also provided. Figure 1 presents a hard copy of the 'Help' screen which is accessed by clicking with the mouse on the 'Question Mark' button located among the four
FIG. 1. Help screen.

FIG. 2. Fuel handling system overview screen for a single-unit station.
buttons in the upper right corner of the display. Figure 2 presents a hard copy of
the fuel handling system overview screen for the single-unit station. The mouse is
used to activate the videos and animations associated with the four specific segments
of the fuelling process shown here. Unfortunately, the instructive power of the
animations, video and narration cannot be demonstrated in a paper. They are,
however, very effective training tools when properly utilized.

It is recommended that first-time users, or individuals who are not already
familiar with CANDU operations and station layouts, take the ‘Guided Tour’ option
to advance through all the material in a predefined sequence and at a pace determined
by the user. It is felt that this approach creates the highest probability for assimilation
of the information by the trainee, by ensuring that it is initially experienced in the
most logical order and that no portions are inadvertently omitted while, at the same
time, the complete range of interactive capabilities for each screen are provided. As
an alternative approach, the ‘Reference’ mode is available for use as a menu driven
refresher session with random access to all elements of the module.

5. ADVANTAGES OF COMPUTER BASED TRAINING

The ability to combine various media on a computer makes it possible to con­
vey the information more quickly and realistically than via traditional training
methods; this tends to shorten the duration of training while increasing the probabil­
ity that the trainee will develop a more accurate mental model and a better overall
understanding of the process. Maximum information transfer can be achieved by
including a visual component, since most people find it easier to learn by seeing
rather than by hearing. Also, by selectively combining media on a given topic it is
often possible to preclude ambiguities that are sometimes difficult to avoid when only
a single medium is used. This technique is particularly effective in avoiding mis­
understandings in situations where the language of instruction is different from the
mother tongue of the trainee.

Self-paced, computerized instruction has several other obvious advantages
over conventional training. Although the initial preparation time is usually longer
than that required for a typical classroom lecture, the end result is a standardized
product for all trainees. Further course preparation is not required, discounting revi­sions, which will always be necessary in today’s rapidly changing technological
environment, and the delivery time of the instructor can be as small as zero in the
case of a totally self-paced delivery platform. The user is able to schedule this type
of training for any convenient time period and to cover the material at a self-
determined pace in a single session or in numerous mini-sessions, which can be taken
sequentially, without overlap, by using the convenient ‘Bookmark’ feature, which
maintains a personalized record of the training progress and establishes the appro­pri­ate starting point to complete the module. Future refresher training is essentially a
repeat of this process on a random access basis. With upcoming improvements in computer technology and data compression, the next generation of computer based training modules may further simplify the process by enabling installation of the entire instructional package in the trainee’s personal computer, or on a laptop computer, thereby individualizing the process to the level of a one-on-one tutorial which would be permanently available on demand, possibly in a portable format for in-field reference.

6. CONCLUSIONS

The CPRTM has been very well received by all users, not only by those from the target group but also by training experts and other interested parties who have been invited to evaluate the product. In fact, the only aspect of the module that has received less than rave reviews is the time lag associated with data transfer from hard disk storage to RAM for certain segments of the program, particularly those incorporating extensive animations and video clips. Currently, the CPRTM occupies 55 Mb of computer storage space. It is anticipated that advances in data compression technology and improvements in data handling capability will alleviate this single concern in the near future.

It can already be shown that the CPRTM, as well as the previous generation BCITM offer significant advantages in the implementation of a highly effective and efficient training course. The three major benefits of this approach are:

(a) Reduced training costs,
(b) Improved assimilation of information, and
(c) Standardization of instruction.

Reduced training costs are realized through a decrease in time allocated for formal classroom lectures, with a corresponding decrease in costs for instructors and for administrative and technical support staff.

Assimilation of information is improved through the process of interactive instruction which allows the trainee to control the rate of progress. This factor, combined with the use of text, narration, diagrams, animations, photographs and video, helps the trainee whose normal language of communication is different from that used in the instruction. Experience has shown that very little remedial training is required after completion of training with the CPRTM.

Standardization of instruction is ensured by presenting identical material to every trainee, regardless of the time or location of training.

The CSSP is encouraged by the overwhelmingly positive response to this new training concept and is confident that future technological advances will soon allow computer based approaches to be used in an increasing number of training applications.
1. INTRODUCTION

Implementation of IAEA safeguards is based on nuclear material accountancy systems. Safeguards Agreements between Member States and the IAEA define the basic information to be included in records at facilities as well as reports issued by the Member States for submission to the IAEA. The present reporting system includes primarily information about nuclear material, quantities and description codes. Operational data are reported to a limited extent. Reports are primarily issued periodically, with a time delay of up to two months after the actual event. Although nuclear material accountancy in most countries is computerized, the periodic reporting is maintained. Electronic transmission of data from Member States to the IAEA has been introduced only to a limited extent.
In recent discussions on the future development of safeguards and the application of alternative safeguards regimes, additional information related to operational activities plays an important role. Timely transfer of accountancy data as well as operational data from a Member State to the IAEA is foreseen to be of significant importance. The development of telecommunications and computerized transfer of information in the last ten years have opened new possibilities to handle both traditional safeguards data and more extended reporting under an improved safeguards regime.

Correctly applied, the new technology ought to make it possible to reduce manual work, shorten transfer time of data between Member States and the IAEA, keep the IAEA database more up to date and develop new control mechanisms.

In order to investigate the prerequisites for electronic transmission of accounting reports and related information in accordance with a proposal from the IAEA, the Swedish Nuclear Power Inspectorate (SKI) initiated a research project in December 1991.

A report on the project was delivered on 19 March 1993. Representatives of the SKI and the IAEA met in Vienna in April 1993 and agreed on equipment to be supplied to the IAEA under the Swedish Support Programme. Furthermore, a plan was agreed upon how to test and establish electronic transmission of accounting reports between Stockholm and Vienna.

Communication tests were carried out during autumn 1993. In the same period an Alpha AXP computer from Digital Equipment was installed at the IAEA to handle the communication between Sweden and the IAEA computer system. The machine has capacity enough for modern electronic communication not only with Sweden but also with other Member States.

At SKI in Sweden the communication will be run on a stand-alone personal computer with the OS/2 2.1 operating system. The basic communications software is Pathworks/DECNET from Digital Equipment. The IAEA has developed and supplied to the SKI the necessary user software.

2. PRESENT REPORTING SYSTEM AND INFORMATION TREATMENT

When the amount of reports and entries increased, Sweden changed from reporting on hard copies to reporting on magnetic tapes in 1985 and that is still the way reports are transmitted to the IAEA. The format is fixed, with 80-column records. The tapes are sent by regular mail, whereby the number of days from dispatch to receipt varies from four to, in some cases, more than 30.

As stipulated in Para. 7 of INFCIRC/153 (Corrected) the State has to establish and maintain a State System of Accounting and Control (SSAC) of all nuclear material subject to safeguards under the Agreement. There are two levels of accountancy, one at the facility level and one at the State level.
According to the Swedish SSAC the facility has to report continuously all inventory changes and, at physical inventory taking, material balance reports and physical inventory listings to the State authority, the SKI. This information is today given both on hard copies and diskettes and entered into the central computerized register at the SKI. Changes in the accountancy system in Sweden are currently being made to allow for exchange of information between the facilities and the SKI on magnetic media and in the future through electronic transmission.

Reports required by the Agreement between the IAEA and Sweden are made at the SKI from information received from the facilities. The reports are sent to the IAEA on magnetic tapes via ordinary mail and entered into the IAEA safeguards information system, ISIS. On the tapes are inventory change reports, material balance reports, physical inventory listings and concise notes. The total number of reports is about 550 per year, comprising about 40,000 records. The number of tapes sent is between 15 and 20 per year. Sweden reports using a fixed format.

3. INFORMATION SYSTEM AT THE IAEA

The ISIS handles information from Member States and from the IAEA. The system contains a database management system and various subsystems and applications. It stores safeguards information and generates reports and other information for Member States, the Board of Governors and the IAEA.

The computer hardware for ISIS is built up on IBM, VAX and PC platforms. An IBM mainframe handles safeguards information, while applications such as word processing, office automation applications, graphic presentation and external communication are handled by other platforms.

A high degree of the ISIS integration has been achieved between the various platforms through the safeguards local area network (SG LAN) based on the IBM token ring LAN architecture and the Banyan/VINES network operating system (NOS). The SG LAN is used as the common platform for a unified information processing environment, using appropriate gateways and, as the common platform for integration of safeguards applications, office automation features and electronic mail throughout the Department of Safeguards.

The accounting reports from the Member States are processed and stored in the mainframe computer at the IAEA HQ in Vienna. Results and data from inspections are also processed and stored in the computer. Special software, the ISIS request processor, which runs on the mainframe, makes it possible for authorized users to generate various reports.
4. ELECTRONIC COMMUNICATION

In order to facilitate matters for the inspectors at the IAEA field office in Toronto, the IAEA has created a communication link between Vienna and Toronto. At the field office the inspectors have remote access to the ISIS request processor.

Following the experience gained during the development and two years of operation of the communication link between the IAEA HQ in Vienna and the IAEA regional office in Toronto, and following the IAEA's strategic plans for 'Enhancement of Safeguards Remote Data Handling Facilities', discussions with Sweden about electronic transmission of safeguards data were initiated in 1991.

The solution for the communication link between Sweden and the IAEA is a further development of the 'Toronto communication'. It has several attractive features. The remote user never logs into the IBM mainframe computer or even into the Alpha AXP machine, but simply deposits a set of files which are automatically taken care of by the IAEA computer system. The fact that no log-on into the IAEA computer system takes place is an important security factor. It also reduces the time during which communication actually takes place to a minimum and thus enables the IAEA computer system to be easily available for a maximum number of remote users.

An important argument for a hardware solution as powerful as the Alpha AXP has been that this machine will have capacity enough to open possibilities for other Member States to use the communication for exchange of information.

Bearing in mind that future transmission of advanced electronic information, for instance in the form of pictures, has been discussed and since such information could be very extensive, it is essential to make sure that adequate hardware capacity is available.

Logging of remote accesses can be carried out in a satisfactory way, which is good from the point of view of computer security. It is always a protection for a computer system, linked to public communication, if all accesses from outside as well as all attempts to access the computer which have failed are properly recorded and can be tracked down.

Several different methods of long distance electronic data transmission are available today. Such methods range from transfer over a normal telephone line with standard modems to communication via direct satellite links.

When deciding about hardware, operating system and communication protocols in connection with the electronic communication of safeguards data, it is essential that choices are made which ensure:

— reliable transmission of data
— compatibility over several computer platforms
— possibility for future development
— a wide international base for communication.
The X.25 service, which is offered by many national telephone companies, has been chosen for the communication between Sweden and the IAEA. Even if more sophisticated solutions may be available and/or developed shortly, X.25 meets the present requirements at reasonable costs.

For the future an interesting trend is the fast growing Internet with e-mail. Within e-mail a satisfactory security level for transmissions of safeguards data is currently not available. However, it may well be that e-mail can be used for transmission of messages which are important for the safeguards work and not critical from a security point of view.

When changing to transmitting accounting reports through electronic communication there is a need to build up administrative routines to assure that the reports are handled correctly. The State has to be sure that its reports reach the IAEA unchanged during the transmission and the IAEA has to be sure that the reports received are the ones sent from the State concerned. Today reports are sent together with a signed letter: how will this be handled in the future? On both sides, the State and the IAEA, routines for receiving and transmission of information have to be built up.

5. SAFETY ASPECTS

From today’s point of view the worst security threat to a system where accounting reports are transferred electronically is probably if the transmission is not reliable. In such a case a situation will arise where uncertainty about the true figures may give room for loss of material which cannot be traced quickly enough. The prime goal from the point of view of computer security must therefore be to ensure that transmissions are reliable in the sense that information transferred arrives at the receiver in an unchanged condition.

X.25 transmissions are good in this respect. The data are divided into ‘packages’ when transmitted. The size of the packages is determined by the quality of the line. If a package is not received unchanged the phenomena will be monitored and the transmission interrupted.

In connection with the remote PC-based integrated resource planning job request system, the IAEA has developed a security facility, which automatically compresses the data and encrypts them before they are sent on to the X.25 network. This facility will be used for the electronic transmission. Sufficient measures have thus been taken to ensure a reasonable level of security. This is obtained by protection against unauthorized access to the IAEA and the SKI data systems, through X.25 and by way of encryption of the information transferred.

The establishment of an electronic communication link between the SKI and the IAEA based on X.25 opens a range of possibilities to enhance remote data
handling within the safeguards field. In the future the communication may be developed for use not only for transfer and acknowledgement of accounting reports but also within the field of computerized inspection reports.

In a longer perspective, when methods of data compression improve and when higher transmission speeds become available for international communication, more advanced information, such as pictures, could be transferred in support of safeguards work.

On a national basis, connections may be established with the nuclear plants to enhance and speed up the information handling in connection with the accounting reports and the inspections.
INTEGRATED SAFEGUARDS SYSTEMS

(Session 8)

Chairman

T.A. SELLERS
Institute of Nuclear Materials Management
IAEA INTEGRATED SAFEGUARDS INSTRUMENTATION PROGRAMME (I²SIP)

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Abstract

IAEA INTEGRATED SAFEGUARDS INSTRUMENTATION PROGRAMME (I²SIP).

During the last two decades a substantial amount of safeguards instrumentation has been developed and implemented by the IAEA. For most safeguards verification applications there is now an instrument available for deployment. However, the safeguards instrumentation was developed through various Member States Support Programmes and commercial suppliers, and includes a disturbingly large variety of dissimilar equipment which often has overlapping functions. The net result is an additional burden on inspectors and support staff, and subsequent inefficient use of resources. In 1992, I²SIP was formally included in the Safeguards Departmental Plan for 1993–1997 as a systematic programme aimed at optimizing the instrumental support infrastructure and minimizing the equipment inventory with respect to the number of different instrument types. An important component of I²SIP is the definition and acceptance of a particular instrument standard that can help define the hardware and software of the future. The instrumental bus VXI has been selected as an appropriate international standard. A major new system, the Mark II bundle counter, is under development as the prototype instrumental system for proving the efficacy of the overall concept.

1. INTRODUCTION

On-site inspection supported by instrumentation has become the primary verification mode for international safeguards. Custom designed instruments for measurement, monitoring, sealing and containment have been developed as essential tools in support of inspectors. The quality and quantity of instruments has matured to the extent that, for most of the regularly encountered safeguards applications, an appropriate instrument is available for implementation.

Safeguards instrumentation has been developed to meet the individual safeguards application needs while conforming to the international environment. This means that, to a large extent, the application of instruments and the expected specific operational deployment were the prime considerations governing the design and development of instruments. Consequently, consideration of the overall...
instrumentation programme was somewhat neglected. Finite resources, both financial and human, have shown the need to consider the instrumentation programme as a single entity that can be optimized to derive the maximum benefit from minimized resources.

The IAEA Integrated Safeguards Instrumentation Programme (I²SIP) is a recently established programme of the Department of Safeguards that is intended to address systematically instrumentation aspects in order to optimize the IAEA infrastructure needed for development, support and deployment of safeguards instruments.

In general terms, the I²SIP objectives are the following:

— To reduce the proliferation of customized instrumental systems;
— To reduce all costs that depend directly on the number of implemented instrumental systems;
— To better utilize Member States Support Programmes (MSSPs) development resources;
— To enhance the capability of the IAEA to meet specific facility requirements with modular hardware and software components.

Conceptually, all IAEA safeguards instrumentation (for present and future equipment) is included in the I²SIP, although practicality will prevail in implementing changes as a function of time. By necessity, the I²SIP is a long term programme; however, a vigorous initiation activity is necessary if the first benefits are to be realized over the next few years.

2. PRESENT STATUS OF IAEA SAFEGUARDS INSTRUMENTATION

Most of the instruments currently deployed by the IAEA were developed over the last two decades through the co-operation of Member States. It was recognized early that technical requirements and the application environment would dictate the use of custom designed instruments. Instrument applications include situations where inspectorate instruments either are permanently installed in a facility or are stored there for rapid set-up and use. In situ instruments, when appropriately maintained, offer several positive features:

— Continuous operation at key measurement points;
— More reliable operation, avoiding risks of damage during transportation.

The other general class of inspection instruments includes those that are genuinely portable. Preferably, this equipment can be carried by the inspector to the facility when it is needed. To be practical, the equipment must be as small and light in weight as possible; ideally, it should fill only half a briefcase. For a general purpose measurement instrument, we have not yet reached this ideal stage of miniaturization.
**TABLE I. IAEA INSTRUMENT INVENTORY STATISTICS FOR 1993**

<table>
<thead>
<tr>
<th>Equipment group</th>
<th>Number of types</th>
<th>Items</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Measurement instruments</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Low resolution $\gamma$ ray measurement devices</td>
<td>2</td>
<td>51</td>
</tr>
<tr>
<td>Portable multichannel analysers</td>
<td>4</td>
<td>99</td>
</tr>
<tr>
<td>High resolution $\gamma$ ray spectrometers</td>
<td>4</td>
<td>36</td>
</tr>
<tr>
<td>Neutron coincidence measurement units</td>
<td>11</td>
<td>51</td>
</tr>
<tr>
<td>Cerenkov viewing devices</td>
<td>2</td>
<td>65</td>
</tr>
<tr>
<td>Spent fuel verification instruments</td>
<td>4</td>
<td>66</td>
</tr>
<tr>
<td>Ultrasonic thickness gauges</td>
<td>1</td>
<td>42</td>
</tr>
<tr>
<td>Load cell based weighing systems</td>
<td>1</td>
<td>64</td>
</tr>
<tr>
<td>Detectors not allocated to systems</td>
<td>2</td>
<td>141</td>
</tr>
<tr>
<td><strong>Monitoring instruments</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Spent fuel bundle counters</td>
<td>1</td>
<td>30</td>
</tr>
<tr>
<td>Core discharge monitor systems (number of monitors: 16)</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>Portal/penetration monitor systems</td>
<td>2</td>
<td>3</td>
</tr>
<tr>
<td><strong>Optical surveillance</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Single-camera TV units</td>
<td>3</td>
<td>164</td>
</tr>
<tr>
<td>Multi-camera TV systems (number of cameras: 103)</td>
<td>2</td>
<td>15</td>
</tr>
<tr>
<td>Photo surveillance units</td>
<td>3</td>
<td>234</td>
</tr>
<tr>
<td><strong>Seals</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Metal cap seals</td>
<td>1</td>
<td>14 700</td>
</tr>
<tr>
<td>Fibre optic loop seals</td>
<td>2</td>
<td>880</td>
</tr>
<tr>
<td>Ultrasonic seals</td>
<td>2</td>
<td>170</td>
</tr>
</tbody>
</table>

At present, over 70 types of instruments or instrumental systems (both NDA and C/S) are authorized for inspection use by the IAEA. Some of the instrumental systems are, in fact, families of instruments that function conceptually in the same manner, with similar components organized to conform to a specific application. Consequently, there are actually over 100 different instrument versions available to IAEA inspectors. The total instrumentation inventory consists of approximately
5000 items, of which roughly half are in the field. Table I presents a summary of the IAEA instrumentation inventory for 1993, broken down into major instrument groups.

It is of interest to note the present trends in the application of measurement instruments and systems. The number of quantitative NDA measurements is rather static or only slightly increasing; qualitative NDA is increasing somewhat more; unattended monitoring systems are increasing significantly. The change-over from photographic surveillance to video optical surveillance continues; however, films are still being utilized; the number of metal cap seals used is static or decreasing; the use of in situ verifiable seals is increasing substantially.

Effective implementation of a large number of relatively sophisticated custom designed instruments on a routine schedule is a major undertaking. The worldwide implementation of instruments at nuclear facilities increases the difficulties encountered by several orders of magnitude. In any case, effective utilization of instruments requires a substantial supporting infrastructure. Consequently, it is necessary to avoid, if possible, a situation where several instruments fulfil similar functions. For example, for all unattended radiation monitoring systems the count rate must be recorded as a function of time. This function is implemented now in the core discharge monitor (CDM), the Consulha containment and surveillance system, the spent fuel bundle counters, and the monitors of the Power Reactor and Nuclear Fuel Development Corporation.

The large number and variety of instruments imposes severe strains on the financial and human resources of the IAEA. Instrument hardware and software need to be maintained. Inspectors have to be properly trained to be able to operate the instruments. Support staff with an intimate knowledge of the instruments should be available. The operating procedures need to be documented. The equipment inventory must be maintained and managed so that properly operating and calibrated instruments are available when they are needed at facilities around the world. Finally, since the IAEA deploys custom designed instruments in relatively small numbers (at most a few hundred) it is sometimes difficult to find a manufacturer of these instruments.

The resources required to support safeguards instrumentation can be classified into two categories: initial resources (one-time efforts) and continuing resources (operating). Although it is nearly impossible to precisely determine the average resource requirements for such a disparate set of instrument types, it is informative to extrapolate from experience.

It has been estimated that 3 person-years are initially required to prepare a major instrumental system to be used for safeguards purposes. It should be emphasized that these resources are primarily scarce IAEA resources; they include:

1.5 person-years
— to develop, write, review and test application procedures and software;
— to write and review instruction manuals;
— to obtain approval for procedures and instruction manuals.

1 person-year
— to develop support services including:
  procurement specifications; maintenance training; acceptance tests; functional
  and calibration tests; installation procedures; shipping and receiving packages
  and procedures; performance monitoring procedures.

0.5 person-year
— to develop and make administrative arrangements for training programmes and
  training aids (e.g. videos).

The continuing (operating) costs to support an instrumental system vary widely, depending on the type of instrument, the number of instruments that the IAEA has in store, and the way and place of application. One of the more mature instrumental systems is the neutron coincidence counting system. There are actually 20 variations in use (primarily with different detector geometries). The typical support costs are as follows:

— Software maintenance, testing and training instruction 1.8 person-years
— Hardware maintenance, calibration, inventory control, training instruction 1.5 person-years
— Training courses 1.3 person-years

For some instrumental systems that are integrated into a facility the State (or operator) may maintain instruments and provide the training on the site. This, of course, relieves the IAEA of the direct expense for maintenance of instruments. In 1992 the maintenance cost of the Consulha system at La Hague was US $40 000 for an on-call commercial maintenance contract. Training consisted of two separate sessions of three days each, with three instructors in the professional category who were attended by seven inspectors. When maintenance is performed by the facility operator or the State, additional inspector time may be required to recheck the functionality of the instrumental system.

Resource requirements for instrumental support are a consequence of deploying instruments for international safeguards purposes. These resource requirements are substantial and are primarily dependent on the numbers of new and different instruments deployed. A primary objective of the I²SIP is to optimize the number of fundamentally different instrument building blocks in order to minimize the resources needed to support the instruments.
3. FUTURE TARGETS FOR THE IAEA INSTRUMENTATION PROGRAMME

If the present rate of increase of newly authorized instrument types continues, the IAEA will have more than 100 authorized instrumental systems in three years, and more than twice the present number of 76 instrumental systems in ten years. Figure 1 illustrates the historical growth of authorized safeguards instruments since 1983.

This is a disturbing situation, since some aspects of the IAEA instrumental support infrastructure are already saturated. In particular, for those instruments that are widely used, the tendency over the last few years has been to ask Member States providing support programmes to supply cost-free experts (CFEs) to help the IAEA in its initial efforts to bring instruments to a smoothly running implementation stage. In most cases the Member States have been most generous in providing assistance. Unfortunately, management of the MSSP contributions to instrumentation development with regard to standardization and optimization of the safeguards instrument inventory has turned out to be more difficult.

Whether the statistics presented in Fig. 1 will be an accurate basis for predicting future needs for safeguards instrumentation is uncertain. However, the likelihood of substantial additional changes in equipment is high, for the following reasons:

— The periods between inspectors' visits will probably become longer;
— There will be more integrated, in situ instrumental systems at large facilities with limited access;
— The technical capabilities of the IAEA in the area of detection of undeclared nuclear activities will be implemented.

FIG. 1. Historical growth of IAEA authorized safeguards instruments since 1983. In November 1993, 76 instrumental systems were authorized, of which 37% are in situ instruments (16% in 1983).
Experience has also shown that although instruments become outdated and are superseded by more modern and more effective equipment, reduced use of old instruments may continue for an extended period of time. Consequently, both the new instruments and the old ones will have to be supported.

What are the possibilities to alleviate this situation? It is unlikely that the budget situation will dramatically improve in the foreseeable future, nor can we expect a decreasing dependence on the use of safeguards instruments. After all, one of the main reasons for using the appropriate technology is to reduce the dependence on even more costly alternative systems while retaining a credible degree of effectiveness. It might be possible to apply measures of arbitrarily specifying a reasonably limited set of instrumental tools, designed to meet all currently known verification situations. This would create, at least initially, substantial hardships for inspectors who are familiar with the present wide variety of instruments; it might also create problems for Member States that wish to demonstrate their willingness to accommodate international safeguards.

The I²SIP attempts to achieve more optimization through an evolutionary approach. New equipment shall be designed in accordance with the overall instrumentation infrastructure, and parallel development of equipment that can be used for similar verification needs shall be avoided, as far as is practicable. It is necessary to develop modularized hardware and software that is reusable in a variety of situations. New models of old equipment will be designed to fit into the more standardized approach. At the same time, the technology must not be restricted to a limited set of equipment since this would prevent improvements and adaptation to specific applications.

The evolutionary approach will not yield immediate results. Achievement of first benefits will require a period of at least 3–5 years. Over a period of 10 years the programme should mature to the extent that a single set of hardware modules will provide all the capability needed to conform to any safeguards situation that is likely to be encountered in safeguards implementation. The modules must be designed to incorporate new components when needed. In turn, the standardized modules can utilize standard software, especially the higher level software needed to interface with an inspector or with any other instrumental systems/data processing systems that may be co-located at a specific facility.

In principle, a single multipurpose data acquisition module should suffice for a variety of detectors and should be able, for example, to monitor and count radiative (gamma and neutron emitting) CANDU bundles as well as to provide some spectral gamma data for enrichment measurements. This is already possible by using modern, remotely configurable gate arrays. Time will tell how far-reaching this concept is in practice.

A standard data acquisition module coupled to a digital image surveillance module would provide a powerful versatile tool for most safeguards situations. However, digital image surveillance is on the verge of being developed to a standard
that will provide the reliability and cost range required for safeguards purposes (one or two fundamental developments are still needed). These improvements will be made in the next two to three years.

The above discussion pertains primarily to integrated instrumental systems that are permanently installed and continuously operating. Most, if not all, of the above concepts are equally applicable to single-purpose measurement systems that are stored at the facility and operated by an inspector during a physical inventory inspection. A major advantage of the standardized approach is that it uses the same inspector interface as other, more integrated systems. It is expected that the special training and operating requirements will be reduced.

The overriding design goal for truly portable instruments is minimum size and weight, and battery operation; all other features are secondary. However, incorporating those standard features that can be accommodated will provide corresponding benefits. Inspector interface and standard software are two aspects that should always be considered.

Fortunately, the required functions of portable instruments are limited and can be achieved with a minimum number of instrument types. Support for a variety of portable radiation detectors for spectral and threshold measurements of unirradiated and irradiated nuclear materials is required. Battery operation and computer support for these detectors during set-up, data acquisition and data analysis are essential. It is technically feasible to perform these operations with a single basic instrumental platform.

Ideally, a systematic programme such as the I²SIP will provide a collection of modular instrumental building blocks for in situ instrumental systems and a unique instrumental platform for applications for which portable inspection instruments are required.

4. PRESENT STATUS OF THE I²SIP

In October 1992 the Department of Safeguards formally incorporated the I²SIP into the departmental plan, after informal discussion inside the Department and with Member States. A Consultants Meeting on Integrated Safeguards Package was held in November 1991. A second Consultants Meeting on the I²SIP was held from 30 November to 3 December 1993. Careful and deliberate actions are being taken because of the required MSSP input and support for the I²SIP and the wide ranging implications of the programme.

The I²SIP, as it is presently formulated, includes a number of activities. Evaluation of the efficacy of instrumental standards is one of the more important pursuits. Another near term activity is an analysis of the IAEA instrument inventory, with the goal of optimizing it by reducing the number of instrument types and building blocks needed to perform the required verification tasks. We must find the opti-
mum way between two extremes — the use of custom built instruments for each verification task and the restriction to a few standardized instruments that cannot be used for specific applications in an optimum manner. Other prominent I²SIP features include:

— Unattended monitoring systems: the Mark II spent fuel bundle counter;
— Briefcase instrumentation systems (BIS);
— Digital image surveillance (DIS).

The development of the second generation Mark II spent fuel bundle counter is particularly important since this is the prototype for the next generation of unattended monitoring systems. It is also the test case for the use of the standard VXI instrumental bus.

It is of interest to report the results of the 1993 Consultants Meeting on the I²SIP. Twenty-three representatives from eight Member States participated. Developers, MSSP co-ordinators and experienced industry representatives were present. The discussion topics were wide ranging, including the concept of the I²SIP, the use of an international standard instrumental bus, a standard software protocol, a graphical user interface and back-end data processing. The international standard instrumental bus was of special interest since standard software requires a standard hardware platform. If it is to serve the needs of the IAEA, the standard instrumental bus must be recognized internationally; it must also be able to accommodate the analogue and digital data signals that are currently used or are expected to be used in the next few years. It was proposed at the meeting to adapt the standard (IEEE standard 1055) VXI instrumental bus.

It was concluded that standardization of hardware and software are appropriate means to help optimize instrumentation. Because of its open architecture, the standard VXI instrumental bus does not appear to limit technical options; moreover, it is an industrial standard. No viable alternative was presented. A number of recommendations were made, and it was noted that standards are of value as long as they meet the technical needs. Regular assessments of the viability of the VXI bus will, of course, be necessary.

5. FUTURE I²SIP ACTIVITIES

Over the next three to five years the most important I²SIP activities will include the following points:

(a) Demonstration of a standardized, modular instrument design on at least one example — the CANDU Mark II bundle counter;
(b) Acceptance by Member States and developers of the selected VXI bus as an instrumental standard for in situ equipment;
(c) Integration of radiation monitors, digital image surveillance and seals using modular building blocks and networks;
(d) Definition of a single flexible hardware and software platform for portable instruments;
(e) Definition of a software standard to be used on portable computers providing database and instrument support tools.

In the near future, completion of the Mark II bundle counter is of prime importance. It must be shown that the concepts of hardware and software modularity can be realized. It must be demonstrated that certain software packages can be used to reduce the software development time. Later, it must be proven that transportability of the hardware and software to other instrumental systems is advantageous.

At present, the hardware of the Mark II bundle counter is basically complete. This includes the data acquisition module and the instrumental crate. The software for the module is also nearly complete, as is the interface software that connects the module with the embedded controller of the VXI bus instrument crate. The design of the graphical user interface is at an advanced stage.

The Mark II data acquisition board can be remotely configured. It is easy to adapt the board to various detectors through the use of software. Collection of data will be autonomous, in the sense that a loss of mains power to the instrumental crate will not cause a loss of data (10 megabytes of non-volatile memory will be available). Using this flash memory will enable the system to recover from problems, since the operating parameters are non-volatile in the memory, or a separate set of default parameters can be used. Consequently, there should be no difficulties with regard to unrecoverable 'hang-ups'.

Provided the development of the Mark II bundle counter is successful, some of its components can be directly incorporated into other unattended monitoring systems that are under development or are being contemplated, thus realizing a major I²SIP objective. The proposed unattended monitoring systems for which the application of the modular VXI bus concept is considered include: (1) the core discharge monitor being developed in Canada; (2) the Consulha containment and surveillance system being developed in France; and (3) the integrated verification system to be developed in Germany. The safeguards design for the JNFL reprocessing plant at Rokkasho Mura could also benefit from this concept.

At present, one of the main activities in connection with the I²SIP is the completion of the software modules of the Mark II bundle counter. The concept and the specific modules will be carefully evaluated in order to ensure that design problems are not passed on to other systems that incorporate Mark II components. Eventually, this will help realize another I²SIP objective, namely that of developing more reliable systems from proven components.

Under the DIS programme, in the next three years modules will be developed that can be used to reach the I²SIP goal of integrating safeguards instruments in
standard (VXI) configurations. Utilization of the standard VXI bus will enable instruments to be truly integrated, synchronized and incorporated into networks in large nuclear facilities.

Once the initial design and evaluation of these instrumental systems based on the standard VXI bus are completed, there will be sufficient experience to specify the modifications that should be built into new versions of other safeguards instruments. We are confident that with the integration of NDA and C/S equipment into an international instrumentation standard, the variations in present instrumental systems can be drastically reduced so that, instead of having to face a doubling of the number of authorized systems in the next ten years, we can look forward to a reduction or at least only a modest increase in these systems.
INTEGRATION OF SURVEILLANCE DATA WITH OTHER SENSOR DATA

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Abstract
INTEGRATION OF SURVEILLANCE DATA WITH OTHER SENSOR DATA.

Integration of video surveillance with the recording of signals from other sensors into a single system will improve the effectiveness of safeguarding complex nuclear installations and reduce the effort required for data review. Sensors that could be combined with optical surveillance are radiation monitoring devices, electronic seals, and environmental and event sensors. The paper deals with some aspects of system integration that are related to the acquisition of data from different sensor types, their processing and the control between sensors. Several examples of technical solutions, already implemented or in development, are presented.

1. INTRODUCTION

The sensors that could be integrated with video surveillance in nuclear safeguards applications are mainly of the following types:

(a) Radiation monitoring devices such as gamma and neutron detectors;
(b) Electronic seals;
(c) Environmental sensors for temperature, etc.;
(d) Event sensors such as proximity switches or motion detectors.

The sensors in groups (a) and (c) normally generate analogue output signals, whereas the sensors in groups (b) and (d) generate logical (on/off) signals. Data from radiation monitoring and video surveillance require a more intensive and dedicated pre-processing than those from the other groups.
A major problem in integrating video surveillance with other sensors arises from the different nature of the data. Video surveillance produces video sequences, whereas other sensors generate measurement values or on/off information. For an interpretation of the surveillance data an inspector has to review the recorded video sequence. The computer only assists in selecting scenes in which an activity has been detected (with or without safeguards relevance). The introduction of digital surveillance systems will facilitate the storage and retrieval of mixed information from sensors. This paper discusses some problems connected with integrated systems and refers to some technical solutions.

2. DATA ACQUISITION

The data acquisition system converts the sensor signals, acquired either in analogue form or in digital form, to a computer readable format. Analogue-to-digital conversion requires a sampling rate that is appropriate for the signal bandwidth. Moreover, the modules performing this conversion should be near the sensors with a low analogue signal level (group (a)), whereas digital and video signals can be sent over longer distances via transmission lines. This requirement influences the overall system architecture, including the location of acquisition stations and the central collection/review station as well as the layout of the local area network.

Many commercially available hardware products, especially for the PC and the VME bus extension for instrumentation (VXI), perform acquisition from different sensor types and include software drivers and utilities. However, they do not have features that would allow authentication, tamper resistance or encryption. Therefore, authenticated data transmission lines must be inserted between the sensors and the acquisition system if they are in different housings and if they are unattended.

An example of an integrated system for the acquisition and transmission of nuclear instrumentation data and video surveillance is the burnup device (BUD) developed by the CEA [1]. In this system the gamma/neutron detector signals are acquired by dedicated instrumentation modules, pre-processed by a PC based local station and sent to a remote review station. The video signals are sent over a separate fibre optic cable to the review station. Since it is possible to code the NDA data and the video data in both transmission links, they can be compared in the review station to perform authentication.

A surveillance/monitoring system, which acquires images from several surveillance cameras and analogue/digital signals from different sensors, is the computer aided video surveillance CAVIS-2S system [2]. Its requirements, in particular the inspector interface, were specified by Euratom, Luxembourg. In the basic configuration of this system, the input capacity is 16 analogue channels and 48 digital channels, besides the 16 video channels. The flexible hardware/software structure allows connection with systems of other developers. Tests in the LaSCo laboratory
at JRC, Ispra, showed that the sensor network integrated monitoring system (IMS), developed by Sandia National Laboratory (see Ref. [3]) could be interfaced easily with the CAVIS-2S system.

It is planned to integrate the BUD and the CAVIS-2S systems for a future application in France. Figure 1 shows a block diagram of the integrated system. The alarms detected by the BUD computer on the gamma and neutron chains are transmitted over digital I/O lines to the CAVIS control computer. A serial communication line links both computers for file transfer.

Other integrated systems based on the VXI bus are in development in Canada (spent fuel bundle counter Mark-II) and in Germany (PKA — an integrated verification system).

Another aspect of an integrated data acquisition system is the synchronization of the local acquisition stations. Independent time keeping based on local computer clocks can lead to significant timing errors over long periods. A possible solution is to use a single computer system as a master system which sends the time information over the network in order to synchronize the other stations.

3. DATA PROCESSING

The first and principal processing task is the detection of alarm conditions in the different sensor signals. The detection of alarm conditions by on/off sensors is

FIG. 1. Block diagram of the integrated BUD/CAVIS-2S system.

UPS = uninterruptible power supply.
### ALARM HISTORY

<table>
<thead>
<tr>
<th>Date</th>
<th>Time</th>
<th>Status</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>5.07.93</td>
<td>11:04:12</td>
<td>Normal</td>
<td>Start of a new surveillance period</td>
</tr>
<tr>
<td>5.07.93</td>
<td>11:04:13</td>
<td>ALARM</td>
<td>Front doors of cabinets</td>
</tr>
<tr>
<td>5.07.93</td>
<td>11:04:20</td>
<td>ALARM</td>
<td>Read after Write signal from recorder</td>
</tr>
<tr>
<td>5.07.93</td>
<td>11:04:40</td>
<td>LOW</td>
<td>Mains supply for air conditioner No.1</td>
</tr>
<tr>
<td>5.07.93</td>
<td>11:04:54</td>
<td>Normal</td>
<td>Read after Write from recorder</td>
</tr>
<tr>
<td>5.07.93</td>
<td>11:05:13</td>
<td>ALARM</td>
<td>Emergency exit</td>
</tr>
<tr>
<td>10.07.93</td>
<td>10:44:33</td>
<td>ALARM</td>
<td>Interruption for a system test</td>
</tr>
<tr>
<td>10.07.93</td>
<td>10:56:11</td>
<td>Normal</td>
<td>Restart after a system test</td>
</tr>
<tr>
<td>10.07.93</td>
<td>10:56:50</td>
<td>ALARM</td>
<td>Power failure</td>
</tr>
<tr>
<td>10.07.93</td>
<td>10:56:50</td>
<td>ALARM</td>
<td>Temperature limit exceeded</td>
</tr>
<tr>
<td>15.07.93</td>
<td>10:07:00</td>
<td>ALARM</td>
<td>Gamma detector No. 2</td>
</tr>
<tr>
<td>16.07.93</td>
<td>15:10:93</td>
<td>ALARM</td>
<td>Motion detector</td>
</tr>
<tr>
<td>16.07.93</td>
<td>16:30:45</td>
<td>HIGH</td>
<td>Container vibration</td>
</tr>
</tbody>
</table>

### DIALOGUE

Use cursor keys or the mouse to scroll the alarm display. Press "Esc" to terminate.

![Diagram](a:/cvs log.da2)

**FIG. 2.** Example of integrated alarm history list of a CAVIS-2S system.

straightforward, whereas the alarm condition of a simple analogue sensor (environmental sensor) is defined by programmable upper and lower thresholds. An example of an integrated alarm history list of a CAVIS-2S system is given in Fig. 2. This list includes in chronological order the following information: date and time of each alarm or of the return to normal conditions, the digital or analogue input line involved, the type of alarm detected (threshold, video, etc.) and the description of the sensor causing the alarm.

Alarm detection in images from video cameras and in nuclear instrumentation signals requires more intensive processing and involves dedicated hardware and software. In most safeguards applications, only low level processing (on a pixel level)
of the digital surveillance images is carried out. The objective of such processing is the detection of scene changes without any influence of variations in illumination. Several methods have been developed. One of these is the Polyline method, which correlates a subset of pixels from each image with that of a reference image.

High level image processing methods such as image segmentation and object recognition require long execution times which are normally inadequate for safeguards applications. Powerful computer architectures, based on multiprocessor configurations or dedicated processors, have to be introduced to apply these methods. Moreover, the application of high level processing in safeguards has to face at least two problems: the difficulty of defining a priori the objects to search for, and the possibility of tampering by modifying the shape of an object.

FIG. 3. Correlation of surveillance pictures with other sensor data.
Processing of radiation monitoring data has been improved over the last years, and good results in detecting alarm conditions are being obtained. The BUD system, which processes outputs from gamma and neutron detectors, is able to identify transfer movements of fissile material with good results: of 800 movements, only 3% have not been identified.

Another processing task concerns the retrieval of the alarm data and the sorting of alarms according to certain criteria, e.g. time period and sensor type. An integrated system should provide tools for correlating information from different sensors. For instance, if there is an alarm from a gamma/neutron detector, the system should be able to display the surveillance images taken by selected cameras at the same time. A possible mode of correlating surveillance pictures with other sensor data is illustrated in Fig. 3. It has been developed for the computer aided review station CARES [2].

An integrated graphical display of all sensor information and control functions on a single screen facilitates data interpretation in complex systems. This technique has been applied with success to the human computer interface of the remote verification project using mobile robotics [4].

4. CONTROL BETWEEN SENSORS

An integrated system could provide control of a set of sensors, taking into account the results of data processing. One of these control parameters is the sampling rate of the data acquisition process. It is often useful to acquire more data from some sensors at an increased sampling rate when an alarm is generated by another sensor type. Such a control feature allows for a better time synchronization during later review and analysis. The following example illustrates this possibility. If a neutron or gamma detector sees an alarm, the system should increase the image capture frequency of the video cameras pointing to that area. The sampling rate for normal conditions should be as low as possible to reduce the amount of recorded data. The same control feature will trigger a measurement cycle of the nuclear instrumentation if an activity is detected by the corresponding video cameras. This interaction between sensors will be developed in the integrated BUD-CAVIS system. The sampling rate of video channels in CAVIS is computer controlled and can be modified in alarm situations. The additional information from different sensors during an alarm situation increases effectiveness and facilitates data interpretation.

5. CONCLUSIONS

System integration covers many areas, such as system architecture, networking, data acquisition, authentication, data processing and remote interrogation. The
numerous combinations of sensors and the variety of plant characteristics require flexible, modular systems. We have discussed some aspects of system integration that are related to the acquisition of data from different sensor types, the processing of the data and the control of the sensors.

Several integrated safeguards systems are under development. The experience that will be gained from these developments will contribute significantly to the preparation of valid guidelines for future systems. The large efforts required for these developments should stimulate stronger collaboration between developers to facilitate the future integration of their systems.

REFERENCES


INTEGRATED SAFEGUARDS SYSTEMS*

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Abstract

INTEGRATED SAFEGUARDS SYSTEMS.

The paper describes integrated non-destructive assay (NDA) and measurement systems that have been developed and installed. These systems reduce inspection manpower requirements and lessen the impact of inspections on the operator. Systems that have been developed for safeguards measurements and monitoring include the following: the dynamic materials control accounting system for a plutonium processing facility; an integrated NDA system for a plutonium scrap recovery facility; continuous, unattended radiation monitoring/measuring systems for a range of specific applications; and the Hanford integrated waste tank monitor. Integrated systems of the future are also discussed, as are some of the key capabilities of integrated systems.

1. INTRODUCTION

Integrated measurement and monitoring systems of various types for safeguards have been in use for the past several years. Systems that have been developed include a dynamic materials control (DYMAC) [1] accounting system for a plutonium processing facility, an integrated non-destructive assay system for a plutonium scrap recovery facility [2], continuous, unattended radiation monitoring/measuring systems for a range of specific applications, and the Hanford integrated waste tank monitor [3]. Experience gained from operation of these systems is applicable to future systems that might be used by the IAEA. The IAEA is interested in using integrated systems as a means for reducing inspection manpower required to achieve safeguards goals at facilities. Some of the key components of these integrated systems include the capability to link equipment, to provide front end

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triggers, to automate data collection, and to provide on-line data processing, continuous unattended operation and remote access. Non-destructive assay (NDA), containment and surveillance (C/S) and other equipment would be linked. Many data would be collected by these systems, but only pertinent information would be routinely provided to inspectors. Inspection manpower is potentially saved by reducing the number of trips to facilities, the time needed to collect information and the time spent in reviewing data.

In this paper we describe various integrated systems that have been developed and discuss what we have learned from operation of these systems. We also identify key elements that should be considered in building future integrated systems.

2. EXPERIENCE GAINED WITH INTEGRATED SYSTEMS

2.1. DYMAC

The DYMAC system was developed for the plutonium processing facility in Los Alamos. The system automatically transmits data from measurement terminals located throughout the facility into a central computer where the data are processed. The original system of installed instrumentation included 37 video terminals, six supervisory hard-copy terminals, five printers, 38 5.5 kg balances, three 15 kg balances, 20 neutron coincidence counters, three solution assay systems and two segmented gamma scanners. In the end, only the balances were monitored on line; data collected by the other instruments were entered manually. The application of the DYMAC system was our first experience with integrating equipment and data. The system was limited by the computer hardware and by the database management systems that were available at the time. The personal computers, networking and shared database systems that are now available would eliminate many problems that limited the use of the DYMAC system.

2.2. New special recovery facility

An integrated system of automated NDA instruments was developed to provide nuclear materials control and to process monitoring information for the plutonium scrap recovery facility at the Savannah River site. The goal of the development was to provide an accountability system to draw frequent material balances with minimum reliance on laboratory measurements of analytical samples. Ten NDA instruments were developed by four laboratories to provide assays of plutonium process solutions, bulk materials and solid waste. A list of the instruments is shown in Table I. Each of the NDA instruments receives information from and reports measurement results to an integrated control computer where information for the accountability system is maintained.
TABLE I. NDA INSTRUMENTS FOR INTEGRATED MEASUREMENT SYSTEMS

<table>
<thead>
<tr>
<th>Instrument</th>
<th>Supplier</th>
<th>Measurement type</th>
</tr>
</thead>
<tbody>
<tr>
<td>Feed coincidence counter</td>
<td>LANL</td>
<td>Effective $^{240}$Pu mass</td>
</tr>
<tr>
<td>Calorimeter (4)$^a$</td>
<td>Mound</td>
<td>Heat</td>
</tr>
<tr>
<td>Solids isotopic analyser</td>
<td>LLNL</td>
<td>Plutonium isotopic fractions</td>
</tr>
<tr>
<td>Turbidimeter</td>
<td>SRL</td>
<td>Suspended solids</td>
</tr>
<tr>
<td>Densitometer</td>
<td>SRL</td>
<td>Solution density</td>
</tr>
<tr>
<td>X-ray fluorescence (2)$^a$</td>
<td>LLNL</td>
<td>Plutonium concentration</td>
</tr>
<tr>
<td>Gamma pulse height analyser (2)$^a$</td>
<td>LLNL</td>
<td>Plutonium concentration</td>
</tr>
<tr>
<td>Low solution assay instrument</td>
<td>LANL</td>
<td>Plutonium concentration (low)</td>
</tr>
<tr>
<td>Waste coincidence counter</td>
<td>LANL</td>
<td>Effective $^{240}$Pu mass</td>
</tr>
<tr>
<td>NaI monitor array</td>
<td>LANL</td>
<td>$^{239}$Pu mass</td>
</tr>
</tbody>
</table>

$^a$ Number of individual measurement instruments.

2.3. Continuous, unattended radiation monitoring/measuring systems

Various types of continuous operation, unattended radiation monitoring, NDA systems are in routine operation. Information about these systems is included in a companion paper [4]. Summary information is provided here for completeness. Unattended monitoring systems are an important part of integrated systems being planned for the future.

At the thermal oxide reprocessing plant (THORP) in Sellafield, United Kingdom, a $\gamma$ ray and neutron detector (GRAND) electronics unit connected to radiation sensors triggers a video system that records images of the serial numbers on racks containing spent fuel when the racks are moved to and from the main storage pool. This system was developed in conjunction with Sandia National Laboratories (SNL) and Euratom. Los Alamos National Laboratory (LANL) provided the radiation detection equipment, and SNL provided the video surveillance system, including an underwater camera system. The system has two types of triggers. Ion chambers detect the movement of radioactive materials through the transfer channel, while other switches sense the movement of the rack transfer machine. Pertinent information from the triggers is annotated on the recorded video image. The GRAND also stores data that can be downloaded on command. These data include the time and
date of movement, the direction of motion, the highest measured signal level, and information concerning unusual events that may have occurred. Some performance monitoring data are also stored. The GRAND data provide redundant information in support of the video recording system.

The THORP system is being updated. The video system is now connected by a local operating network (LON) controller using Echelon technology that is being developed by SNL and LANL. Eventually, the GRAND will also be connected to the LON through interface nodes. The GRAND will send information on the network to trigger the video system and to communicate with the computer in the inspector's room at the facility. Data are automatically authenticated by the interface nodes.

A total of nine NDA systems using neutron coincidence counting units have been installed in the Plutonium Fuel Production Facility (PFPF) of the Power Reactor and Nuclear Fuel Development Corporation (PNC) in Japan. Many of these systems have duplicate electronics and Collect computers for redundancy. The systems monitor continuously automated movement of nuclear material and perform assays to determine the masses. In some cases, camera systems are triggered by the NDA stations to record sample identifications. For these systems, data are stored in the instrument computer and are manually retrieved and then transferred to a review computer. There is no electronic network linkage other than a direct connection between the NDA equipment and the camera systems.

Four neutron coincidence counting NDA instruments have been built for the Siemens mixed oxide fuel fabrication facility in Germany. These instruments will monitor and assay automated movement of fuel-pin trays into and out of a storage area. The NDA computer is time synchronized with an electronic-mechanical sensor (EMS) computer that monitors tray position sensors. The NDA and EMS data stored on the individual computers are manually transferred by a computer disk to a computer that merges the data for review by an inspector; system integration is done manually.

Core discharge monitors (CDMs) installed in on-load reactor facilities detect removal of irradiated fuel bundles from the reactors. In these systems, some performance monitoring data are enunciated and used to monitor for problems in near real time. Data from the CDMs are retrieved every 90 days and reviewed to verify fuel movement. This system is integrated by connecting two Collect computers to multiple GRAND units. A 'watchdog' processor controls routing of GRAND data to the Collect computers.

Fuel flow monitors have been installed at various locations in the fast reactors Joyo (four systems, one with redundant electronics) and Monju (three systems, two with redundant electronics) at the PNC in Japan. These radiation detection systems monitor input into the fresh fuel and the spent fuel stores as well as other areas in which fuel is moved in these fast reactor facilities. Inspectors visit the facilities every 30 days to gather and review data from the monitoring stations. These systems are not integrated into a single network.
2.4. Hanford integrated waste tank monitor

A data acquisition and control system (DACS) has been developed and installed to control and monitor tank 101-SY at the Hanford site in the USA. The DACS is used to monitor physical parameters in the waste tank, such as the level and temperature and gas characteristics, including pressure, flow and hydrogen percentage. The DACS is also used to monitor and control mitigation equipment such as the 150 hp (112 kW) pump used to circulate the waste. Genesis process control software was selected to be the core of the DACS, and Modicon hardware was selected for I/O. The real time operating system has multi-user and multifunction capability. Real time data are displayed on screens developed for operator action and monitoring. The DACS has more than 200 I/O channels. These are scanned as fast as once per second or as slow as once every 30 seconds. The software and data review technology developed for the DACS project might prove useful for applications by the IAEA to complex integrated safeguards systems with multiple sensors.

3. INTEGRATED SYSTEMS OF THE FUTURE

In the era of zero growth budgets and expanding requirements due to inspection of additional facilities, the IAEA is interested in integrated, unattended, continuous operation systems as a means for maintaining safeguards goals while reducing inspection manpower requirements. Use of integrated systems reduces inspection activities:

- Networking equipment and centralizing data collection in the facility: data are automatically collected and stored at one location, thus reducing the time spent by inspectors at the facility;
- Front end triggering: data are stored when significant events are recognized, thus reducing the amount of data being stored and the time required to review data;
- Automated data review analysis and reporting;
- Electronic transmission of data to regional offices or headquarters, eliminating the necessity of some visits to the facility by inspectors.

Integrated systems can be used for various purposes. In some facilities, the system will be installed by the IAEA or the operator strictly for safeguards purposes. In other cases, data from integrated systems, installed by the operator for process control and material accountancy purposes, could also be used for safeguards purposes. Application of generic systems may be possible in some cases, for example in reactor facilities, but most integrated systems will be customized adaptations. Usually, for large bulk facilities, operators will pay a significant portion of the capital costs associated with installing the equipment.
A conceptual design of a complex integrated safeguards system is shown in Fig. 1. This complete system has various components, including C/S and NDA instrumentation, a review station and a facility data acquisition station. Components are linked by networks. Connections could be made through Echelon interface nodes, through a standardized bus such as VXI, or through any other means for transmitting data. All forms of data, for example video, sensor and NDA data, would be transmitted on the networks. Individual instruments will analyse and process data. For instance, an NDA system or radiation detector could send a trigger signal to a video surveillance system. When the surveillance system is triggered, it would be switched on and transmit information over the network to a video recorder and to the review computer.

FIG. 1. Combined material control and accountancy system.
The integrated system would be used by both facility operators and IAEA inspectors. Safeguards information would be analysed by the review computer. Facility process control and accountancy data would be processed by the facility acquisition computer. Both users would share information from the NDA and C/S instruments. Safeguards information extracted from the network must be authenticated.

Integrated systems will be software intensive. Software must control the instrumentation, interpret and react to sensor information, interact with users, collect data automatically and reliably, analyse and process data, provide performance monitoring for components, identify anomalous events and recover from them, and transmit data on demand. When possible, commercially available software products should be used. However, specific software will still be needed to meet the specialized requirements for IAEA surveillance and nuclear material accountability.

To achieve the goal of reducing inspection manpower, two applications of integrated systems are envisioned.

One obvious application is in bulk processing facilities with large material throughputs. Automated data collection at key material locations using integrated systems will reduce inspection time compared with the time required to gather information manually. These systems will provide additional information because some equipment could be used in areas that might not be accessible when the facility is in operation. The accuracy is improved when systems are designed as part of the facility, rather than as add-on retrofits, thereby optimizing measurement geometries.

The second application is for existing item facilities, where integrated systems with front-end triggering and remote data transmission may be the mechanisms to reduce inspection time. Triggers would enable the system to eliminate excess stored information by recording only significant events. For many facilities, the ability to remotely access and transmit data may be the most practical means for reducing inspection person-days at a facility. Using the video time radiation analysis program (VTRAP) [5], continued surveillance and improved control and accountability of nuclear materials can be achieved by integration and analysis of data from multiple sensors comprising a continuous, unattended measurement system.

4. SYSTEM CAPABILITIES

In the following sections we discuss some of the key capabilities of integrated systems, particularly those that include NDA equipment.

4.1. Reliability of hardware

The hardware must be reliable. When possible, redundant systems can be used. We have often used completely redundant electronics systems to avoid loss of data
in the event of a failure. After four or five years of experience with large unattended, continuous data collection systems, we are establishing databases on long term equipment operation. In general, the NDA electronics equipment has functioned well. Problems with computers and computer components have been the principal failure mode.

4.2. Robustness of software

Software has to be robust and user friendly. Software must be capable of linking components, collecting and processing large amounts of data, and displaying data in a simple, easy to understand manner. Software for monitoring NDA equipment is evolving as a result of our experience gained from unattended, continuous operation.

4.3. Standard user modes for software

In most cases, facilities have unique features that require facility specific software. Completely generic software is not possible. However, software can be structured to perform basic functions and to adapt to many specific requirements. This software would provide the basis for adaptations to specific applications. When possible, commercially available software products should be used.

4.4. Compatibility and adaptability of hardware

Hardware must be compatible with the integrated networks. It should be relatively easy to add components to the system and to enable these components to communicate with other components in the network.

4.5. Data redundancy and back-up

This is one of the principal aspects of long term unattended, continuous operation. Data should be backed up. If data are transmitted over a network, there should be a mechanism for storing and retrieving the data if the network transmission is interrupted. Data transmission must be authenticated.

4.6. System redundancy

Common failure modes should be avoided and system redundancy should be incorporated. For instance, the primary function of an integrated system might be to record video images and radiation signatures of material being moved to and from a storage facility. Correlation of information from the radiation detectors, such as time, date and signal levels, with video image processing would be an effective
combination of signals to provide redundancy. Similarly, motion detector triggers would provide redundancy in the event of a failure in the radiation triggers.

4.7. Performance monitoring

A mechanism for monitoring the performance of components should be included in the integrated system. For components in which large amounts of data are being collected and stored, performance monitoring data can enable problems to be identified and solved without having to analyse vast amounts of data. Performance monitoring helps identify potential problems before actual failures occur.

4.8. Integrated safeguards systems simulator

Currently, continuous, unattended systems are being tested mainly in the field. Isolated parts of these systems are tested, but, since adequate facilities are not available, complete systems with several components are not fully tested. We need the capability to test continuous, unattended hardware/software by using a simulation facility. The simulator would be used for prototype development and for reviewing and isolating problems identified by performance monitoring field data. We would be able to improve the ability to test, enhance, update and maintain safeguards systems instrumentation and software.

5. CONCLUSIONS

Integrated NDA systems in various forms have been used for several years for process control, material accountancy and inspection purposes. We have shown that integrated NDA systems in large bulk processing facilities do reduce the inspector’s presence. As new large bulk processing facilities are built, we see that trend continuing. Typically, operators provide systems, and inspection agencies access all or part of the equipment for safeguards purposes. Perhaps the largest benefit is data centralization. With the currently available technologies provided by expanded capabilities of personal computers and components, it is easier to connect instruments to networks and to collect data electronically at a central location in the facility. It is also easier to transmit data off the site. For existing facilities, integrated systems are a potential benefit, for two reasons: unattended continuous operation and remote data transmission.

REFERENCES


A CONCEPT FOR INTEGRATED SAFEGUARDS SYSTEMS

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Abstract

A CONCEPT FOR INTEGRATED SAFEGUARDS SYSTEMS.

Safeguarding large automated bulk handling facilities requires new or modified instrumentation for verification, but above all it necessitates a different approach to collecting, archiving, authenticating and evaluating both verification data and operating records. The total volume of data to be processed by safeguards inspectors in these plants easily exceeds anything that can be dealt with manually. In addition, data from different measurement stations, sensors, seals, and optical surveillance must be correlated. The solution proposed (and already partially put into practice) is to network all safeguards instruments through an industry standard data bus (Ethernet) to a central collection and evaluation computer in the inspector’s office.

1. INTRODUCTION

Between the end of 1986 and the beginning of 1988 the Euratom Safeguards Directorate embarked on three major new projects, namely the design of safeguards systems for the thermal oxide reprocessing plant (THORP) at the British Nuclear Fuels plc (BNFL) Sellafield works, the UP3 reprocessing plant on the Cogéma site at La Hague and the Siemens MOX II fuel fabrication facility at Hanau. In comparison to existing facilities these plants had larger nuclear materials throughput and inventories, were extensively automated and used remote material handling. Consequently, safeguards instruments had to be designed and installed in or around process lines to operate in unattended mode. Wherever the instruments formed part of the containment of stores they had to operate in monitoring and logging mode as well. The design included large numbers of video cameras (≥ 20) to maintain continuity of knowledge. Finally, on-site facilities for sample analysis formed an essential part of the safeguards systems.

The volume of information emanating from the safeguards instrumentation in these plants was clearly going to exceed anything that could be dealt with manually. In addition, data originating from different sensors covering the same event (e.g. the video images of the passage of a spent fuel element and the corresponding gamma and neutron signatures) would have to be correlated (matched).
The Euratom Safeguards Directorate therefore began investigating possible ways of integrating instrumentation both locally (e.g. combined neutron/gamma measurement stations) and across the whole facility by networking different local measurement stations.

2. INTEGRATION

2.1. Definition

Integrated safeguards systems comprise a collection of different sensors complete with their power supply, signal processing and data processing equipment, located in different positions across a facility where

— information from different sensors is already combined locally to the extent possible
— local groups of sensors are networked through a suitable bus connected to a central archiving and evaluation computer
— safeguards verification information is sent through the network to the central computer
— safeguards verification information and operating records are matched and compared automatically whenever possible
— a special graphical user interface allows input of inspectors' know-how and judgement.

2.2. Background

Attempts to implement at least partial integration go back more than ten years. Euratom staff at the Joint Research Centre at Ispra proposed a computerized integrated safeguards evaluation system in 1983 [1, 2]; R. Haas and co-workers at Ispra suggested integration of different types of intrusion sensors and seals with video systems [3, 4]; Sandia National Laboratories (SNL) produced the intrusion monitoring system (IMS) combining the MINISTAR video system with contact or other types of switches; SNL and the German Support Programme to the IAEA co-developed the integration of variable coding seal systems and modular integrated video systems (VACOSS-MIVS and VACOSS-MOS [5, 6]); the Commissariat à l'énergie atomique (CEA) built an integrated system for spent fuel containment and surveillance at La Hague (CONSULHA [7]), and SNL and Los Alamos National Laboratory jointly built and tested a channel monitor where video, gamma detectors and position detectors are combined.

None of the systems described in the above paragraph satisfies all the criteria defined in Section 2.1. Integration was effected through cross-triggering of existing
equipment (bottom up approach) and in most cases the resulting information is stored separately with little chance of automatic processing. When designing a plant-wide integrated safeguards system it is necessary to analyse the origin and flow of information and the processing that is required to arrive at safeguards conclusions (top down approach).

2.3. Information flow and information processing

During inspections and evaluations, information contained in operating records, verification results, accountancy reports and material balance area (MBA) rules is collected, archived, collated, compared and judged. Figure 1 shows a simplified scheme for the flow of information from operating records and verification activities. Information is collected at several collection points, moved from the collection point to the inspectors' office where it is sorted, where verification results are grouped by event (different measurements and observations related to one particular material transfer, material stock control, etc.), and where this information is archived in the form of event records. Information originating from the operator is matched to information resulting from verification activities and the resulting paired records are evaluated using information contained in the basic technical characteristics (BTC) of the plant, performance and target values and inspectors' experience.

The verification results are collected at key measurement points and strategic points throughout the MBAs of the plant. They are derived from optical surveillance, seals, measurements of the quantity and quality of nuclear materials by neutron counting, gamma spectroscopy, K-edge densitometry, X ray fluorescence, calorimetry and, where on-site analytical facilities are available, results of chemical analysis received from the on-site laboratories or from the inspectors' sample glove box (Fig. 2), as well as from additional measurements of weight, level, density, temperature, and, last but not least, the item ID, the position of the item during a measurement and the direction of travel of an item (e.g. confirmation that a spent fuel element is moving to the shearing cell and not back).

2.4. Local integration

Each instrument (non-destructive measurement station, sensor, seal and camera) has its own local computer or controller which controls the measurement parameters, collects the raw data into files and adds a header containing detector identity, synchronized date and time, and other parameters relating to the measurement or sensor event. The resulting data are raw event records or files. These are buffered locally to protect against bus failures and to provide independent checks for authenticity of transmitted data files. Each local computer or controller is also fitted
FIG. 1. Data flow and processing of safeguards information.
FIG. 2. Origins of verification results.
with a watchdog type self-diagnostic module which will transmit emergency messages about failures of the local device to the central computer.

Non-destructive measurement stations must also have automatic identity readers (for measurements of different items), sensors to confirm correct positioning of items and to ensure proper measurement conditions (i.e. no interference with shielding, no addition of sources, etc.). The operating system, at least for the non-destructive assay (NDA) stations, should therefore be capable of multitasking.

The surveillance cameras, will either be analogue cameras, with digitization and compression carried out in the camera box, or fully digital cameras such as the Gemini system.

All event records and files are encrypted locally using the data encryption standard (DES) or Rivest, Shamir, Adleman (RSA) encryption algorithms. At pre-defined intervals or when prompted by local logic or from the central computer, event files and diagnostic files will be transmitted to the central computer. Encryption keys will most likely be held on one-time key pads in read only memory (ROM) both locally and in the central computer.

Wherever possible, data from different sensors located at or near the same point are integrated locally, that is, data relating to the same event are marked with the same unique event ID (e.g. results of neutron counting and gamma spectroscopy of the same item, concentration found by K-edge and liquor density of the same input accountancy sample, etc.). In smaller plants this is mostly done manually by sticking a printout onto the same working paper and by entering additional data into results listings. For unattended operation this requires either a local evaluation station (LES) or at least a method of synchronizing date and time on the different local data processing systems and of adding trigger marks. The functions required for local integration are shown in Fig. 3 for the situation with an LES. The local safeguards equipment consists of a number of sensors, their power supply, their signal processing electronics and measurement control electronics. The results from the different sensors are transferred to the input interface of the LES. A control module in the LES manages the two-way communication with all the sensors' signal processing units. It routes the raw data through validation to the appropriate evaluation modules. The results of local evaluation are buffered, encrypted and sent to the central evaluation station at regular intervals. The corresponding data flow diagram is shown in Fig. 4.

2.5. Moving verification data

Data collected locally are transmitted to the inspectors' office. Depending on the volume of data, their frequency and the ease of access to the local evaluation stations, data can be moved manually as hard copies, as files on floppy disk or other suitable media, or through an electronic data bus to the central evaluation station(s). Similarly the operating records may reach the inspectors' office as hard copies, as
FIG. 3. Functional requirements for local evaluation stations.
FIG. 4. Data flow in local evaluation stations.
FIG. 5. Data flow in central evaluation stations.
files on floppy disks or through a link to the operator's own electronic data bus. During their transfer from the collection point to the central system, safeguards verification data have to be protected against falsification of their content (integrity) and their source (authenticity).

2.6. Central archiving and processing of data

The verification data are decrypted and archived at the central evaluation station (Fig. 5). At this stage, information originating from different plant areas referring to the same event (e.g. tank level from pneumercator and Pu concentration from K-edge or from chemical analysis) is also combined into event records (MATCH I). The combined safeguards verification event records are then matched to operators' events extracted from operating records received from the plant (MATCH II). The paired events are evaluated (operator-inspector difference for nuclear materials quantity and quality, confirmation of events or no events for control/surveillance (C/S) devices) and the results of the evaluation form the basis for safeguards reports.

3. AUTHENTICATION

If local verification data are transmitted to the central evaluation station through a data bus, they have to be protected against tampering. To this end they are encrypted by the DES algorithm using a combination of private and public keys generated by RSA algorithms. The keys also identify the origin of the signals. In addition, raw signals and the resulting event records are stored in the LES. The inspectors can always check the integrity and authenticity of records transmitted by comparing records buffered locally with those held in the central evaluation computer(s). Where safeguards verification data are obtained from branching operators' instrumentation, signals have to be branched as close as possible to the raw data source.

4. AVAILABILITY

For each measurement point the consequences of equipment failure have to be assessed. Where appropriate, local equipment, data buses and central evaluation stations must be duplicated. As the most serious weakness of any data bus is the loss of the whole safeguards verification system in the case of bus failure, the bus should be duplicated; the local measurement systems have to operate independently of correct bus functions and they have to provide sufficient buffer memory to cover the longest possible bus failure. Similarly, the central data evaluation station should be duplicated and the power supply to the two central evaluation stations should be from
different bus bars. Local and central storage should be on RAID 5 disk arrays. Where the consequences of the loss of a measurement station are too costly (e.g. flow measurements of items entering or leaving a Pu store), at least the local data processing and storage system should be duplicated.

5. IMPLEMENTATION

5.1. Local systems

Unattended software for combined neutron/gamma measurement and monitoring stations was written by the CEA and Canberra [8]. The first system of its kind, NEGUS 1, has been successfully operated for more than four years at the Cogéma UP3 reprocessing plant at La Hague, France [9]. The software also includes the transmission of measurement and monitoring data to two central evaluation computers. The system is based on special local data acquisition systems (DAS=PC with acquisition interface module (AIM) functionality plus local evaluation capabilities). Copies of NEGUS will be installed in the MELOX plant at Marcoule, France, and in the new reprocessing plant, UP2-800, at la Hague. Software for attended operation of combined neutron/gamma measurement stations has recently been delivered to Euratom by Canberra. This software is based on the Canberra GENIE ESP package. Information from temperature, position and direction sensors is integrated through the use of a DECSERVER 90 device, which allows transparent transmission of data from equipment with serial ports via Ethernet. K-edge and X ray fluorescence software written by H. Eberle of the German nuclear research centre at Karlsruhe, KfK, integrating sensor information on sample positioning, sample temperature and liquor density, has been in use at the La Hague UP3 reprocessing plant since its startup in November 1989. This software also uses Canberra’s GENIE package. Attended neutron/gamma software and all K-edge software is implemented on VAX stations 3100, 4000 and AXP 3000-300L under the VMS operating system.

Tank levels, liquor densities, temperatures and weights for key tanks or silos are monitored and logged on Ranger loggers. Loggers are also employed as local backup for unattended neutron/gamma measurement evaluation.

5.2. Data bus

For the Siemens MOX II project a broad band bus solution (Siemens H2B) was chosen to allow for the bandwidth of analogue video signals. In the meantime digital video has become a reality and all safeguards verification results can now be handled by Ethernet. The protocols are either DECNET or TCP/IP. The cable runs do not
FIG. 6. Block diagram of System 7.
have to be protected against tampering or eavesdropping provided that the signals are encrypted.

5.3. Central evaluation

Currently, central archiving and evaluation are carried out on VAX stations 3100 and 4000 operating under VMS. In future this will migrate to AXP 3000-300 X stations. The software running on the central measurement stations is part of the NEGUS package. Specifications for the system collecting and processing all safeguards verification measurements (System 7) have been completed and a call for tender has been issued. It is hoped that the first system will be in place by the end of 1994. Figure 6 shows a block diagram of the first implementation of System 7. Event matching and evaluation form part of what is known as System 8 at the Euratom Safeguards Directorate. Special working groups are currently working on the final specifications for this system. Event records are archived in relational database management systems (INGRES for NEGUS, ORACLE for other packages).

6. CONCLUSIONS

The design of integrated safeguards systems requires thorough knowledge of nuclear material flow and processing in the facility, of technical options for local verification measures and, above all, a clear concept for the use of the information. Interfaces have to be defined between local evaluation stations and safeguards sensor systems and between local evaluation systems and the local area network (LAN). It will be very difficult to safeguard large automated bulk handling facilities without these systems in place.

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SAFEGUARDS EQUIPMENT OF THE FUTURE
Integrated monitoring systems
and remote monitoring*

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Abstract
SAFEGUARDS EQUIPMENT OF THE FUTURE: INTEGRATED MONITORING SYSTEMS AND REMOTE MONITORING.

From the beginning, equipment to support IAEA safeguards could be characterized as that which is used to measure nuclear material, destructive assay and non-destructive assay, and that which is used to provide continuity of knowledge between inspection intervals, containment and surveillance (C/S). C/S equipment has often been considered in terms of cameras and seals, with a limited number of monitors being employed as they became available. In recent years, technology has advanced at an extremely rapid rate, and continues to do so. The traditional film cameras are being replaced by video equipment, and fibre optic and electronic seals have come into rather widespread use. Perhaps the most interesting aspect of this evolution, and that which indicates rather clearly the future trends, is the integration of video surveillance and electronic seals with a variety of monitors. This is demonstrated by safeguards systems installed in several nuclear facilities in France, Germany, Japan, the United Kingdom, the United States of America and elsewhere. The terminology of integrated monitoring systems (IMSs) has emerged, with the employment of network technology capable of interconnecting all desired elements in a very flexible manner. Also, the technology for transmission of a wide variety of information to off-site locations, termed remote monitoring, is in widespread industrial use, requiring very little adaptation for safeguards use. The paper examines the future of IMSs and remote monitoring in international safeguards, including technical and other related factors.

1. INTRODUCTION

Containment and surveillance (C/S) devices and non-destructive assay (NDA) tools are routinely used by the IAEA in the conduct of its safeguards activities. These technologies are being incorporated in unattended integrated monitoring systems (IMSs) that have the capability to collect and store information on the site or to

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provide remote transmission, either automatically or on command. This is demonstrated by safeguards systems which are installed in several nuclear facilities in France, Germany, Japan, the United Kingdom, the United States of America and elsewhere.

A key element of state-of-the-art integrated systems is a modular nodal system which accepts information from sensors and provides information to both an on-site data storage unit and a transmitter. The information from the sensors is processed within the nodal elements for authenticity as well as for sensor identification. Since the processing for authenticity occurs within the nodal elements, existing communication wiring (e.g. twisted pairs and coaxial cables) can be used. Intelligent nodes (nodes that can send command information to other nodes) allow information from one sensor to trigger another sensor.

There are many examples of everyday applications of remote monitoring: security sensors monitor homes and businesses; data from seismic stations are remotely transmitted; and land-mobile satellite communication systems send and receive test messages to and from mobile vehicles as well as determine the location of the vehicles. In general, these systems were not designed to provide authenticated messages from their location through the system centre to the customer’s communications centre; however, encryption techniques are readily available and can be employed to secure text information. Decryption could be performed at the customer’s communications centre, and the text message could not be intercepted by the system centre.

This paper examines the future of IMSs and of remote monitoring systems (RMSs) in international safeguards, including technical and other related factors.

These technologies are very important elements of the United States Department of Energy’s (USDOE) International Safeguards Programme [1]. A principal goal of this programme is to enhance verification techniques and capabilities of international, regional and bilateral regimes to support US non-proliferation objectives. The programme promotes technology exchanges in a broad range of technologies to ensure improvement in the effectiveness and efficiency of domestic and international safeguards, and international acceptance of new developments.

2. INTEGRATED MONITORING SYSTEMS

In the 1980s, under the US Programme of Technical Assistance to IAEA Safeguards (POTAS) and the USDOE International Safeguards Programme, Sandia National Laboratories (SNL) developed and successfully tested an IMS [2], which monitored the movement of spent fuel shipping casks to and from light water reactors and to storage facilities away from reactors. The system used a crane location monitor and gamma radiation detectors; it recorded the information in a tamper detecting, tamper resistant enclosure for later inspection, and triggered a camera for photo-
graphic assessment of anomalies. An outgrowth of this system is the channel monitoring system developed by SNL/LANL and used by Euratom at the thermal oxide reprocessing plant (THORP) in the UK.

A number of 'unattended monitoring systems' are currently in safeguards use or under development, some of which are the subjects of other papers in this symposium. These include unattended high level neutron coincidence counters, CANDU fuel bundle counters, unattended data loggers, the neutron-gamma unattended system, Consulha, GRAND, and the previously mentioned channel monitoring system.

It is important to note the classic difference between systems which simply trigger optical surveillance devices and systems which log all information (apart from that of the optical device) as well as trigger these devices. This latter dual function was in fact incorporated in the IMSs of the 1980s. From an engineering point of view, if any optical surveillance data are absent, information from detectors could lead to identification of an anomaly that could form the basis for investigation by an inspector.

2.1. Technical factors

In the past several years, under the USDOE International Safeguards Programme, SNL has developed a network based IMS. A new flexible technology is now available to design sensor and control networks based on a protocol embedded in an intelligent communications processor. The flexibility allows system designers and/or system installation personnel to make appropriate trade-offs between simplicity, function and cost in the design of network nodes and their installation. This is especially important in designing the installation scenario for a safeguards network. The network technology permits several choices of installations with the same basic node hardware, providing the ability to interconnect a number of different types of sensors and control devices in a simple network which can communicate over a single cable at a relatively low cost. Existing network technology meets these features by providing each node of the IMS with distributed intelligence capable of handling both the communication protocol and the data processing requirements of the individual sensors. Most sensors for safeguards and security systems seldom need to transmit messages since the sensors are not activated very frequently. The messages sent by the sensors are generally very small amounts of data consisting of a very few bits indicating situations such as 'alarm', 'tamper' or 'status ok'. Infrequent messages with low data content can be handled easily by the IMS, which can communicate over a number of different physical media, making it possible to configure different network configurations with interconnecting bridges. The same basic nodes can be configured to provide data collection in any number of different types of facilities and from networks varying in size from a very few sensors to a large number of sensors, including unattended NDA equipment, as well as a variety
of detectors from the physical protection area. A network can be installed using network management software and a computer, referred to as a network management tool, which offers full flexibility to change the network during installation. Such tools can provide different degrees of complexity, depending upon the safeguards applications and the number of changes that need to be made during installation.

The low cost integrated communications processor upon which the IMS is based contains all the software for the communications protocol and additional applications processing power to accommodate a large number of the safeguards sensor and control requirements.

Another microprocessor can also be added if still more processing power is required. An important feature of the IMS is its capability to authenticate all the data transfers over the network.

From this description it is clear that much care must be taken in the configuration of an IMS, and numerous adjustments will have to be made before full safeguards implementation in a particular facility will be possible. This is quite sensitive to facility configurations and environments. This task is not likely to be one that the IAEA should be expected to perform alone; rather, significant support will be required by system developers, facility operators and States.

2.2. Other related factors

The transition from conventional C/S and unattended NDA to IMSs raises issues of cost and facility/State acceptance. Important features of the IMS are both the low cost and the capability to install the same basic types of sensor nodes in many different facilities. The simple interconnection of all the sensors through one cable is very important for minimizing installation costs.

Acceptance will clearly depend on what benefits will be achieved by the IAEA and the State/facility operators. There will always be the question of "why additional measures?". The answer that will probably be found to be most acceptable is that the additional measure(s) will mean less inspector effort and presence. This in turn may require another look at the current safeguards procedures and criteria.

3. REMOTE MONITORING

Remote monitoring of nuclear facilities is not a new concept, nor is it a severe technical challenge. We need only recall the images presented daily on the television networks throughout the world. Also, as stated in the introduction, there are many examples of everyday applications of remote monitoring: security sensors monitor homes and businesses; data from seismic stations are remotely transmitted; and land-mobile satellite communication systems send and receive test messages to and from mobile vehicles as well as determine the location of the vehicles.
In 1978, the US Arms Control and Disarmament Agency (ACDA) developed and tested a secure system for remote verification of the status of C/S instruments employed at nuclear facilities by the IAEA; this system was called RECOVER (remote continual verification) [3]. As many will remember, this was at a time when the failure rate of the Minolta camera systems was quite high; it was, perhaps more important, at a time when many in the international community were simply not willing to accept the concept of transmitting safeguards data across national borders.

Shortly after the development of the RECOVER program, the Japanese developed the TRANSEAPER (transportation by sea verification) system as a potential safeguards measure.

More recently, the USA and Japan jointly developed the prototype containment and surveillance data authenticated communication (CASDAC) system [4]. This system was used for a feasibility test of remote monitoring of unattended sensors conducted by SNL and the Japan Atomic Energy Research Institute (JAERI) under a bilateral agreement between the ACDA and JAERI. The purpose of the system was to perform remote monitoring of the sensor status through the international telephone network; the system was a prototype developed by JAERI for nuclear safeguards and physical protection. The design was based on experience gained from RECOVER and TRANSEAPER.

Under the USDOE International Safeguards Programme, SNL is engaged in an active project termed ‘Remote Monitoring — System Integration and Field Trials’. The objectives of this project are: (1) to demonstrate that remote monitoring techniques can save inspection resources while at the same time maintaining or even strengthening the effectiveness of safeguards, and (2) to promote international acceptance of remote monitoring for safeguards applications.

The first field trial in this project is under way in Australia and is the subject of another paper in these Proceedings [5]. Future field trials are expected to be conducted at several facilities in Europe, North America and the Far East.

3.1. Technical factors

The existing technology enables IMSs to be interrogated via various communication links such as telephone, satellites, or radio-frequency transceivers. The data can be collected from storage devices on the network, or the individual sensors can be queried to determine their status. Data management, including presentation, will present a technical challenge when a large number of facilities are to be monitored.

The block diagram in Fig. 1 shows an example of the equipment that could be installed at facilities for remote monitoring systems. A network of nodes would collect data from a number of different sensors and security devices. Detection devices would be installed to complement each other for C/S applications. In addition, unattended NDA equipment, as well as simple gross attribute, yes/no, radiation detectors could be used.
FIG. 1. Remote monitoring system.
Referring again to Fig. 1, the authenticated item monitoring system (AIMS) [6] could be used to monitor drums or other storage containers. The AIMS sensor transmitters (ASTXs) can be easily attached to the items. The number of AIMS transmitters used would be determined by the items being monitored. Several infrared motion detectors with AIMS transmitters could be installed to detect motion in selected areas. Single ASTXs could be installed on other items of safeguards interest to detect their movements. The data from the AIMS devices can be collected in two different ways. A receiver processor unit could operate independently to collect AIMS information. An AIMS receiver node could also collect the same information for storage in the data logger and for remote interrogation.

Microwave motion detectors connected to the network could be used to determine if any activity is occurring in the area. Detection of any activity could trigger video recordings on the video recording module. The number of microwave motion detectors would be determined by the requirement of the selected area.

Ultra-wideband radar motion sensors could also be installed to detect activity. The pulses emitted from these sensors are well below one microwatt and are spread over several gigahertz. Their coverage consists of a spherical shell around the sensors that has an adjustable radius.

Other types of sensors such as photoelectric sensors could be used on the network, depending on the area to be monitored. Door switches, temperature sensors, radiation detectors, etc. could be connected to the nodes. Computer data interface devices with RS-232 data outputs or inputs can also be connected through the network.

Dual video systems could be utilized to collect video images. An analogue recording system, such as the portable surveillance unit, could be connected to the network. It could be programmed to record when certain sensors or combinations of sensors detect activity in the area under safeguards. A second video system, using digital compression technology and connected to a data logger/system controller, could collect digital images and store the images on removable data disks, or it could be accessed for remote transmission. The digital video recorder should also contain an internal hard drive for storing images. Data and images from the area under safeguards can be remotely accessed via telephone lines from a distant monitoring centre. Similarly, this information could be accessed via satellites where such a medium is required. From time to time, it would be necessary to remove data disks from the data logger and video tapes from the video recording modules.

3.2. Other related factors

One of the most important aspects of remote monitoring is the potential constraints related to the transmission of data out of a facility or beyond national borders. In connection with such transmission, a number of questions arise: must the data be encrypted?; in what form must the data be encrypted?; will the facility or
State require all information transmitted?, etc. In addition to transmission of data from an IMS beyond national borders, IMSs could be used for remote monitoring in large facility complexes and for transmission of data from facilities within a State. This latter application was, in fact, the principal objective of the German local verification (LOVER) project of the 1980s. Another important factor that must be considered is the overall cost effectiveness of remote monitoring. Here, very important issues must be addressed, for example whether acceptable safeguards assurances can be achieved through acquisition of data from remote monitoring and reduced inspection effort.

4. SUMMARY

This paper, as well as other papers in these Proceedings, clearly demonstrates the abundance of technology that supports the configuration of IMSs and RMSs. The technology presents a rather minimal challenge, except in the area of standardization. The situation with RMSs is further complicated by policy issues related to State rights, transparency, safeguards criteria, and other issues.

It is quite clear that much care must be taken in the configuration of an IMS, and numerous adjustments will have to be made before full safeguards implementation in a particular facility will be possible. This is quite sensitive to facility configurations and environments. This task is not likely to be one that the IAEA should be expected to perform alone; rather, significant support by system developers, facility operators and States will be required.

As the IAEA and the international safeguards community consider the current safeguards procedures and criteria and all that they mean in today's world, it is necessary to realize that technology can make significant contributions to the goals of safeguards. However, it is doubtful that these contributions can be realized to their maximum potential unless much more importance is placed on the qualitative parameters that could contribute to safeguards.

REFERENCES


CO-OPERATIVE ACTIVITIES OF THE AUSTRALIAN SAFEGUARDS OFFICE AND SANDIA NATIONAL LABORATORIES IN ADVANCED CONTAINMENT AND SURVEILLANCE TECHNOLOGIES

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Abstract

CO-OPERATIVE ACTIVITIES OF THE AUSTRALIAN SAFEGUARDS OFFICE AND SANDIA NATIONAL LABORATORIES IN ADVANCED CONTAINMENT AND SURVEILLANCE TECHNOLOGIES.

An agreement between the Australian Safeguards Office (ASO) and the United States Department of Energy was concluded to provide for co-operation in research and development activities for international safeguards. Pursuant to that agreement, the ASO, the Australian Nuclear Science and Technology Organisation (ANSTO) and Sandia National Laboratories are conducting collaborative work on the development of advanced containment and surveillance systems, including integrated safeguards systems. The paper describes the installation and the components of a remote monitoring system at ANSTO's Lucas Heights Research Laboratories in Australia. Over the next few years, this system will be operationally tested and evaluated to assist in the development of an international safeguards remote surveillance/monitoring concept. It is anticipated that this concept will provide one potential solution to IAEA inspection resource limitations, because remote monitoring technology has the potential to minimize the workload of the inspectorate while maintaining efficient and effective safeguards.
1. INTRODUCTION

An Agreement between the Australian Safeguards Office (ASO) and the United States Department of Energy (USDOE) was signed in early October 1992 in Washington, DC, to provide for co-operation in research and development activities for international safeguards.

Pursuant to the Agreement, the planned first co-operative activity, as specified in Action Sheet 1, entitled Advanced Containment and Surveillance, was concluded between the ASO, Sandia National Laboratories (SNL) and the Australian Nuclear Science and Technology Organisation (ANSTO). This activity had the following objective:

"... the joint development, testing, and/or evaluation of components for an Integrated Safeguards Verification System (ISVS) for the maintenance of the continuity of knowledge of nuclear material at remote sites, with the aim of improving the efficiency and effectiveness of International Atomic Energy Agency (IAEA) safeguards."

Both ASO and SNL have programmes relevant to advanced containment and surveillance (C/S), and the conclusion of the Agreement and the associated Action Sheet formalized a collaborative programme between the parties.

This paper gives details for both current and proposed joint activities.

2. CURRENT AND PROPOSED ASO, SNL AND ANSTO ACTIVITIES UNDER THE AGREEMENT

The conclusion of a collaborative programme under Action Sheet 1 allowed ASO and SNL to integrate and develop further the existing, advanced C/S activities. These are listed below:

(a) The ASO has, under its Australian Safeguards Assistance Programme (ASAP) for assistance to the IAEA, conducted and commissioned advanced C/S activities. These include: the vulnerability assessment of video authentication systems, the development of standards for IAEA optical surveillance and the development of a remote monitoring system.

(b) SNL activities, sponsored by the USDOE, range from the development and evaluation of C/S monitoring devices and systems, including integrated C/S systems and related work on video/digital image processing (data compression, authentication, encryption, etc.), to the development and evaluation of C/S based safeguards concepts for use in international safeguards.

(c) The collaborative activities between ASO/ANSTO and SNL provide for the installation and testing of a remote monitoring system (RMS) comprising SNL
developed sensors, encryption, authentication and data compression technologies. The work will be performed in two phases over the next several years:

**Phase 1.** The ASO and SNL will jointly assess and test the applicability of SNL developed sensor technology and authentication schemes for use in the Australian based RMS.

**Phase 2.** The ASO and SNL will jointly develop a hybrid RMS system, use it to perform an operational test at ANSTO’s Lucas Heights Research Laboratories (LHRL), assess and document the effectiveness and efficiency of the hybrid system and develop an international safeguards remote surveillance/monitoring concept based on these activities.

As part of the Phase 1 activities, SNL personnel conducted a pre-installation survey of several areas within the LHRL. This survey identified the dry spent fuel storage and the spent fuel pond areas as the key areas of interest from a safeguards perspective.

3. **REMOTE MONITORING OF THE DRY SPENT FUEL STORAGE FACILITY**

Following the pre-installation survey of the dry spent fuel storage and the spent fuel pond areas it was decided that, initially, effort should concentrate on the dry spent fuel store.

Consequently, an RMS was installed at the Dry Spent Fuel Storage Facility (DSFSF) located in ANSTO’s building 27 at Lucas Heights. Spent fuel from the High Flux Australian Reactor (HIFAR) is transferred to the DSFSF for long term storage following cooling in the spent fuel pond located elsewhere on the LHRL site.

In order to transfer spent fuel from the cooling pond, a spent fuel transfer flask having a weight of 8 t must be lifted by crane and positioned above the cooling pond, where ‘cropped’ spent fuel elements are transferred from the pond to the flask. This flask is then transported from the pond to the DSFSF, where another crane is used to position the flask above one of the spent fuel storage tubes located at the floor of the store. Following the transfer of the spent fuel from the flask to the storage tube, the flask is either returned to the cooling pond to collect additional spent fuel or parked in the DSFSF building.

A total of 50 spent fuel storage tubes are located at the DSFSF. Each tube can contain up to 22 cropped spent fuel elements and is secured by a substantial metal ‘plug’ which both seals the tube and provides shielding.

HIFAR is a 10 MW(th) research reactor, which is maintained and operated by ANSTO as a national neutron source.
4. FUNCTION AND BENEFITS OF UNATTENDED INTEGRATED SYSTEMS

The RMS was designed to test a number of different concepts that would be useful for unattended remote monitoring activities.

The function of any unattended system is to collect safeguards relevant information to verify the operations at a nuclear facility. The safeguards information collected is grouped under two general classes: NDA and C/S. Limitations in the available technology have in the past segregated the acquisition of these two classes of information. However, it is now possible to develop systems that can integrate NDA and C/S data collection and thereby develop alternative safeguards approaches that would have been either impossible or too costly to implement previously [1, 2].

Although IAEA safeguards approaches rely increasingly upon technical developments, they are still heavily dependent on human involvement. Specifically, the safeguards inspector and his physical presence at the facility remain critical to IAEA safeguards implementation.

Combining C/S and NDA technologies into integrated safeguards systems can save IAEA resources, automate safeguards measures and lower the requirement for inspector presence in the field. This system may obviate regular access for verification of the type of material and for determining nuclear material movement. It may also provide a more effective safeguarding of the nuclear material by the provision of more definitive information. This may be achieved, for example, by triggering, at the optimum time, for optical surveillance and at the same time verifying a material attribute.

One potential result from the use of unattended integrated systems could be the discontinuation of strategically less important IAEA inspection activities (e.g. attendance of inspectors at interim inspections of LWRs) and the concentration of resources on material and facilities of greater sensitivity and/or size.

Ideal situations are those where there is little expected movement or flow of nuclear material (e.g. in long term storage facilities), where nuclear material is accepted as being of low strategic value (and therefore with a long timeliness goal) and, in certain cases, where inspector access is limited and representative sampling (by inspectors) for NDA is not practical.

ANSTO's DSFSF is an ideal test site for an RMS. Since the spent fuel store is for long term storage, the nuclear material inventory is essentially static. Continuity of knowledge of this inventory is maintained by the application of IAEA seals to the tube and 'plug' assembly.

At present, IAEA inspectors must physically verify the integrity of the IAEA seals and thereby confirm that spent fuel has not been removed. Furthermore, operations involving the transfer of spent fuel to the DSFSF require the presence of IAEA inspectors so that they can witness the transfer of spent fuel and apply IAEA seals to the filled tubes at the conclusion of the transfer.
An objective of the RMS will be to demonstrate that safeguards relevant information (e.g. seal integrity data) may be obtained remotely and that, therefore, there is no longer a need for physical presence of an inspector.

5. DESCRIPTION OF THE REMOTE MONITORING SYSTEM

The RMS components were installed in the DSFSF for the first phase of the remote monitoring tests. A network of nodes collects data from a number of different sensors and devices, including the authenticated item monitoring system (AIMS), reusable in situ verifiable authenticated (RIVA) seals and motion detection devices. All of these components were integrated into a local operating network (LON) termed the local surveillance network (LSN), which uses commercial off-the-shelf LON software executed on conventional PCs. Essentially, these networked components provide a logically interconnected system.

5.1. Technical discussion

The RMS block diagram (Fig. 1) identifies the equipment installed in ANSTO's DSFSF and provides the relationship between the RMS components.

FIG. 1. Block diagram of the remote monitoring system.
Commercially available LON hardware and software was used to interconnect a network of devices which collect data from a number of different sensors and security devices. This network is termed an intelligent distributed control system, consisting of intelligent control devices or nodes which communicate with each other without the need for a central processor. A total of 32,385 nodes can be configured to a network. Each node within the network contains an embedded intelligence that implements a common communications protocol and performs control functions. In order to protect the information on the network, communications between the nodes are authenticated using proprietary algorithms.

A four-conductor network cable is used to interconnect all the nodes on the network. Two conductors are used to supply power to all nodes, while the other two conductors carry RS-485 network signals to all nodes. A number of different sensors and detection devices have been installed to study how they can be used to complement each other for C/S applications.

5.2. Local surveillance network sensors

The authenticated item monitoring system (AIMS) is used to monitor any activity that suggests attempted access to the spent fuel storage tubes. AIMS sensor transmitters (ASTXs) have been attached to the metal covers of the tubes with Velcro tape. The majority of the ASTXs contain motion sensors. Any motion of the ASTX or of the storage tube cover to which it is attached will generate an alarm signal which is transmitted over a radiofrequency (RF) carrier to the AIMS receiver processing unit. Several of the ASTXs, called RIVA seals, are electronic seals which detect the status of a fibre optic loop. The RIVA seals also transmit an alarm over an RF carrier whenever the fibre optic loop is opened. In this installation, 50 AIMS transmitters are attached to the storage tube covers. RIVA seals have been applied to the fuel transfer flask and to some spent fuel storage tubes. In addition, three (fibre optic) electronic seals developed by the Lawrence Livermore National Laboratory (LLNL) have also been applied to the fuel transfer flask and to two storage tubes.

In the current configuration, all data from the AIMS devices will be collected in two different ways. A redundant stand-alone receiver processor unit (RPU) is operated independently of the LSN to collect AIMS information. AIMS data can then be collected from this RPU by connecting a computer to it and transferring the data from the RPU to the computer.

Another RPU linked to the LSN, termed the AIMS RPU node, also collects the same information, which is stored in the RMS data computer. The RMS data computer may also provide the AIMS information for remote interrogation via the encryption modem to a remote monitoring station.

Microwave motion detectors on the network are also used to determine whether any activity is occurring in the area. The microwave motion detectors send
out RF pulses at approximately 10 GHz to detect motion in the area. Two microwave motion detectors are used to monitor the storage area.

Ultra-wideband radar motion sensors have been installed to test the operation of these detectors. The pulses emitted from these sensors are well below 1 µW and are spread over several gigahertz. Their coverage consists of a hemispherical shell around the sensor that has an adjustable radius of up to 7.3 m. In this installation, these sensors function as another type of motion detector. The data collected from these motion detectors will be compared with event data received from the microwave detectors and the three-dimensional interactive space (3-DIS) video motion detection system.

The 3-DIS video motion detector is a security based system, developed in Australia, which will be incorporated into the LSN. The 3-DIS is a software based video motion detection system that simultaneously analyses the video signals from two or more closed circuit television cameras and detects movement within multiple, predetermined three dimensional (3-D) spaces. The advantage of the 3-DIS over conventional two dimensional (2-D) video motion detection systems is its ability to alarm only when the protected 3-D space is violated, while ignoring innocent movement around the protected space or area. In this proposed system, a total of four cameras will be placed at the positions indicated for the radar sensors in Fig. 2. The 3-DIS system will be configured to protect a 3-D space around each of the spent fuel tube covers. As a precaution against camera masking, each of the four cameras will be protected, and the system can be configured to negate the effect of masking all but one camera. (Should only one camera be operational, the 3-DIS system reverts to a 2-D video motion detection system.)

FIG. 2. Layout of the sensors and equipment.
In order to confirm the functionality of all local surveillance system sensors, each of the sensors provide 'state of health' messages at pre-programmed intervals of time. Furthermore, all sensors provide additional information, including operational state and tamper alarms.

Other types of sensors such as photoelectric sensors could be easily added to the network in the future. Door switches, temperature sensors, radiation detectors, etc. can also be connected to the network. Furthermore, computer data interface devices with RS-232 data outputs or inputs can be connected through the network nodes. The data collected from the network would allow a comparison of the performance of the various types of sensors under the same set of conditions.

5.3. Video systems

In this system, one COHU 4810 series CCD camera provides a view of the spent fuel tubes (see Fig. 2) in the DSFSF. Detection of any alarm signals from the AIMS transmitters or the motion detectors will trigger video recordings to be made on two video recording systems connected to the network. Dual video systems are utilized to provide a measure of redundancy should one of the video systems fail. One system, an analogue recording system termed the video surveillance unit (VSU), is programmed to make still image recordings on magnetic tape at timed intervals and when it receives alarm signals indicating that there is activity in the storage area. The second video system, using a digital compression board in the RMS computer, collects digital images and stores these on the RMS hard drive and on a 'write once, read many' (WORM) optical disk on an optical drive that is also installed in the RMS computer.

Another feature of the system is the ability to 'lock out' images for a selected period of time following the initial alarm event. That is, should any sensor alarm, an initial image will be captured, and all other subsequent images triggered by other alarms within a specified period of time will not be stored. This form of front end data reduction will ensure that superfluous image data will not be stored in preference to safeguards significant image data. It should be noted that sensor alarm data will continue to be stored, even if the image data are being 'locked out'.

As noted above, the 3-DIS will be configured with four cameras, providing an overlapping field of view of the spent fuel tubes in the DSFSF. Initially, the 3-DIS system will only provide a trigger for image capture on the COHU CCD camera, pending an examination on how the 3-DIS system features may be more fully utilized and integrated with the existing system.

5.4. Remote transmission

All image data and the associated data collected by the LSN and stored on the RMS computer are available to the remote monitoring stations, located in Canberra
and Albuquerque, via the public switched telephone network. The LSN, installed in building 27 of ANSTO, is configured so that the remote stations must call the RMS computer to retrieve data and images. This mode of data transmission is considered to be more cost effective than a mode in which the system would individually transmit to remote stations triggered alarm event data as these are generated.

All data (including images) are data encryption standard (DES) encrypted before transmission. In order to achieve remote access to the RMS computer, correct encryption keys and passwords must be acknowledged by the systems before data transfer can be effected.

Periodically, on-site personnel from ASO or ANSTO will remove the optical data disk from the RMS computer and video tapes from the video recording module for analysis and comparison with the remotely collected data. The WORM optical disk will be used to test the 'mail-in concept' and will provide a full backup of all information collected by the LSN. The data on the WORM disks are protected against tampering, and each disk could be uniquely tagged with a reflective particle tag.

5.5. Equipment installation

Personnel of SNL and ANSTO installed the equipment in the storage area. As noted above, a total of 50 ASTXs, two LLNL electronic (fibre optic) seals and two RIVA seals were attached to the covers on the storage tubes. An additional LLNL electronic seal and a RIVA seal were affixed to the fuel transfer flask. The four radar motion sensors and two microwave motion detectors were attached to I-beams in the walls of the DSFSF. The network cable was strung along the walls and tie-wrapped to the existing building supports. An equipment rack was used to house the RPU, the RMS computer, modems, and the VSU. This rack contains the power supplies for the network as well as an uninterruptible power supply for the computer. The layout diagram (Fig. 2) shows the location of the sensors in the storage area of the DSFSF.

6. SUMMARY AND FUTURE WORK

While the application of international safeguards poses some practical difficulties, it is anticipated that this work will provide one potential solution to IAEA inspection resource limitations, because remote monitoring technology has the potential to minimize the inspectorate workload while maintaining efficient and effective international safeguards.

Regarding integrated systems, all relevant information provided by optical surveillance, NDA equipment, electronic seals and other C/S devices could be processed and remotely transmitted, providing to the IAEA more or less continuous
access to information about the inventory of nuclear material. Furthermore, it is conceivable that the flow of nuclear material, for example the loading and sealing of spent fuel flasks, can be verified by the use of such integrated remote monitoring systems.

It is envisaged that the RMS will be extended to include the spent fuel pond at the LHRL where both NDA and C/S devices will be incorporated, making this a truly integrated remote monitoring system.

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CONTINUOUS REMOTE UNATTENDED MONITORING FOR SAFEGUARDS DATA COLLECTION SYSTEMS*

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Abstract

CONTINUOUS REMOTE UNATTENDED MONITORING FOR SAFEGUARDS DATA COLLECTION SYSTEMS.

To meet increased inspection requirements, unattended and remote monitoring systems have been developed and installed in several large facilities to perform safeguards functions. These unattended monitoring systems are based on instruments originally developed for traditional safeguards and for the domestic nuclear industry to assay non-destructively nuclear materials. Through specialized measurement procedures, these instruments have been adapted so that they can be used as unattended monitors. The paper defines the parts of these un­attended monitoring systems, describes the systems that have been installed in the field and their status, and discusses future trends for unattended systems.

1. INTRODUCTION

The Los Alamos Safeguards Assay Group has been involved in the development and installation of several unattended monitoring systems in Europe, North America and Asia [1]. The need for these systems was prompted by the increased demands on the various inspectorates to inspect more facilities and to maintain around-the-clock inspections of operations in some of the newer facilities, but to do this increased amount of work without using additional inspectors.

2. UNATTENDED MONITORING SYSTEMS

Fully implemented systems could be composed of several parts (see Fig. 1) for data acquisition, collection, review and analysis, as well as for accountability, record keeping and report generation. The various parts can be thought of as representing

* Work supported by the Office of Arms Control and Nonproliferation of the United States Department of Energy, and the United States Programme of Technical Assistance to IAEA Safeguards (POTAS).
different levels, with the required data being passed on to the next highest level. A description of a complete generic, unattended system for monitoring radiation is given in the following sections. Actual systems may contain the complete generic system or only part of it.

2.1. Hardware components

Regarding the hardware, the first part of the system is the detector or detectors that sense the radiation. The detectors transform the radiation signal into signals that can be measured by data acquisition electronics (DAEs): GRANDs\(^1\) or JSR-11s\(^2\) for our applications, or systems based on VME bus extensions for instrumentation (VXI) for future applications. The DAEs supply the power for the detectors and process the signals from them. The DAEs are connected to Collect computers using a serial line. Normally, the higher level functions (review, analysis and accountability) are performed on another (other) computer(s) located away from the Collect computer; the data are transferred between the two computers via disk, but electronic

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\(^1\) D.S. Davidson Co., 19 Bernhard Road, North Haven, CT-04673, USA.
\(^2\) Canberra, Inc., Jomar Division, 110 Eastgate Drive, Los Alamos, NM-87545, USA.
transfer would be possible. Since remote and unattended monitoring systems must operate continuously, battery backup or uninterruptible power supplies, special racks, seals, visible cable runs and other hardware components may be necessary to provide authentication.

2.2. Data acquisition

Data acquisition is performed by the Monitor program, which resides in firmware in the DAEs. The Monitor program continuously collects the data at specified intervals, eliminates statistically insignificant data, temporarily stores the data, checks for tamper situations and dumps data upon request. A DAE, with its data storage capabilities and a Monitor program, is the only continuously operating system component required. Not all DAEs have the capability for a Monitor program, and in these systems the Monitor functions are performed by the software at the next level, i.e. Collect.

2.3. Collect

The software component Collect off-loads the data from the DAEs, organizes and stores the data, shows the inspector the current status of the systems, and is used by the inspector to copy the relevant data from hard disks to floppy or Bernoulli disks.

2.4. Review

The next software component, Review, organizes the data from several data collection systems into one database, allows the large amount of data to be quickly examined graphically and summarized, and selects subsets of data for transfer.

2.5. Analysis

Data analysis uses the packet produced by the data acquisition, collection and review system to perform the unique analysis needed for the particular system. Programs such as the high level neutron coincidence counter (HLNCC) of the IAEA or the neutron coincidence counter (NCC) at Los Alamos are used for analysis when it is necessary to calculate the amount of material present. Other analysis programs might be neural network programs used for pattern recognition and event occurrence.
2.6. Accountability

The data accountability system uses the results from analysis to update its total safeguards information database, draw conclusions and generate reports. This level is defined and implemented by the IAEA.

3. PRESENT SYSTEMS

Systems installed in the field can be categorized as non-destructive assay monitoring systems (NDAMS) or radiation monitoring systems (RMS). NDAMS are installed at facilities for which a quantitative special nuclear material (SNM) value must be obtained by the inspector. RMS are used when the inspector needs to know when a valid event has occurred; a quantitative measure of the amount of SNM is not required.

3.1. RMS at the Thermal Oxide Reprocessing Plant (THORP)

A GRAND electronics package connected to a pair of ionization chambers detects the presence and determines the direction of movement of spent fuel in containers at THORP in the United Kingdom [2]. Detectors are placed under water in the channel through which containers move to carry irradiated spent fuel into or out of a storage area. The GRAND interfaces to a camera system that records the IDs and the information supplied by GRAND.

3.2. NDAMS at the Plutonium Fuel Production Facility

At the Plutonium Fuel Production Facility (PFPF) of the Power Reactor and Nuclear Fuel Development Corporation (PNC) in Japan [3, 4], JSR-11 shift-register-based systems are installed at the input and output paths of a fuel fabrication plant to continuously monitor material moving into and out of the plant; assays are performed to determine the amount of material present. Other systems, installed at various locations along the process path, operate either continuously or unattended for shorter periods of time; some systems interface to camera systems that record sample IDs. The JSR-11, Collect computers and cameras are located in sealed cabinets near the detectors. Inspectors collect data from the systems every 30 days; the data are reviewed and analysed to determine the amount of material present.

3.3. RMS for core discharge monitoring

The core discharge monitoring (CDM) system at the Darlington facility in Canada monitors the radiation levels in the reactor containment shielding of a
CANDU reactor [5] to detect when irradiated fuel bundles are removed from the reactor. Each reactor has four detector enclosures placed inside the containment: two on each end of the reactor face on opposite sides of the containment room. Cables run through the penetrations to GRANDs located nearby. The two primary Collect computers are in a remote room several hundred feet away from the GRANDs, and each computer services eight GRANDs. A 'watchdog' processor controls which GRANDs are routed to which Collect computer and annunciates messages to the regional office so that problems can be relayed in near real time. Inspectors retrieve data every 90 days and use Review to determine when fuel moved.

3.4. RMS in fast reactors

Other RMS applications are the fuel flow monitors at PNC's Joyo and Monju reactors in Japan. Both reactors have a monitor at the input to the fresh fuel storage and another monitor at the input to the spent fuel storage. At Joyo, two other monitors detect movement of assemblies into and out of the reactor core and movement of fuel along a fuel transport corridor. At Monju, two other monitors detect radiation in the transport carts that move along the fuel flow path. Inspectors visit the reactor every 30 days and use the data to determine when fuel moved.

3.5. RMS for spent fuel

In another RMS application, GRAND based systems installed in the field monitor the movement of spent fuel assemblies into a storage pond to determine when fuel was moved.

3.6. NDAMS for the Siemens MOX facility

Significant work has been done on a second major NDAMS for the Siemens MOX facility [6], but the systems have not yet been installed. Detectors will be placed at the entry and exit of the two fuel pin storage areas to monitor the movement of trays into and out of the storage areas. JSR-12/Collect computer systems assay the material; the electronic-mechanical sensor (EMS) computer interfaced to a Siemens S5 controller reads the position of several sensors indicating tray position and tray ID. A serial interface between the Collect and EMS computers provides time synchronization. Software handles the time synchronization of the two computer systems and merges the NDA and EMS data. The data are reviewed and analysed to determine the amount of material present in the trays.
Forty-two unattended monitoring systems that collect data continuously are operating in the field or are ready for installation. An additional five systems that collect data are unattended for shorter periods of time. Of the 38 installed systems, 10 are NDAMS and 28 are RMS. All NDAMS are JSR based systems while the RMS

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<td>60 s</td>
<td>SC, SR</td>
<td>11-88</td>
</tr>
<tr>
<td>PFPF FPAS-B*</td>
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<td>60 s</td>
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<td>$^3\text{He}$</td>
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<td>20 s</td>
<td>SC, SR</td>
<td>3-91</td>
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<tr>
<td>Joyo ENGMB-B</td>
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<td>SC, SR</td>
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<td>Joyo CCRM</td>
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<td>60 s</td>
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<tr>
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<td>$^{10}\text{Be}$, 2 IC</td>
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<td>5 s</td>
<td>GM, GCS, GR</td>
<td>3-91</td>
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<tr>
<td>Monju ENGMA-A</td>
<td>$^3\text{He}$</td>
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<td>20 s</td>
<td>SC, SR</td>
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<td>$^3\text{He}$</td>
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<td>SC, SR</td>
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<td>GRAND</td>
<td>60 s</td>
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are a mixture of JSR and GRAND electronics based systems. The total operating years, as of March 1994, of the NDAMS are 45 years and those of the RMS are 77 years. During this time, the NDAMS have acquired and processed over 300 Mb of data and the RMS have handled approximately 9 Gb of data. The JSR based systems are all connected to $^3$He based coincidence type detectors. The GRAND based systems connect to a variety of detectors, such as FC, IC, NaI, $^3$He, $^{10}$B and Si. The count times on the various systems range from 60 s to 1 s.

This wide spectrum of systems is covered by two families of Collect/Review software. JSR based systems software consists of shift-register Collect (SC) and shift-register Review (SR); the JSR electronics units have no internal monitor capability. Parameters in programs handle the specifics for the different facilities; the basic programs are the same for all facilities. SC and SR are used with 19 systems that are installed at three different facilities. Additional Siemens requirements resulted in custom-tailored extensions of the basic shift-register software; Siemens Collect (SSC) and Siemens Review/Merge (SSR) fulfill the unique Siemens needs. The GRAND based systems software is made up of GRAND Monitor, Collect and Review. One version of the GRAND Monitor (GM) program handles all 24 systems at four different facilities. One Collect program handles all single-unit systems (GCS); the CDM systems are covered by the multi-unit Collect (GCM) provided by the Canadian Safeguards Support Programme. One GRAND Review (GR) covers all 24 GRAND systems. Table I summarizes these unattended monitoring systems.

### TABLE I (cont.)

<table>
<thead>
<tr>
<th>Facility/System</th>
<th>Detector(s)</th>
<th>DAE</th>
<th>Time</th>
<th>Software</th>
<th>Installation</th>
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<td>5 s</td>
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<td>1 s</td>
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<td>SFRC LOWER</td>
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<td>JSR-12</td>
<td>10 s</td>
<td>SSC, SSR</td>
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<td>JSR-12</td>
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<td>10 s</td>
<td>SSC, SSR</td>
<td>Not installed</td>
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</table>

* These systems operate unattended overnight only.

5. **FUTURE TRENDS**

We expect the demand for unattended measurement systems to increase; and for these systems additional requirements and features will be needed. We see the following trends developing as these types of systems evolve:
(a) Current systems measure only coincidence/gross neutron data, or gross gamma data, or both. Future systems will need to handle additional data types and combinations, and to automate scanning large samples in a continuous unattended manner.

(b) It must be possible to combine, with minimum effort, various detectors and different DAEs for use in new facilities. Commercial process control packages are being evaluated to see whether they could provide the required features with the reliability demanded by the long unattended periods, the flexibility to handle typical safeguards data and the ability to quickly examine data collected for 30–90 days. If commercial software is not adequate, then the existing software must be extended and made more flexible so that it meets the 'quick configuration' need.

(c) Capabilities are needed to integrate radiation (neutron, gamma) data and non-radiation (mechanical, video and operator) data into one system. Several different hardware solutions can be used to integrate systems. A more difficult problem is to define the software protocol and data structures to handle all the types of data that might appear in these systems. Interfaces between the data acquisition, collection and review, data analysis and data accountability systems must be defined.

(d) The amount of data collected by these systems is very large and, even with time saving features, such as data compression and graphical review, the time required to review the data for a 30–90 d period can be significant. New techniques such as neural networks can be used to automate part of the pattern recognition analysis now done by inspectors.

(e) Centralized data collection would make the systems easier to use by inspectors and would provide the possibility of real time data review and performance monitoring. This could be extended to Vienna for remote review. The hardware needed to integrate the system parts exists already. However, collecting data remotely from the acquisition systems introduces the problems that the data must be verified as actual data from the system and that all data from that system are transmitted. At present, authentication between different parts of the systems is handled by seals or by carrying data disks. New advances using the local operating network technology may provide authentication, together with options for network media such as radiofrequency and power line transmission that may significantly reduce installation effort.

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REFERENCES


PROJET DE CONFINEMENT-SURVEILLANCE ASSOCIANT MESURES NUCLEAIRES ET VIDEO SURVEILLANCE POUR L'AUTO-AUTHENTIFICATION DES DONNEES ET DES SCENES

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Abstract–Résumé

A CONTAINMENT/SURVEILLANCE CONCEPT COMBINING NUCLEAR MEASUREMENTS AND VIDEO SURVEILLANCE FOR SELF-AUTHENTICATION OF DATA AND VIDEO PICTURES.

The pools of the COGEMA reprocessing plant at La Hague are safeguarded jointly by the IAEA and Euratom. Initially it was decided to improve the verification of dry-unloaded assemblies in workshop T0 by developing CONSULHA. This device has been in routine use since February 1991. Subsequently it was planned to carry out verification of assemblies during unloading under water. The NPH2 workshop will be fitted with equipment appropriate for this configuration. The CEA/DAMRI/SAR team, in co-operation with the IAEA and Euratom, carried out tests of safeguards detectors in June 1991. An original feature of the NPH2 device is that the authentication system is incorporated in the hardware itself. The paper proposes a tamper resistance concept to supplement the existing systems. The device is designed on the basis of experience gained and will provide an efficient tool at a highly competitive cost. It is expected to be installed by the end of 1994.

PROJET DE CONFINEMENT-SURVEILLANCE ASSOCIANT MESURES NUCLEAIRES ET VIDEO SURVEILLANCE POUR L'AUTO-AUTHENTIFICATION DES DONNEES ET DES SCENES.

Les piscines de l'usine de retraitement de COGEMA La Hague sont contrôlées conjointement par l'AIEA et l'Euratom. Dans un premier temps, il a été décidé d'améliorer le

1. INTRODUCTION

Les piscines de l’usine de retraitement de COGEMA La Hague sont contrôlées conjointement par deux organismes internationaux, l’AIEA et l’Euratom. Ces contrôles ont été négociés entre la France et les organismes d’inspections. Cette négociation a donné lieu à un document de référence, le «Facility Attachment F2H1».

Dans le cadre du Programme français de soutien aux garanties, la France s’est engagée à mettre à la disposition des organismes internationaux les moyens les mieux adaptés aux contrôles.

Dans un premier temps, il a été décidé d’améliorer le contrôle des assemblages du déchargement à sec dans l’atelier T0, en développant un matériel et un logiciel d’acquisition et de dépouillement. Ceci a donné lieu à la réalisation du dispositif CONSULHA et du logiciel REVUE. Ceux-ci sont utilisés en routine d’inspection depuis février 1991. Ensuite, il a été prévu d’effectuer les contrôles des assemblages lors du déchargement sous eau. L’atelier NPH2 sera équipé d’un matériel adapté à cette configuration (Fig. 1).


2. CONTROLE DES DECHARGEMENTS SOUS EAU

2.1. Les essais

L’immersion des détecteurs a nécessité une préparation technique lourde. La réalisation d’enceintes étanches et la préparation des câbles devant être immergés demandent des soins particuliers pour ce type d’application. En immersion, le détecteur de neutrons doit présenter un bruit de fond faible et une insensibilité au rayonnement gamma.
Compte tenu des faibles doses neutron liées à la présence de l’eau, le détecteur le plus sensible aux radiations neutron disponible est la sonde CFUL01 (Philips). D’autres détecteurs neutron tels que $^3\text{He}$ et $\text{BF}_3$ ont été abandonnés en raison de leur sensibilité aux rayonnements gamma. Pour les mesures gamma, une sonde silicium (Eurorad) a été retenue.

Ces mesures sur le terrain ont montré les difficultés du travail sous eau. La préparation de l’immersion a demandé l’arrêt du fonctionnement de la piscine. La présence d’un pontonnier a été nécessaire durant toute la durée du travail. La présence des câbles pouvant se déplacer librement dans l’eau pose des problèmes à l’exploitant. Le positionnement des détecteurs sur un support lui-même disposé au fond de l’eau sur l’enquilleur n’a pas été facile.

L’opération de récupération des détecteurs et des enceintes étanches a nécessité la présence de deux opérateurs de la COGEMA en combinaisons spéciales avec leurs masques à gaz. La décontamination de tous les détecteurs a été nécessaire. La complexité de ces opérations impose de concevoir à NPH2 un dispositif immergé le plus simple possible. On cherchera à limiter la maintenance sur les détecteurs lors
de la conception du dispositif. Toutes les activités de déchargement de la piscine devront être arrêtées, ce qui ne pourra être effectué que lorsque le programme de la COGEMA le permettra.

La conjonction de ces contraintes aura des conséquences négatives sur la maintenance de l’équipement. L’efficacité du dispositif de contrôle en souffrira. C’est pourquoi nous prévoyons d’utiliser des détecteurs dont les hautes tensions et les seuils sont régulables à distance par le PC de commande. Les réglages de seuils et de haute tension sont déterminants pour optimiser l’efficacité du dispositif de détection. On comprend alors aisément que ce réglage serait impossible s’il fallait procéder à la sortie d’eau des détecteurs neutron de façon répétée.

2.2. Contraintes liées à la mise aux normes françaises de sûreté et de tenue aux séismes

L’exploitant s’est vu imposer par les organismes de sûreté nucléaire des modifications importantes sur les bâtiments et les équipements des piscines, pour
FIG. 3. Dispositif de codage des lignes vidéo.


Les équipements de transfert sous eau des éléments combustibles ont donc été modifiés. Une cage anti-chute entoure le combustible au cours du transfert entre le château de transport et le panier de stockage en piscine (Fig. 2). La présence de cette cage augmente la distance entre les détecteurs et le combustible. L'épaisseur de cette lame d'eau est déterminante pour l'efficacité de la détection des neutrons.

Pour augmenter le temps de comptage et diminuer l'épaisseur d'eau faisant écran, nous avons proposé de disposer les détecteurs sur cette «cage anti-chute», mais cette suggestion n'a pas pu être retenue en raison des câbles qui auraient perturbé le fonctionnement de la cage anti-chute. Les détecteurs seront donc disposés sur l'enquilleur de part et d'autre du passage du combustible (Fig. 1). L'éloignement des détecteurs du combustible a pour effet d'augmenter l'épaisseur de la lame d'eau.
Pour réduire celle-ci, il a alors été proposé de réaliser une boîte à air (Fig. 3) qui sera intégrée au bâti de la cage anti-chute.

Le schéma de la figure 2 précise la disposition des détecteurs par rapport aux combustibles PWR et BWR. Le sens de déplacement s'effectue du Nord vers le Sud. Le pont qui supporte la cage anti-chute est commandé manuellement. La boîte à air aura une hauteur de 400 mm. Les détecteurs ont été installés de part et d'autre de l'enquilleur, pour connaître le sens de déplacement des combustibles.

2.3. Dispositif nucléaire et vidéo

Le dispositif d’acquisition installé sera issu de l’expérience de CONSULHA. Après deux années de fonctionnement à T0, CONSULHA a enregistré les données relatives à plusieurs milliers d’éléments combustibles PWR et BWR. Les mesures ont été sauvegardées en même temps sur disque dur et sur papier enregistreur. La redondance de deux moyens de sauvegarde, numérique et analogique, indépendants l’un de l’autre, a prouvé son efficacité. Le programme de revue automatique des mesures a demandé de nombreuses études de mise au point. L’intérêt des inspecteurs pour ce genre de dispositif nous pousse à le perfectionner. M. F. Tola du CEA/DAMRI/SAR, présent durant le symposium, pourra vous faire une démonstration de son fonctionnement.

L’autonomie du dispositif CONSULHA est de 33 jours, entre deux visites des inspecteurs à La Hague. Elle sera étendue à trois mois pour le projet à NPH2. Une différence est à noter, elle concerne les programmes de revue automatique. En effet, ceux-ci dépendent de la géométrie du lieu où sont effectuées les mesures, ils seront donc propres à chaque site.

3. SYSTEME D’AUTHENTIFICATION

L’originalité du dispositif de NPH2 consiste aussi dans le futur système d’authentification. Un des principes présenté par G. Daniel lors du symposium ESARDA de Rome devrait être installé dès qu’auront été prouvées son efficacité et sa fiabilité.

L’authentification des données physiques et des scènes filmées est un point crucial du domaine du confinement-surveillance. Nous proposons ici un principe anti-fraude complémentaire aux systèmes existants.

3.1. Principe

Une ou plusieurs caméras vidéo filment des scènes dans lesquelles des objets apparaissent. Simultanément, un ou plusieurs détecteurs effectuent des mesures caractérisant ces objets.
A partir de ces éléments, il s'agit de véhiculer des données ou des signaux physiques par un ou plusieurs chemins (câbles de tous types, liaisons radio-électriques, téléphoniques, hertzziennes, analogiques ou digitales), et de les mélanger avec un second signal dont les lignes de transfert doivent être identifiées. L'authentification sera effectuée par comparaison entre les données émises et celles reçues à un ou plusieurs endroits de la chaîne. Une comparaison de ces données et une corrélation avec des lois physiques connues correspondant aux phénomènes observés permettra d'authentifier les scènes.

3.2. Applications

3.2.1. **Authentification au niveau du système d'acquisition et des données physiques (Fig. 3)**

Prenons le cas du système de confinement-surveillance utilisant deux caméras vidéo, un détecteur gamma et un détecteur neutron qui sera installé à NPH2.

Le principe et la réalisation consisteront donc à se servir des mesures neutron et gamma pour authentifier les images vidéo et les données nucléaires elles-mêmes. D'un côté, une ou plusieurs lignes de transmission vidéo relieront les deux caméras à des enregistreurs d'images (magnétoscopes couplés à un ordinateur ou à un système de stockage numérique par exemple). D'un autre côté, une ou plusieurs lignes de transmission relieront le ou les systèmes d'acquisition des mesures physiques à un poste de stockage et de traitement en vue de leur dépouillement. Enfin, une ou plusieurs lignes de transmission relieront le système de stockage et de traitement des données physiques aux caméras.

Les données physiques, horodatées, seront transmises depuis le système d'acquisition vers le poste de revue et de stockage et vers la ou les caméras vidéo. Au niveau de la ou des caméras vidéo, ces valeurs physiques seront codées et transmises dans le signal vidéo vers le système de revue et de stockage des images. Un boîtier spécialement conçu pour cette fonction extraira les valeurs transmises dans le signal vidéo et les retransmettra vers le poste de stockage et de traitement.

Ce dernier effectuera une comparaison entre les valeurs stockées en mémoire et celles reçues après extraction via le signal vidéo. Si les valeurs sont identiques, on pourra alors dire qu'il y a une très grande probabilité pour que les lignes et les valeurs transmises n'aient pas été falsifiées.

Pour authentifier à coup sûr la scène et les mesures physiques, une corrélation entre les mesures physiques et les images vidéo sera réalisée. Si l'histogramme des événements correspond au phénomène physique attendu et si on a une corrélation parfaite avec les images vidéo, on pourra alors affirmer que les valeurs, les lignes de transmission et les scènes sont authentifiées.
Si l’authentification donne un résultat laissant craindre un risque de falsification (valeurs des données différentes ou mauvaise corrélation avec la ou les lois physiques et le dépouillement des images vidéo), une alarme sera fournie.

3.2.2. Authentification au niveau du poste de revue des données vidéo

Une seconde possibilité serait de transmettre les valeurs physiques depuis le système d’acquisition via les caméras vidéo au système de revue et de stockage des données vidéo. Ce dernier ferait alors la comparaison entre les valeurs provenant du système d’acquisition et celles extraites et décodées venant via le signal vidéo. Le principe décrit précédemment serait alors appliqué pour effectuer l’authentification.

3.2.3. Authentification au niveau du système d’acquisition des données physiques avec inversion de sens

Une autre possibilité envisageable serait d’inverser le sens de circulation des données physiques. Il peut être possible d’émettre les données physiques depuis le système d’acquisition et de stockage et de les transmettre au système de revue et de stockage de données vidéo. Les données physiques seraient alors codées dans le signal de commande de la ou des caméras (commande à distance de la tourelle, de mise au point ou d’ouverture du diaphragme, etc.). Au niveau des caméras, les valeurs physiques, extraites et décodées puis transmises au système d’acquisition des données physiques où les données d’origine auraient été préalablement stockées. La comparaison serait alors effectuée, puis l’authentification.

3.2.4. Autres possibilités

L’exploitation des données physiques de la radioactivité de sources ou d’éléments combustibles irradiés peut donner lieu à l’exploitation de la distribution aléatoire du nombre de désintégrations (suivant le nombre de désintégrations, la distribution des comptages répondra à une loi de Poisson, à une distribution binomiale ou à une loi de Gauss).

Il serait aussi possible d’utiliser:

— Un seuillage en énergie (exemple: rapport d’environ 4 entre un seuil à 50 keV et 600 keV pour le $^{137}$Cs) et de vérifier aléatoirement ou périodiquement ce rapport.
— Une sélection d’énergie pour un radioélément déterminé et de vérifier sa période.
— Les caractéristiques des éléments irradiés comme l’émission neutronique en fonction du taux de combustion ou la décroissance gamma en fonction du temps de refroidissement, etc.
D’autres éléments physiques de l’environnement peuvent être exploités sur le même principe: température, intensité lumineuse, mesures de distance, ultrasons, infrarouges, radar, analyses de spectrométrie, pesées, pH, mesures physico-chimiques, etc.

3.3. Choix préliminaire pour NPH2

Dans le cas de NPH2, la première solution retenue pour les essais du système d’authentification sera la solution avec authentification au niveau du système d’acquisition des données physiques.

4. CONCLUSION

MODULAR DESIGN OF INTEGRATED DIGITAL NETWORKED DEVICES FOR USE IN COUNTER-PROLIFERATION ACTIVITIES

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Abstract

MODULAR DESIGN OF INTEGRATED DIGITAL NETWORKED DEVICES FOR USE IN COUNTER-PROLIFERATION ACTIVITIES.

The scope of counter-proliferation activities will expand greatly over the next several years. Credible monitoring to prevent proliferation will require a very significant increase in the capacity to gather, analyse and present information. The necessary instrumentation includes various devices as well as the use of seals, non-destructive assay and surveillance. The resulting data must be concentrated, collated and stored so that they can be reviewed if discrepancies are found in future analysis. To develop the required infrastructure, some fundamental precepts will have to be adopted: The majority of the technology and devices used must be commercially available, and installations must be designed for modularity and integration into an evolving data management system. Sensors will have to be developed that can be used for specific needs; they must be modular, with as much commonality of technology and implementation as possible; and they must be network ready, i.e. the data generated by the sensor must be easily integrated into an overall information handling system. Of great concern is the assurance of data integrity. The paper discusses this problem and presents a number of technologies that can be used.

1. INTRODUCTION

It can be safely predicted that the scope of counter-proliferation activities will expand greatly over the next several years. The range of critical material and technology is being augmented (e.g. inclusion of chemical and biological facilities), and increased activities in dismantling of weapons require monitoring of processes and storage, both for purposes of direct verification and for ensuring that there is no diversion of material. Credible monitoring to prevent proliferation will require a very significant increase in the capacity to gather, analyse and present information.

The information flow structure is dendritic, as illustrated in Fig. 1.
As the number of facilities grows geometrically, the increasing volume of information will clog the upper level channels, bringing the entire process into a state of gridlock.

Compounding the problem is the fact that the entire data collection and validation process is human intensive, with no economics of scale. Even though much of the equipment acquires data automatically, the data must be collected from the equipment, transported, verified, validated and analysed by trained persons. There is nothing in the process that would permit inspectors to handle a greater workload as the number of installations increases. Consequently, payroll, travel and training budgets must be anticipated to grow at the same rate as the number of installations. This implies a hundredfold increase in those budgets over the next seven years — a highly unlikely event.

Finally, the increase in the number of monitored facilities implies a concomitant increase in the number of monitoring instruments. These instruments are currently unique custom items, and even a hundredfold increase in volume would not introduce commercial economies of scale in production. Consequently, it is certain that equipment and maintenance budgets will increase by two orders of magnitude if the current instrumentation technology is to be retained. Therefore, it is unmistakably clear from simple budgetary considerations that the current practices of instrumentation and information flow cannot satisfy the demand for the increasing scope of counter-proliferation activities. In fact, the demand for increasing monitoring and surveillance over the next seven years can be met only if the following requirements are fulfilled:

- Equipment must be constructed entirely of commercial off-the-shelf (COTS) assemblies in order to lever economies of scale in equipment acquisition and maintenance budgets.
- Information must be acquired, collected, transported, validated and analysed without human intervention, in order to induce economies of scale in the manpower budgets.
The existing COTS technology meets both requirements simultaneously, provided only that the requirements are combined: *The equipment and information infrastructure shall be a network with all-digital data from the point of origin through transport and final use.*

We review in this paper the data acquisition, collation and presentation activities necessary to address the anticipated increase in verification, in the context of current technology. The basic thesis is that the technical requirements can be met with the existing technology. A derivative conclusion is that, without absolute concentration on using standard, commercially available technologies, it will not be possible to handle the anticipated increase in data acquisition and analysis. Implicit in adopting standard technologies is that the overall data management resources be implemented as modular, logical pieces. Such an approach minimizes the differences between applications, lowering the costs of development, deployment and maintenance. Constructing the solution of modular pieces improves the potential for distributing the system across a network. This can mean connecting multiple instruments within a facility to expedite inspector visits, or concentrating information on a regional or worldwide basis for better analysis. In the latter case, use of networks can offer the obvious benefit of reduced travel for inspectors (who would perform analyses in regional centres). Less obvious, but of equal importance, is the potential to greatly improve the quality and timeliness of analysis. Bringing data to central locations allows automatic, real time analysis, leading to immediate recognition of significant events. These can then be followed up (e.g. by local visits), thus maximizing the probability of correctly identifying a significant event. In addition, networking instrumentation will be critical for such applications as tracking material during transport, and correlating data, acquired over time and geographical separation, relevant to a single set of material.

![Diagram](data_services.png)

**FIG. 2.** Data services for implementing counter-proliferation and verification tasks.
Data services for implementing counter-proliferation and verification tasks can be viewed at three levels, as shown in Fig. 2. In current systems the connection between these levels is usually accomplished by inspectors who use written records. Analysis is frequently done without the benefit of any modern data reduction or presentation tools. In order to meet immediate and future requirements of verification, a greater degree of integration and automation must be brought to bear. It is clear that, at least in the areas of data analysis/presentation and data transmission for system integration, industrial standard solutions are available. The following sections will present specific examples and indications of the kinds of solutions that are available and/or emerging. In the area of sensors there will be some need to address specific requirements (e.g. security) and configurations (e.g. gamma ray detectors) for which special development work will always be required. Current technology allows the design goal of this kind of development to be almost exclusively an integration of commercial components.

There are three areas that will be explored with respect to existing technology:

- Instrumentation (specifically some guidelines for development);
- Data security (with respect to both privacy and authentication);
- Networking infrastructure (available and emerging technologies that are useful for verification activities).

Underlying all of these is that the single enabling technology is the capability of digital data processing techniques. The fundamental point is that digital technology is being universally adopted as the basis for all data processing and transmission requirements. The incredible explosion of digital data processing capability is based upon the enormous economy of scale, since digital technology is inherently flexible, comprising generic components that can be focused on specific requirements. This allows the development costs to be absorbed by very diverse markets, which fuels competition and lowers costs. Once it is accepted that all instrumentation correlates and emits digitally encoded data, it becomes possible to concentrate on integrating existing technologies (almost always COTS).

2. INSTRUMENTATION

The necessary instrumentation includes all kinds of devices, including seals, non-destructive assay (NDA) and surveillance. Some fundamental precepts for evaluating data acquisition solutions are:

(a) The majority of the technology and devices used must be commercially available;
(b) Installations must be designed for modularity and integration into an evolving data management system.
In some cases, sensors will have to be developed to address specific needs, i.e.

(a) They must be modular, with as much commonality of technology and implementation as possible;
(b) They must be network ready, i.e. the data generated by the sensor must be easily integrated into an overall information handling system.

Using modular design techniques, with a mandate that they integrate into an extended data handling system, instruments developed specifically for verification will have improved efficiency and effectiveness through:

- Reducing the costs of initial development and certification;
- Lowering the time required by the inspectors to monitor targeted materials and facilities;
- Reducing maintenance and training costs.

In evaluating these points, it is important to consider the cost/performance improvements in digital technology. As recently as five years ago, it would not have been credible to assert that common components and standards could be used to address sensor needs. Confirmation that this has changed can be found in a diverse set of industrial applications. Examples include the adoption of standard microprocessors within automobiles for controlling many aspects of operation, from braking (antilock braking systems) to interior climate control. With increasing frequency, industrial process control applications (e.g. glass forming and steel manufacturing)\(^1\) are using standard, commercial processors and sensors to implement their specific control algorithms. Note that there is a very great diversity in these applications; however, all of them can be implemented using a single class of commercially available components. A specific example from within the verification community is the Gemini family [1] of surveillance cameras developed for the CEC and the IAEA. Gemini (illustrated in Fig. 3) is an all-digital, network-ready surveillance instrument constructed almost entirely of standard components. The digital camera, image storage and system control are all commercial products. The only development specific for Gemini is the power management system, which has to meet the unusual isolation conditions found in remote surveillance sites. Gemini demonstrates two attributes of our thesis: (1) using commercially available parts, in a modular design, greatly reduces the development time and cost while it increases the flexibility; and (2) there should be network access to broadcast significant events or to enable general review. The development cycle of Gemini is distinctly different from that of its predecessors. For the initial development of the modular integrated video system (MIVS), three years were needed; an additional two years were needed for

\(^1\) COMSOC, implemented and manufactured by Aeon Systems, Inc., for Owens Brockway, IL, USA.
problem resolution during the first manufacture of the system. In contrast, a prototype Gemini was delivered to the CEC six months after discussions of the initial requirements. On the basis of experience gained during evaluation of this prototype equipment, initial units have been procured for delivery in six months. Since the system is modular and based upon standard components, there is strong expectation that evolutionary improvements of the present design can be made. A design study is anticipated for 1994, with results to become available late in the year.

3. SECURITY

Of great concern is the assurance of data integrity, i.e. security and authentication when data are transmitted over channels that are not under the direct control of the IAEA. There are two functions of data security:

(a) Privacy — the assurance that only appropriate individuals/agencies can examine the data;
(b) **Authentication** — confirmation that the data come from a specific, trusted source (e.g. a sensor).

Digital encoding increases the opportunities for implementing both aspects of security by encouraging the development of mathematical techniques to achieve the desired function. Improvements in the performance of digital processors now allow these techniques to be implemented in a standard and cost effective manner. Security requirements are not unique to the safeguards community; governmental, financial and general commercial users of networks have made considerable efforts to develop solutions for all aspects of data security. Traditional cryptographic techniques, e.g. data encryption standards (DES), have been used for some time by the safeguards community to ensure data privacy. While these techniques are appropriate for this purpose, they do not readily provide authentication and they are inherently difficult to manage. The problem with standard encryption techniques is that they are symmetric, i.e. there is a single key, used both to encrypt and to decrypt the information. If the key is compromised, so are all data encrypted with it. Since data transmission implies a sender and a receiver, it is intrinsically necessary for the key to be distributed to both. This makes it necessary to have a secure data channel for distribution or to secure storage for the key (or both).

For the past decade, many users have augmented their protocols with asymmetric encryption methods, which address both authentication and the secure distribution of cryptographic (symmetric) keys. Asymmetric encryption uses two keys (usually called the public key and the private key), which work in conjunction with one another. Information encrypted with one key can only be decrypted with the other key. Implicit in this technique is that, if one key is given, it is not possible to derive the other key. This is the basis for describing one of the keys as public; this key may be generally known, and yet information encrypted with it is only available to the holder of the (matching) private key. Implementations based upon the RSA encryption technique [2] are available and have been widely accepted by governmental and commercial authorities. (The RSA encryption and authentication methods are also described in a Poster Presentation [3].) Public/private key pairs can be generated in any quantity, allowing a unique set to be associated with every sensor.² This

allows complete assurance of the authenticity of information from a sensor, without requiring continuous security of data channels. In addition, RSA encryption can be used to provide secure key distribution for a standard, symmetric technique (e.g. DES). This is advantageous, since the RSA technique is considerably more computer intensive than (for instance) DES; thus, utilizing a combination of the two techniques can decrease implementation costs.

4. NETWORK TECHNOLOGY

The technology for distributing information has increased over the past decade, to the point that the digital networking infrastructure is becoming totally commonplace. Over the past decade, the foundations have been laid, in terms of basic technology and the necessary protocols to handle large, distributed systems. The fruits of this work are becoming apparent in the general availability of local (e.g. within a facility or a campus), regional and international connections. While there is general familiarity with networks for general information processing (e.g. personal computers and workstations), more specialized applications for data acquisition and industrial control have also been adopting distributed technology. This can be immediately adapted to NDA and surveillance systems for data concentration within facilities. There are ongoing efforts to reduce the single highest cost involved in first implementing a network: installing the physical transmission media. This is done by introducing such methods as infrared signalling and using telephone lines. Also of interest are devices that use existing power lines for digital signalling. This technology has been developed for more than a decade; recently, the signalling needs have been combined with the necessary logical protocols within dedicated integrated circuits. One example is the local operating network (LON) family of components from Echelon. These technologies or similar ones are likely to become ubiquitous, driven by manufacturers who are implementing ‘smart appliances’.

Cellular telephone technology allows rapid creation of local and mobile networks on a regional basis. This will soon be coupled with satellite communications to extend digital connectivity on a low cost, worldwide basis. (Currently, there exists moderate cost, portable commercial equipment that enables worldwide access to national telecommunications via satellite.) This technology suggests that it will be possible to track individual shipments of significant material on a real time (or an almost real time) basis, using standard commercial data transmission infrastructure. It is concluded that the assumption of an infrastructure for digital data transmission, on a facility basis, a regional basis and a worldwide basis, is a valid design premise.

3 Echelon Corporation, 4015 Miranda Avenue, Palo Alto, CA 94304, USA.
One of the most important services that the integration of sensors will provide is the (necessary) capacity to track material over broad geographical areas and over extended periods of time. This is absolutely essential in order to ensure that portions of materials undergoing multiple processing steps are not diverted. By combining all of the acquired information in a single database, sophisticated analysis can be performed on an ongoing and completely automated basis. Further, the system can evolve; as techniques are certified, they can be rapidly deployed across a diverse set of verification instances.

Modular development allows flexible implementation, and new facilities can combine the existing instrumentation with new capabilities in a confident way and with low cost. The scope of counter-proliferation efforts will increase, and at the same time it will be necessary to improve effectiveness. Increased data collection is required for these efforts, which implies exponential expansion of the data acquisition rates and repositories. These goals can only be accomplished through determined adoption of integrated systems, with major dependence on commercially available components. The good news is that basic components are available and that additional capabilities become available almost daily. The great majority of technologies that can be applied for counter-proliferation purposes is not (or will not be in the very near future) on the 'leading edge'. Therefore, it is important to appreciate the continuum of technology maturity. True 'leading edge' efforts reflect conceptual development in laboratories. Technologies referred to in this paper as 'leading edge' technologies or as emerging ones are those which are being transmitted to standard commercial use. Introduction into the commercial market implies not only technical robustness but also a strong commitment by one or more vendors to support the products over time. It is absolutely critical that technologies for verification purposes be chosen from those which are available from the general market. The range of technologies necessary to deal with verification is far too extensive to allow agencies to become involved in futuristic developments. Further, as the examples cited in this paper demonstrate, there is simply no need to undertake such activities. The requirements of almost all portions of the verification process can be met with standard, commercial components. Those few components (primarily sensors and seals) that do need some specific configurations can be constructed using industry standards and commercial components.

Following the prescription to rigorously utilize commercially available products and technologies allows responsible agencies to concentrate on system definition and analysis and not on technology development. With such a policy, it will be quite feasible to deal with the increased scope of the counter-proliferation and verification mandate, and even to improve the depth of coverage.

5. CONCLUSIONS
REFERENCES


TECHNOLOGY FEATURES OF A NETWORK TECHNOLOGY FOR SAFEGUARDS APPLICATIONS*

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1. INTRODUCTION

A new flexible technology is now available to design sensor and control networks based on a protocol embedded in an intelligent communications processor. The flexibility allows a system designer and/or a technical installer to make appropriate trade-offs among simplicity, functionality and cost in the design of network nodes and their installation. This is especially important in designing an installation scenario for the safeguards network. The network technology permits several choices of installations with the same basic node hardware. A pre-installed network offers maximum simplicity and no flexibility, since it will operate as programmed during manufacture, or during the pre-installation set-up and check-out. At the other end of the spectrum, a network can be installed using network management software and a computer. The combination of the network management software and computer hardware is generally referred to as a network management tool (NMT). The NMT option offers full flexibility to change the network during or after installation. Different NMTs can provide different degrees of complexity, depending upon the applications and the amount of changes that need to be made during installation.

The low cost integrated communications processor upon which the integrated monitoring system (IMS) is based contains all the software for the communications protocol and additional applications processing power to handle most of the safeguards sensor and control requirements. Another microprocessor can be added if more processing power is required. An important feature of the IMS is the capability to authenticate all the data transfers over the network.

* This work was supported by the United States Department of Energy under Contract No. DE-AC04-94AL85000.
The network uses a carrier sense multiple access (CSMA) contention protocol to handle data. Each communications processor at a node will decide when it wants to transmit a message. The node preparing to transmit will first determine that there is no message traffic on the network before attempting to transmit. If a collision does occur with some other node, then the nodes will back off and attempt to transmit at a later time. The back-off delay time is random at each node, thereby allowing one of the nodes to transmit before the other node attempts to transmit again.

2. HARDWARE

The basic hardware shown in Fig. 1 is one of many possible node designs. This design for the IMS uses dual processors and still produces a low cost node for field applications. It provides interfaces for accepting computer data through an RS-232 port. Bi-level inputs and outputs are provided to interface to a number of different sensors and control devices. An analogue to digital converter (ADC) provides the capability to accept analogue signals and to process them before transmitting the value to another node.

![Node block diagram for the IMS.](image-url)
FIG. 2. Flexible network configurations.

FIG. 3. Integrated monitoring system.
3. NETWORK CONFIGURATIONS/REMOTE MONITORING

The flexibility of the hardware permits the design of simple networks of a few sensors or complex networks of many sensors. Figure 2 shows a very simple network and a more complex configuration that can be assembled from nodes. Network bridges can be used to interconnect networks of different physical media. Bridges can be used to interconnect networks in different buildings or areas, or to extend the transmission range of networks in one area. Routers can be used to pass specific messages between two or more different networks.

4. AN IMS CONFIGURATION

The block diagram in Fig. 3 shows one possible configuration of IMS equipment that could be installed at facilities to collect data from a number of different sensors and security devices. A number of different detection devices can be used to complement each other for containment and surveillance (C/S) applications.

5. INTEGRATION OF SENSORS FOR NETWORK OPERATION

Almost any type of sensor can be adapted for operation on the network. Bi-level and other types of digital data output can be directly interfaced to the network integrated circuit. ADCs can be interfaced to the network integrated circuit so that the node can handle analogue signals from sensors by converting voltages to digital values. Since the network node is an intelligent processor, the ADC output can be processed before sending the data over the network.

5.1. Authenticated item monitoring system

The authenticated item monitoring system (AIMS) can be used to monitor drums or other storage containers. AIMS sensor transmitters (ASTX) can be attached to the containers. All the data from the AIMS devices are collected through a receiver processor unit (RPU). The AIMS receiver node collects the information and sends it over the network for storage at a remote location.

5.2. Microwave detectors

Microwave motion detectors attached to the network can be used to determine if any activity is occurring in an area. Detection of any activity triggers video recordings. The occurrence of motion detection is stored and transmitted over the network to remote locations for storage.
5.3. Radar sensors

Ultra-wideband radar motion sensors have been used on the network. The pulses emitted from these sensors are well below one microwatt and are spread over several gigahertz. Their coverage consists of a spherical shell around the sensor that has an adjustable radius. Their function is different from that of other types of motion detectors and they can be used to complement microwave detectors.

5.4. Other sensors and node control devices

Other types of sensors, such as photoelectric sensors, have been used on the network, depending upon what is to be monitored. Door switches, temperature sensors, radiation detectors, etc. can be connected to nodes. Computer data interface devices with RS-232 data outputs or inputs can also be connected through the network nodes. The flexible nature of the network permits more types of sensors to be installed in order to provide both redundant and complementary operation for increased reliability. Control signals can be sent over the network to turn on lights or to operate electromechanical devices.

5.5. Video systems

Multiple video systems can be utilized with the network to collect video images. An analogue recording system, such as the portable surveillance unit (PSU), can be connected to the network. The PSU can be programmed to make recordings when certain sensors or combinations of sensors detect activity in the area under surveillance. A digital video system using digital compression technology can be connected to the network and receive the same trigger signals. The digital video system can collect digital images to be stored on removable data disks or accessed for remote transmission.

6. NEW CONCEPTS

The network flexibility coupled with a wide range of sensors offers the capability to develop new concepts for safeguards applications. It is now possible to install video systems with front-end triggering devices other than video motion detection. A suite of other motion sensors can provide very reliable triggers to record digital video images and thereby reduce the number of images that an inspector has to review. Furthermore, the images can be directly linked to the sensors that caused the image to be taken. Another concept to be explored is the integration of C/S safeguards applications with the collection of information from NDA instruments. The capability to combine all safeguards relevant data into one database would greatly decrease inspector review time and improve overall inspection efficiency.
EXPERIENCE WITH AN AUTOMATIC NEUTRON/GAMMA DATA ACQUISITION SYSTEM

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1. SYSTEM

An automated NDA measurement system has been installed in a plant which automatically handles containers of plutonium oxide. There are four identical combined neutron/gamma counters, which are mounted around all the exits and entrances of the plutonium store [1]. They provide a measurement of all containers during standard movements and also serve as monitoring devices to indicate any other plutonium movements. The neutron detection cavities are open ended cylinders through which the containers enter or leave the store. Because of space constraints, the length of the cavity is about half of the length of a standard container and so the measurement of each container is made in two parts (two 5 min measurements). The gamma detectors are of the conventional low energy germanium (LEGe) type, but the gamma spectrum has to be acquired through the thickness of the stainless steel container (16 mm). The data acquisition system (DAS) consists of ‘blind’ PCs [2] which monitor the data for the presence of containers or anomalous events [3, 4]. The data are automatically transferred over a fibre optic link to a central host system [5] in the Euratom inspectors’ office. This host system is used for data storage, review and analysis. A typical session for a user consists of:

- Entering the declared data (name, mass, isotopics, dates);
- Checking the time of occurrence and the shape of the neutron count rate profile (the shape of the profile is characteristic of the container movement and also shows whether the container was entering or leaving the store);
- Comparing the declared isotopics with the measured ones;
- Calculating the Pu mass using the operator isotopics and/or the measured isotopics;
- Comparing the measured Pu mass with the declared Pu mass.

All these operations are done using a graphical user interface in a DEC Windows environment. The system is in routine use.
2. CALIBRATION

The calibration of the detectors was done with containers selected to give as wide a mass range as possible. These containers were measured with a universal fast breeder counter (UFBC) with a known calibration curve. The UFBC measurements confirmed the declared values, which were then used to calibrate the present system. Separate count rates are obtained from each half of the container. Different ways of treating these data were investigated. It was found that a reasonably linear calibration could be achieved by taking the average raw Totals and Reals count rates of the two measurements and applying the standard corrections to them (deadtime, background, normalization, multiplication correction). The errors on the fit are relatively large (giving $\sigma \sim 1.6\%$), mainly owing to the inclusion of the low mass range. Low mass containers represent a rather small fraction of the population, and the historical data show much better performance than this calibration curve error would suggest.

3. RESULTS

3.1. Gamma performance

A set of measured isotopics and their uncertainties produced by a multigroup analysis (MGA) run on a typical gamma spectrum from fresh material (using the standard 'new' $^{242}\text{Pu}$ correlation of MGA) is shown below.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Weight %</th>
<th>Error (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-238</td>
<td>1.846</td>
<td>3.37</td>
</tr>
<tr>
<td>Pu-239</td>
<td>54.947</td>
<td>2.71</td>
</tr>
<tr>
<td>Pu-240</td>
<td>27.663</td>
<td>3.74</td>
</tr>
<tr>
<td>Pu-241</td>
<td>10.082</td>
<td>3.42</td>
</tr>
<tr>
<td>Pu-242</td>
<td>5.46</td>
<td></td>
</tr>
<tr>
<td>Am-241</td>
<td>-0.050</td>
<td>25</td>
</tr>
</tbody>
</table>

The results are reasonable, taking into account the thickness of stainless steel of the container and the measurement time.
3.2. Neutron detector stability

The performance of the detector is checked by small $^{252}$Cf sources that are permanently installed in the detectors. This Reals 'background' count rate is small and does not affect the precision of the measurement. These sources provide a continuous check on the correct operation of the detectors. The 'background' count rate as a function of time for each detector for the period September 1992 to August 1993 is shown in Fig. 1. The observed variation is consistent with the statistical error on the individual measurements (0.3%).

3.3. Plutonium mass results

Figure 2 shows the Pu mass results (measured–declared), given as a percentage deviation, for data collected during 12 months, calculated using both declared isotopics and measured isotopics.

The overall bias is $-0.06\%$ when declared isotopics are used and $1.5\%$ when measured isotopics are used. The standard deviation of the results for declared isotopics is $0.71\%$ and that for measured isotopics is $4.0\%$. The measurements shown consist typically of batches of ten containers with a single isotopic composition. There is some indication of systematic differences between different batches. This may be due to the way in which the data are treated (as described above), which stretches the assumptions in the derivation of the multiplication correction technique. However, the effect is relatively small.

![Graph showing Pu mass results](image-url)
4. PROBLEMS AND POSSIBLE IMPROVEMENTS

4.1. Shift register problem

Before the VAX system was installed, a method of unattended data acquisition based on data loggers was used. The JSR11 measurement time used with this system was 30 s. In the VAX system the data acquisition time is normally 5 s. When both systems were running in parallel, the recorded Totals count rates were identical. When recording the background, the Reals count rates were also identical, but during a measurement there was a systematic difference of about 10%. This was finally traced to a flaw in the basic design of JSR11 and JSR12 (and of shift registers from other manufacturers). At the beginning of a measurement, the 'R+A' and 'A' registers are cleared, as are the contents of the long delay register. The 'R+A' and 'A' registers simultaneously started, but no counts reached the 'A' register until after one long delay period. This means that when the 'R+A' and 'A' registers are simultaneously stopped, the contents of the 'A' register are too small by an amount $T^2 g A t$, where $T$ is the Totals count rate, $g$ is the gate length and $A t$ is the long delay period. The fractional error in the Reals is $T^2 g A t/Rt$, where $R$ is the Reals rate and $t$ is the counting time. If we consider a numerical example, taking $T = 800 000$ counts/s, $R = 70 000$ counts/s, $g = 64 \mu s$ and $A t = 1024 \mu s$, then the percentage error in the Reals is 60/t. When the shift register technique is used manually, $t$ is usually of the order of 100 s and hence the error introduced would be 0.6%. In the case of the present unattended application of this technique, $t = 5$ s and hence the percentage error is about 12%. This effect is
proportional to the Totals count rate squared and inversely proportional to the counting time, so it is not always significant, and in some applications it will be compensated for by the calibration procedure. It should be noted that the manufacturers of JSR12 (and other coincidence electronics) have corrected the problem, but care needs to be taken in certain applications because the new units may necessitate a change in calibration.

4.2. Data transfer failure

The local electronics are supplied with an uninterruptible power supply, but the VAXs are not. The DASs store the data until they receive an acknowledgement that the VAX has successfully received the data. This was tested by switching off the VAX power on many occasions (system maintenance, etc.). However, on one occasion, when the power resumed on the VAXs after a power failure, it was discovered that the data from the DASs could not be transmitted after the communications were re-established. This was traced to a timing conflict in the data transfer protocol and necessitated a change in the DAS software.

4.3. Automatic data entry

The most time consuming part of the data review is the manual entry of the declared data. It is planned to upgrade the system so that the declaration can be entered in a specified directory on the VAX and can be automatically entered into the database. The data transfer to the VAX could then be done by floppy disk or over a network.

4.4. Gamma geometry

Preparations are under way to make small mechanical modifications to the neutron detector housing to enable the gamma detector to be placed closer to the sample in order to increase the count rate. It can be seen from Fig. 2 that a large part of the errors on the Pu mass due to the measured isotopics are random and therefore should be improved by better statistics. Use of a better $^{242}$Pu correlation in the MGA may lead to an improvement in the quality of the results.

REFERENCES


ACHIEVEMENT OF INTEGRITY IN SPECIAL RADIOMETRIC INSTRUMENT SYSTEMS AT SELLAFIELD

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Because of the highly specialized nature of many plant measurement problems at Sellafield, commercially available equipment is often unsuitable. Hence, there is a requirement to design and build specific instruments to provide a solution in these situations.

Intelligent radiometric instrument systems have been developed at Sellafield over a period of 18 years to perform various essential measurements in numerous plant applications across the site. Over this period, the number of demands placed on BNFL by regulatory bodies such as the Nuclear Installations Inspectorate has steadily increased. One result of this is the requirement to ensure the integrity of the systems provided, as they are frequently installed for the purposes of nuclear safety or accountancy, in addition to process control, and often play an integral role in plant safety cases.

These systems can be described in general terms as consisting of either a single radiation detector or multiple detectors which are sensitive to neutrons and/or gamma rays and produce analogue electrical pulses. These pulses are first processed by front end electronic units, which amplify, discriminate and eventually digitize them. Data processing equipment then interprets and analyses the incoming signals
to produce the appropriate response. This may take various forms, depending on the particular instrument, but, typically, it may be the presentation of plant status information to an operator on a control room VDU screen.

Within this scheme, there are numerous opportunities to introduce principles and practices, which contribute to ensuring the integrity of the overall system against unrevealed faults, the consequences of which could be hazardous. Figure 1 illustrates the main areas of consideration for a high integrity system.

**Electrical interference**

Because systems are to be operated in an industrial environment, it is essential to guard against all forms of electrical interference, which may occur on the plant but may not be apparent in the laboratory. This affects many aspects of the hardware design, including the instrument cabinet, the front end electronic components and the arrangement of cables.

**Design methodology and quality assurance procedures**

The instrument design life cycle follows a strict set of approved quality assurance procedures. A widely recognized software design methodology, Yourdon, is used and extensive independent testing and reviewing is carried out.

**Physical access**

In order to prevent accidental or deliberate misuse by personnel, various measures are adopted, including the provision of lockable cabinet doors and password
protection to prevent unauthorized access to the equipment. Operator interaction is kept to a minimum, and the interface is presented as simply and unambiguously as possible.

Self-checking

Built-in self-checks form part of the normal operating procedures for these instruments. For example, a gamma source of known energy may be exposed automatically by means of a software controlled actuator, prior to each measurement, in order to check the performance of the detector and its associated electronics. Automatic calibration and background checks may also be enforced, at intervals imposed by the system software.

Electronic hardware

In order to ensure the correct functioning of the hardware components of the system, devices such as DC line monitors, cabinet temperature sensors and memory checking are employed.

Software

The software incorporates extensive error checking throughout. Wherever a function or procedure has the potential to return an error, the software handles the situation in a controlled manner, thus avoiding the possibility of the system continuing to operate unaware of the occurrence of a fault.
RESEARCH AND DEVELOPMENT AND ANALYTICAL MEASUREMENTS

(Session 9)

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Abstract


The 1995–1996 Safeguards Research and Development (R&D) and Implementation Support (IS) Programme of the IAEA Department of Safeguards continues the process of identifying and documenting the needs for maintaining an effective and efficient safeguards system. The general aims of the Safeguards R&D and IS Programme are to provide a guide to R&D and IS needs in order to solve the Department’s technical problems and to assist the management in allocating effort and resources to the key needs of the Department. The appropriate documentation of the Programme also serves to improve co-ordination of all support activities of the IAEA with those of Member States. In 1993, the IAEA General Conference and the Board of Governors gave the Secretariat a mandate to further strengthen the effectiveness and improve the efficiency of the safeguards system. In response to this mandate the Secretariat expanded the safeguards R&D activities to stimulate the development of more comprehensive safeguards practices, including methods to enhance the IAEA’s ability to detect undeclared nuclear activities and facilities. This area of R&D activities is specifically emphasized in the 1995–1996 Programme. The Programme also includes the needs in support of the implementation of existing safeguards measures, procedures and instruments. In contrast to the previous editions of the Programme (e.g. the 1993–1994 Safeguards R&D Programme), in the 1995–1996 Programme the R&D and IS needs are integrated into a single document, providing a complete picture of the safeguards needs of the IAEA.

1. INTRODUCTION

The paper describes the Programme for Research and Development (R&D) and Implementation Support (IS) which summarizes the IAEA’s continuing efforts to enhance the effectiveness and efficiency of safeguards activities. The main body of the paper describes the Safeguards R&D and IS Programme in general terms. The Programme is issued by the IAEA Department of Safeguards (the Department) every two years and is reviewed annually. The structure of the 1993–1994
Programme has been revised in the 1995–1996 version. The previous Programme was confined to the R&D requirements of the IAEA for the solution of technical problems identified by the Department and to provide improved and/or new safeguards practices to meet the increasing demands for more effective and efficient safeguards. The current Programme was enlarged, to include certain IS requirements associated with the production, deployment, maintenance and support (e.g. equipment replacement and repair, preparation of users’ manuals, training) in the integrated R&D and IS Programme. The R&D component of the Programme continues to address such requirements as problem analysis, concept development, conceptual and detailed design, development, testing and evaluation. The integration of the two categories of requirements satisfies the interest of the management in a more effective administration of its internal and external resources in support of future safeguards needs.

The Programme presents future safeguards requirements in the form of statements of needs. These statements are general descriptions of current or future problems that have to be resolved. In certain instances the nature of the problem is well defined and a possible solution is apparent. In other instances the problem reflects anticipated trends in the development of the nuclear fuel cycle which the IAEA wishes to analyse and assess regarding their impact on future safeguards approaches, techniques and equipment. In such instances the statement of needs is more general and is not detailed sufficiently to represent the specific work required. For Member States Support Programmes (MSSPs) considering tasks in support of the Department’s requirements, detailed work plans are developed by the Department in co-operation with Member States. These plans are used jointly by the Department and the State to monitor progress in the conduct of tasks.

The 1995–1996 Programme seeks to achieve the following aims:

— To present in a comprehensive and organized manner R&D and IS needs to address specific concerns of the Department;
— To create a management tool to define the objectives of future safeguards needs, to allocate departmental and MSSP resources, and to demonstrate a high level of transparency and accountability for actions in connection with the Programme;
— To establish a firm basis for participation of Member States in safeguards development, including utilization of the unique technical capabilities of each Member State.

The development of the Programme followed procedures established by the Department in the Safeguards Manual. A group of experienced persons who are senior members of the Department and who represent a broad spectrum of expertise (R&D Needs Committee) performed a thorough analysis of future safeguards needs that had been identified within the Department, and defined the priorities of these needs and the time required for their solution. The needs are grouped according to
a predetermined and defined set of ranking factors, which describe them as being essential, important or desirable.\textsuperscript{1}

The R&D Programme is the principal administrative mechanism for managing future safeguards requirements. The requirements derive, in part, from safeguards plans, the key aspects of which are reflected in the Safeguards Departmental Plan and in the IAEA Medium-Term Plan. The systematic development of these requirements involves:

- The identification of problems over a predetermined time span (at present up to the year 2000) and
- The assignment of priorities to identified needs.

The Programme is a contribution to the Department's planning, programming and budgeting process, and drives the development of task proposals and detailed work outlines prepared by staff of the responsible operations and support divisions. Descriptions of the tasks and milestones are added to the division work plans, and financial implications are considered. Detailed task proposals are prepared when support by Member States is desired, and these data are entered in the Support Programme Information and Communication System (SPRICS). Administration of support programmes is the responsibility of the Safeguards Division of Development and Technical Support. Progress in meeting the R&D and IS needs included in the R&D Programme is reflected in the annual updating of the R&D Programme.

Active work to meet R&D and IS needs is in progress in many instances. In the execution of the Programme the Department utilizes internal resources (support from the regular budget), contract and/or procurement support, and support by MSSPs. The Department is further supported by certain Member States programmes which have safeguards applications but which are under the exclusive jurisdiction of a State. For instance, the considerable analytical capabilities of certain Member States are used by the Department to deal with short term safeguards needs. These resources are extensive and are vital to safeguards R&D and IS activities.

\textsuperscript{1} A need is deemed essential if a current condition can lead to a break of the IAEA's safeguards obligations and/or if the consequences of not satisfying this need are considered to be severe.

A need is deemed important if a consensus exists that the IAEA would benefit from the conduct of certain work aimed at satisfying this need.

A need is deemed desirable if a particular Division believes that fulfilment of this need will aid in accomplishing the work programme of the Division.
2. R&D PROGRAMME AND IAEA INITIATIVES TO STRENGTHEN THE EFFECTIVENESS AND TO IMPROVE THE EFFICIENCY OF SAFEGUARDS

An international safeguards system serves the overall objective of non-proliferation of nuclear weapons and thereby facilitates international trade in nuclear material, hardware and technology as well as international co-operation in the nuclear field. The amount of nuclear material and the number and complexity of nuclear facilities submitted to safeguards are growing continuously. This results in an increase in the IAEA's safeguards workload. Discussions about the further development of the safeguards system are centred around the intention to strengthen the effectiveness of the safeguards system while ensuring that the costs are minimized, consistent with effective safeguards. It is realized that the safeguards system of the IAEA must be modified so as to provide greater assurance that no undeclared nuclear facilities and activities exist in States that have entered into comprehensive Safeguards Agreements with the IAEA.

The R&D aspect has been a continuing function of the safeguards activities of the IAEA. This is evident in the increasing comprehensiveness of safeguards approaches and the increasing complexity of the Department's inventory of approved measurement instrumentation and containment and surveillance (C/S) devices.

The impetus for expanded R&D and IS safeguards requirements originates from:

— The necessity to adapt the application of safeguards to new conditions in the continuing development of the nuclear fuel cycle by Member States (e.g. reprocessing facilities such as Rokkasho);
— Proposals for strengthening the effectiveness and efficiency of the safeguards system.

The urgency of measures to achieve increased effectiveness and efficiency is made all the more imperative by the IAEA's zero-real-growth budget.

The adoption in 1993 of the Programme 93 + 2—a comprehensive programme for the development, assessment and testing of measures for improving the effectiveness and cost efficiency of the safeguards system and which will rely on resources of the IAEA and of Member States — raises questions on the relationship between this Programme and the R&D Programme. A comprehensive review of the Programme 93 + 2 indicates that the R&D and IS Programme provides ample opportunity for studies and activities anticipated for the completion of the objectives of the Programme 93 + 2. The Programme 93 + 2 is changeable, and some additional studies can be anticipated. The administration of the R&D Programme is sufficiently flexible to accommodate any new areas of investigation through a process of regular revisions of the Programme.

A Programme structure has been created for incorporating in a single docu-
ment the R&D needs for strengthening safeguards, for improving the cost effective-
ness of the existing safeguards system and the implementation support needs. In the
1995–1996 Programme, the 66 identified safeguards needs are divided into three
groups:

— Group 1: R&D needs for strengthening the effectiveness and improving the
efficiency of the safeguards system;
— Group 2: Facility/application specific R&D and IS needs;
— Group 3: Recurrent IS needs.

A review of the essential research, development and implementation needs follows.

Group 1: Strengthening the effectiveness and improving the efficiency of the
safeguards system

This group of needs requires an examination of new techniques, procedures
and/or equipment that will enable the IAEA to reduce its inspections or to conduct
them with less effort, to enhance its verification practices and the use of more effec-
tive C/S technology, and to expand the scope of safeguards and so broaden the
IAEA’s capability to provide assurance of non-diversion of nuclear material from
peaceful nuclear activities.

The needs in this group also include development activities for modified
safeguards measures, for example:

(a) Analysis of nuclear and nuclear related activities in States on the basis of inform-
    ation from open publications and other available sources, in order to identify
    the possible existence of undeclared nuclear activities, facilities and materials.
(b) Monitoring of indications related to the discovery of clandestine nuclear
    weapons development programmes.
(c) Examination and verification of design information, including information
    provided early and information on facilities closed down.
(d) Analysis of the export/import of nuclear and non-nuclear material, specified
    equipment and materials.
(e) Environmental monitoring for safeguards purposes.

The Programme identifies 17 needs belonging to Group 1. The highest priority
needs in the group include the following points:

(a) The Department continues its efforts to improve its capability for authenticat-
ing data obtained from equipment and procedures that remain under the control
of facility operators. The credibility of the IAEA’s declarations depends on its
ability to secure reliable information.
(b) The Department is expanding its efforts for early warning of activities for the production of nuclear explosive devices in a State. These efforts generate a need for methods for acquisition and systematic analysis of safeguards relevant data for the detection of potential non-peaceful applications.

(c) As part of its detection and early warning activities the Department needs new techniques for the detection of inconsistencies in certain activities of a State. Environmental analysis is one such technique; this requires assistance in the conduct of a comprehensive test programme, including determination of the cost and viability of such techniques under various conditions.

(d) The Department also needs well defined inspection procedures and techniques to be used at sites for which the available information indicates that clarification by Member States is required.

(e) The Department anticipates that certain measures for strengthening safeguards (e.g. detection of undeclared facilities) afford the potential for modifying the existing approaches (e.g. frequency of timeliness inspections of irradiated fuel) and so provide certain improvements in cost effectiveness. The Department requires assistance in developing suitable cost and benefit evaluation methods.

(f) The Department continues to place an essential priority on needs which have existed in earlier programmes and which now have been combined under the subject of alternative approaches based on modifications to the current safeguards system. Techniques requiring further assistance include randomization, zone approaches, mail-in of closed circuit television (CCTV) data by the SSAC, extended access to declared facilities and short notice inspections.

Group 2: Facility/application specific R&D and IS needs

This group covers enhanced and/or modified safeguards measures for power and research reactors (including reactors with Pu and HEU fuel), bulk handling facilities (BHFs) such as fuel fabrication facilities, reprocessing and enrichment plants, and storage and disposal facilities. The emergence of larger and more complex BHFs poses new problems for safeguards. In general, BHFs require a larger inspection effort, and this is expected to grow in the medium term. Therefore, the Department is placing high priority on the development of safeguards approaches leading to the attainment of the inspection goals; and on the development of systems, equipment and techniques that reduce resource requirements without sacrificing effectiveness.

The Programme identifies 28 needs which belong to this group. The highest priority needs in the group include the following points:

(a) The Department seeks to improve methods of verification of material balance accountancy at high throughput plants, including the design and testing of methods involving near real time accountancy (NRTA) and techniques for
direct acquisition of plant operating data, for example through software interfaces to the plant process control system. R&D needs also address large cumulative verification uncertainties in high-throughput plants, with emphasis on developing techniques for measuring/estimating hold-up in process equipment and transfer lines.

(b) The Department needs to address the verification of plutonium in the head-end of large scale reprocessing plants (the plutonium present there can amount to many significant quantities (SQs)). The material is in a flowing state and has a high concentration of fission products, which precludes representative sampling and analysis. Also, the safeguards approach for reprocessing and conversion facilities requires extensive sampling and analysis for verification of inventory changes and timely detection. The time and expense of preparing destructive analysis (DA) samples to be shipped from large scale BHFs to the IAEA laboratory are limiting factors. On-site sample verification measurement techniques are being explored.

(c) Spent fuel accumulation over the next several years dictates more rigorous R&D activities for the implementation of safeguards methods for long term spent fuel storage. Conditioning plants and near surface and deep geological disposal facilities for spent fuel are prospective developments that will require attention with regard to safeguards. Spent fuel repositories are expected to come under safeguards early in the year 2000. Conditioning facilities are expected to come under safeguards in the near future. The Department intends to develop safeguards approaches that can be communicated to future operators/owners of repositories so that appropriate safeguards features can be incorporated into the facility design. Research is under way and will continue in 1995–1996.

(d) The Department requires further assistance in improving the application of Cerenkov viewing devices (CVD) for verifying the spent fuel assemblies in spent fuel ponds of LWRs. This includes the capability to verify spent fuel assemblies of long cooling times (10–30 years) and low burnup (<10 GW·d/t), and the capability to record the CVD observations so that a further analysis of the verification results becomes possible.

(e) The Safeguards Analytical Laboratory (SAL) has a continuing requirement for reference materials that are used in the analysis of safeguards samples. In many cases such materials have a unique isotopic composition and so are not readily available to the IAEA.

Group 3: Recurrent IS needs

This group covers certain areas of recurring activities of the Department. The effective implementation of these activities would benefit from various forms of technical assistance. Examples of such technical assistance include support for inspector
training, upgrading and expansion of the IAEA Safeguards Information System (ISIS), development and implementation of the Safeguards Management Information System (SMIS), and assistance in the preparation of safeguards documentation.

The Programme identifies 21 needs which belong to this category. The highest priority need in this group is as follows:

— The IAEA would benefit from a broader evaluation process which makes fuller use of currently collected inspection information to provide a comprehensive integrated analysis of the goal attainment at the State level. In addition, the IAEA’s requirement for evaluation and reporting of safeguards results means that the current system for evaluating compliance with safeguards agreements for declared activities will have to be combined with the detection of undeclared facilities and activities.

4. PROGRAMME PERFORMANCE

The Programme is the Department’s primary resource for enhancing its capabilities to respond to the changing conditions with which the application of safeguards is faced. To use this resource most efficiently, the Department’s administration of the R&D and IS Programme provides for:

— An annual updating of the complete Programme and the issuance of the revised Programme document;
— A semi-annual review of the complete Programme by the departmental Management Co-ordination Meeting;
— A quarterly review by the Safeguards Co-ordination Meeting of the needs categorized as having an essential priority;
— A quarterly updating of MSSP tasks addressing R&D and IS needs as well as a corresponding updating of the R&D and IS database.

The Department Programme is also reviewed in the biannual MSSP co-ordinator meetings, and SAGSI reviews the biannual updating of the Programme.

During the annual revision and updating of the R&D and IS Programme, all needs are assessed and updated to reflect their actual status and current priorities, and new emerging needs are incorporated into the Programme. A revised Programme, after approval by the Department management, subsumes the previously issued document for the period until the next revision.

SPRICS plays a significant role in the revision process; in this system, all MSSP tasks addressing R&D and IS needs are tracked in accordance with the procedures established by the Division of Development and Technical Support and its Support Programmes Administration Office.
TABLE I. PROGRAMME STATISTICS

<table>
<thead>
<tr>
<th>1993–1994 R&amp;D Programme</th>
<th>R&amp;D needs</th>
<th>44</th>
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<tr>
<td></td>
<td>R&amp;D needs completed or combined with new needs</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>R&amp;D needs carried forward into 1995–1996</td>
<td>21</td>
</tr>
<tr>
<td>1995–1996 R&amp;D and IS Programme</td>
<td>IS needs integrated into the Programme</td>
<td>25</td>
</tr>
<tr>
<td></td>
<td>New and modified R&amp;D needs</td>
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<td></td>
<td>Total needs</td>
<td>66</td>
</tr>
<tr>
<td></td>
<td>MSSP active tasks (July 1993)</td>
<td>227</td>
</tr>
</tbody>
</table>

The Division of Development and Technical Support furnishes a quarterly update of MSSP tasks addressing essential needs, and this information will be entered into the SPRICS database. SPRICS is currently being enhanced so that the task officers will be able to update the status of tasks directly. This will enable system users to secure current information on the status of all tasks. The enhanced version of SPRICS, which will also include access to the R&D and IS Programme, will be fully operational by December 1994.

Since establishing the 1993–1994 Programme, the Department has completed work on twelve needs included in that document. The Department expects to complete work on six more needs included in the 1993–1994 Programme by 1995. The total number of needs in the 1995–1996 Programme has increased to 66 (compared with 44 needs in the 1993–1994 Programme). This increase is partially due to the decision to include IS needs in the 1995–1996 Programme. The majority of new R&D needs are in support of the effort to enhance the capabilities for strengthening safeguards and improving the cost effectiveness of safeguards. Table I gives relevant Programme statistics.

5. CONCLUSION

The Safeguards R&D and IS Programme for 1995–1996 continues the work started in 1988 to document the departmental R&D and IS services. The Programme contains 66 needs which cover all R&D and IS expected to be required during this period for the implementation of current safeguards as well as for the development of a future improved safeguards system. Some of these needs (21) were carried over from the 1993–1994 Programme, with modifications made to reflect the departmental views on the current status of each need and its updated relative priority. The Programme contains 25 IS needs — a category of needs that was not included in the 1993–1994 R&D Programme. A substantial part of the support which the Department receives from Member States is used for the IS needs.
The Programme has been prepared during a pivotal period in the development of safeguards. It takes into account the departmental objectives to strengthen the safeguards system and to increase its cost effectiveness. The Programme also supports a variety of proposed initiatives to increase the level of assurance of Member States' conformance with their safeguards obligations. A specific category of R&D needs has been introduced in the Programme structure to single out the needs stemming from these initiatives.

The Programme focuses on departmental objectives that are based on foreseeable future safeguards conditions. Such conditions include verification of nuclear material under increasingly more complex circumstances. Consideration is also given to potential future IAEA involvement in new areas of nuclear material verification and control. However, such new potential responsibilities for the Department are not yet sufficiently elaborated to formulate specific needs. There are administrative mechanisms that could provide quick Programme response if new responsibilities were to be imposed on the IAEA. Continuing clarification and specification of departmental long range objectives and safeguards options will be integrated during regular updates of the Programme.

The R&D and IS Programme relies on the voluntary participation of Member States. The general information presented in the Programme regarding each specific need (description of the need, priority level and desirable target date) will provide a basis for the departmental and MSSP managements for allocation of their resources in a way that will satisfy the interests and concerns of all Programme participants.
ESARDA: A UNIQUE FRAMEWORK FOR CO-OPERATION IN SAFEGUARDS RESEARCH AND DEVELOPMENT

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Abstract

ESARDA: A UNIQUE FRAMEWORK FOR CO-OPERATION IN SAFEGUARDS RESEARCH AND DEVELOPMENT.

The European Safeguards Research and Development Association (ESARDA) celebrates its 25th anniversary in 1994. A report is presented on the main objectives of ESARDA and the role it plays in the co-operation and harmonization of R&D activities in the European Union, and in the co-operation with the IAEA and with outside laboratories. Its management and working structure are presented, and the modes of interaction based on exchange of information and on the execution of joint activities are illustrated. Examples are given of how joint projects are conducted in the framework of the six existing working groups. Finally, some ideas are presented on the future orientation of ESARDA and the activities of the working groups for addressing new challenges in international safeguards.

1. INTRODUCTION

The European Safeguards Research and Development Association (ESARDA) was founded in 1969 and celebrates its 25th anniversary in 1994.

ESARDA has become a unique international forum in the European Union (EU) for the advancement and harmonization of R&D and for scientific and technical co-operation in the field of nuclear safeguards.

The majority of the research organizations involved in safeguards R&D within the EU, many of the plant operators of the European nuclear fuel cycle, the Commission of the European Communities, including its Joint Research Centre and the Euratom Safeguards Directorate, as well as Government representatives, are represented in ESARDA. In the six working groups there are also observers from several other research institutes and operators outside of the EU, involved in R&D for international safeguards.

The key element of ESARDA's activities is the frequent interaction through exchange of information between R&D staff, plant operators and safeguards inspectors.
The purpose of ESARDA, as defined in Article 2 of the Agreement between the Parties, is "to facilitate co-operation on R&D in the field of safeguards and on the application of such R&D to the safeguarding of source and special material. This co-operation is effected through co-ordination and harmonization of the R&D work of the partners, by the exchange of information and assistance on the personnel and technical levels and by the joint execution of these programmes or parts of it'".

ESARDA has always focused its main interest on the technical aspects of the application of Euratom safeguards in the nuclear fuel cycle. NPT safeguards by the IAEA has added another element to the application of safeguards in the EU, and this is also largely considered within the activities of ESARDA. Important changes have recently taken place in the world, and this has led ESARDA to review its activities and to enlarge also its field of interest to problems beyond those considered by the EU. The new directions considered by ESARDA for its future activities will be mentioned later.

2. MANAGEMENT AND WORKING STRUCTURE OF ESARDA

As of 1994, ten organizations are Parties to the ESARDA contract, including nine research organizations and one major industrial company.

The decision making body of ESARDA is the Steering Committee, which has final responsibility for all ESARDA activities. The subjects considered are mainly related to management and R&D policy matters. This Committee comprises up to four members for each Party and is presently composed of a total of 24 members. Each Party is entitled to select for its representation (members) on the Steering Committee several organizations in its country. These members represent, in practice, ten R&D organizations, seven industrial companies with direct interest in the fuel cycle, four inspectorates and four governmental bodies. In the near future the association of utilities and fuel cycle companies in Germany is expected to join ESARDA as a Party. Table I gives a list of the Parties to ESARDA and the member organizations in the Steering Committee.

A board was established, with one representative for each country, to discuss the management and R&D policy aspects of ESARDA and to streamline the preparation of decisions of the Steering Committee. The ESARDA Board also has to approve the publications made on behalf of ESARDA and the content of the ESARDA Bulletin.

For each country a scientific co-ordinator is assigned. The co-ordinator represents his country in the discussions and analysis of, for instance, the nuclear fuel cycle and its safeguards features, the R&D programmes submitted by the Parties, and the preparation of recommendations on possible joint activities, creation of working groups or ad hoc groups or special meetings. The co-ordinator’s committee is in charge of the management of the scientific activities of ESARDA.
### TABLE I. ESARDA PARTIES AND STEERING COMMITTEE MEMBERS

<table>
<thead>
<tr>
<th>Parties</th>
<th>Member organizations</th>
</tr>
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<tbody>
<tr>
<td>European Atomic Energy Community</td>
<td>DG XII (JRC)/DG XVII (Dir. C/ESD)</td>
</tr>
<tr>
<td>Kernforschungszentrum Karlsruhe GMBH</td>
<td>KfK/BMFT/GNS</td>
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<td>Centre d’études de l’énergie nucléaire — Studiecentrum voor Kernenergie</td>
<td>CEN-SCK/BN/EXT.AFF.</td>
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<td>Ente per le Nuove Tecnologie, l’Energia e l’Ambiente</td>
<td>ENEA/DISP, ENEA</td>
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<tr>
<td>United Kingdom Atomic Energy Authority</td>
<td>UKAEA/DTI</td>
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<td>Commissariat à l’énergie atomique</td>
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<td>British Nuclear Fuels plc</td>
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<td>Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas</td>
<td>CIEMAT</td>
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<td>Wirtschaftsverband Kernbrennstoff-Kreislauf (which will join soon)</td>
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The scientific and technical activities are discussed and performed by the different working groups operated currently by ESARDA. Three of these working groups are plant oriented, namely on safeguards for low enriched uranium (LEU) fabrication and conversion plants, for mixed oxide (MOX) fuel fabrication plants, and for reprocessing plant input. Three other working groups are discipline oriented, namely on analytical techniques, on non-destructive assay (NDA) techniques and on containment and surveillance (C/S) techniques.

Finally, the general Secretariat in the Joint Research Centre assumes the responsibility for the administration of ESARDA and acts as a focal point for the activities of ESARDA with respect to external organizations.

### 3. MODES OF INTERACTION AND CO-OPERATION

The interactions and co-operation between the ESARDA members and observers are based mainly on the exchange of scientific and technical information and on joint activities.
3.1. Exchange of information

Without any doubt, the exchange of scientific and technical information has been the most successful of the various ESARDA activities. This exchange of information takes place in different ways.

One way is the organization of technical symposia on safeguards and nuclear material management on a regular basis. At each symposium, approximately 120 papers are presented and, on average, 200 persons from the industry, inspectorates and R&D laboratories participate. The first of these regular three-day symposia took place in Brussels in 1979; since then, ten other symposia have been organized. At present, a symposium is held every two years. During the last ESARDA symposium in Rome in 1993, a special session was organized by the IAEA.

In order to increase the communication between working group members, the co-ordinators and the Steering Committee, and to discuss themes of common interest, it was decided to organize periodically (every two years) an internal meeting for all the working groups and the managerial bodies of ESARDA. On such occasions, the ESARDA co-ordinators prepare a paper introducing the subject and, in some cases, a questionnaire for consideration by the different working groups. Four such meetings have taken place:

1. In 1986 in Copenhagen on Capabilities and Objectives of the Use of NDA, DA and C/S Measures in Safeguards;
2. In 1988 in Karlsruhe on The Analysis of the Nuclear Fuel Cycle in EC Countries up to the Year 2000 and on Medium and Long Term Trends in ESARDA Working Group Activities;
3. In 1990 in Como on Technology Transfer in Safeguards;
4. In 1992 in Salamanca on:
   a. C/S Safeguards Techniques Applicable to the Intermediate and Long Term Storage of Irradiated Fuels, discussed more specifically in the framework of an enlarged C/S working group; and
   b. Non-destructive Assay Techniques Applicable to Safeguarding Nuclear Material in Waste, discussed more specifically in the framework of an enlarged NDA working group.

The next internal meeting will be held in May 1994 in Ghent. It will be the occasion to celebrate the 25th anniversary of ESARDA. It will provide the opportunity for the six working groups, the co-ordinators and Steering Committee members to reflect on the new orientation of ESARDA and of the technical work in the working groups.

Several topical meetings on subjects of particular interest were also organized by ESARDA, mostly by the working groups. Examples are: Isotopic Correlation and its Application to the Nuclear Fuel Cycle; Seminar on C/S for International
Safeguards; Harmonization and Standardization in Nuclear Safeguards; Joint (ESARDA-INMM) Specialist Meeting on NDA Statistical Problems; Optical Surveillance Data Reduction Techniques; International Workshop on Passive Neutron Coincidence Counting.

Another important initiative to improve the exchange of information is the publication of the ESARDA Bulletin, which is widely distributed (about 2200 copies) and informs about the activities of ESARDA. The Bulletin also presents technical papers in different areas of safeguards interest. The first ESARDA Bulletin was issued in 1981, and 22 numbers have been published since then.

In order to inform external organizations about ESARDA activities or findings of its internal meetings, ESARDA papers were presented at the IAEA Symposium on Nuclear Safeguards Technology 1982 and in INMM annual meetings. These papers were presented by the chairman of ESARDA or by the convenors of working groups. Furthermore, in 1979 and 1980 the INMM organized special ESARDA sessions during their annual meetings. Reciprocally, INMM representatives participated in the series of ESARDA symposia in Europe.

Last but not least, a large basis of communication exists at the level of coordinators and working groups. A permanent agenda item in the working groups (in particular the discipline oriented ones) is the informal presentation and scientific discussion of the ongoing R&D activities in the different participating organizations.

One cannot emphasize enough the fact that the exchange of scientific and technical information between ESARDA partners in working groups and between ESARDA and external organizations is the basis for the harmonization and coordination of the R&D activities and the design of joint activities, which have to respond to the short, medium and long term needs of plant operators and safeguards inspectorates.

3.2. Joint activities

Joint activities include mainly two elements:

- The co-ordination and harmonization of R&D activities of the Parties, and
- The execution of joint technical projects.

3.2.1. Co-ordination and harmonization

Article 3 of the Agreement of ESARDA states that “the R&D or application programmes of the Parties brought up to date annually shall be submitted to the Steering Committee, which will examine them in the light of identified safeguards topics, which require investigation and recommend a basis for co-operation”. The practical implementation of this key element is performed in the following way.
The identification of R&D topics of common interest, and proposals for cooperation and harmonization are based on the analysis of:

— Ongoing R&D activities of the Parties;
— Medium and long term (5–10 years) evolution of the nuclear fuel cycle; and
— Specific (in general short term) needs formulated by potential users of R&D results (safeguards authorities and plant operators).

Additional points considered in the analysis are the existing safeguards approaches and those under discussion as well as the general evolution of technologies of potential interest for safeguards applications.

ESARDA has launched a number of initiatives to facilitate the critical analysis of the above mentioned input data.

The first initiative is the establishment of the so-called ‘ESTABANK’, for facilitating the handling, processing and use of information provided by the co-ordinators on the R&D programmes of ESARDA partners. In this database the activities are labelled according to different criteria, such as the framework in which a certain activity or task is performed, the technical discipline, the status of development, potential areas of application and co-operation partners.

Recent results of the analysis of ongoing R&D activities of partners in the field of C/S techniques, of NDA techniques and of analytical techniques have been published in the ESARDA Bulletin. A similar review and analysis has now been completed on techniques applied or studied for reprocessing plant and fuel fabrication plant safeguards. These analyses provide a good overview of the degree of development and practical implementation of different techniques (laboratory stage, field application, industrial production, implementation for inspection or material control).

The second initiative is the creation of a fuel cycle database, which includes all the industrial nuclear facilities in the EU and their main characteristics. These characteristics comprise the location of the facilities, the type of nuclear materials handled or processed, and the potential throughput and storage. Special attention is paid to the collection of data of general interest to safeguards. In 1988, such a database was established for the EU countries, with the time horizon of the year 2000. The database is now being reviewed, with the time horizon of the year 2020, including nuclear facilities of EFTA countries and, to a limited extent, of Eastern European countries. The database is operated by the JRC at Ispra.

The Euratom Safeguards Directorate participates in all the meetings of the ESARDA co-ordinators and provides regularly input on its general and specific needs. The co-ordinators will also initiate at their next meeting the analysis of the 1995–1996 R&D and Implementation Support Programme of the IAEA in order to find out whether the ESARDA research programmes cover these needs.
3.2.2. Joint technical projects

Joint technical projects performed in the framework of ESARDA working groups aim, in general, at:

— Establishment of capabilities and performances of measurement techniques and data evaluation methods;
— Development and recommendation of procedures for the application of measurement and data evaluation methods;
— Identification of needs and status of development of methods and techniques (including reference materials).

Some examples of such joint projects are given below.

(a) Target and performance values

The Destructive Analysis (DA) working group of ESARDA established in 1979 so-called ‘target values’ for analytical measurements. They correspond to values of the random and systematic error parameters to be aimed for in elemental and isotopic analyses of the most significant types of materials, using common destructive analytical methods. Several revisions (in 1983, 1987 and 1988) of the target values took place. The IAEA has always had a strong involvement in this project. The forum of discussions was also enlarged, including specialized committees of the INMM and other international organizations.

The definition of the target values is based on different sources of information. One source is the results obtained from the statistical evaluation of measurements reported by the facility operators and the results of independent measurements performed on the same material by the inspectors. A second source is the results of laboratory intercomparisons or measurement quality evaluation programmes. Finally, the expertise of measurement specialists for in-depth analysis of the uncertainty components is also largely taken into account. The January 1994 issue of the INMM Journal and the March 1994 issue of the ESARDA Bulletin published a condensed version of the ‘1993 international target values for uncertainty components in fissile isotope and elemental accountancy for the effective safeguarding of nuclear materials’. These data include also target values for bulk measurements and some NDA methods used as accountancy and verification tools. The target values, now called international target values (ITV), are receiving worldwide recognition. Euratom and the IAEA are using the results of this long standing activity for their safeguards evaluation activities.

In the past, a similar effort was undertaken by ESARDA to study and establish target values for uncertainty components in sampling. In this context, one has to distinguish between sampling before chemical and isotopic analysis and sampling in process lines or storage areas. Several working groups have put some effort into
defining the uncertainty of sampling, such as for UO₂ sampling in barrels and sampling from reprocessing plant input accountancy tanks. General discussions were held on sampling in different parts of the process lines in LEU and MOX fuel fabrication plants. The 1988 target values included the random error parameters to be met in the elemental assay as a result of sampling.

The NDA working group established several years ago a list of all NDA techniques available or under development and their range of application. In a similar manner, a list of reference materials (primary and working standards), available in the EU laboratories, was prepared. Recently, the NDA working group started an action with the objective of establishing ‘performance values’ for NDA techniques on different kinds of material. The NDA performance values are defined as “knowledge of the overall uncertainty and error sources associated with an NDA measurement system”.

The following techniques are now being considered:

— Gamma ray spectrometry, passive neutron techniques and calorimetry on plutonium bearing materials;
— Gamma ray spectrometry and active neutron techniques on uranium bearing materials;
— Gamma ray spectrometry and neutron techniques on spent fuels;
— К-edge densitometry and X ray fluorescence on U, Pu and U–Pu solutions.

A preliminary analysis was also performed on data and information compiled from monitoring of waste materials by NDA techniques, in particular during the last internal ESARDA meeting, held in Salamanca in 1992.

These NDA performance values are established on the basis of the experience gained by different laboratories, plant operators and inspectorates in the application of techniques in laboratory and field conditions. They are also the outcome of tailored laboratory experiments based on well characterized standards, interlaboratory measurement programmes and the results of a detailed analysis of field inspection measurements. The latest NDA performance values were issued by the working group at the ESARDA symposium in 1993 in Rome.

The C/S working group has discussed extensively methodologies to express the assurance/performance of C/S devices and systems.

(b) Experimental determination and validation of performance values for measurement and data evaluation methods

As mentioned before, the results obtained from interlaboratory measurement evaluation programmes, quality control programmes or tailored laboratory exercises are an important contribution to the definition or validation of the performance values of measurement and data evaluation methods. In the framework of ESARDA,
several exercises have been conducted or are being discussed in the different working groups. A few of the most recent examples are given below:

(i) The Regular European Interlaboratory Measurement Evaluation Programme (REIMEP), is organized by the JRC (IRMM), Geel, with a very broad participation of laboratories worldwide, including the IAEA. In 1992, the programme included the distribution and destructive analysis of MOX pellets, spent fuel solution and plutonium nitrate solution. In 1993, UF₆ samples were distributed for DA and NDA. The overall results of the programme are presented to the DA working group. The Euratom Safeguards Directorate is providing financial support to REIMEP as part of the quantity control programme for its network laboratories. Furthermore, the status and results of the EQRAIN (Evaluation de la qualité du résultat d’analyse dans l’industrie nucléaire) programme, managed by CETAMA, Paris, are also discussed in the DA working group.

(ii) The LEU working group had a particular interest in the assessment of the error components for the determination of the uranium concentration in UO₂ powder by gravimetry and of uranium in solutions by potentiometry. An intercomparison exercise was conducted by the JRC (IRMM), Geel, for this purpose, and 14 laboratories participated, including the IAEA’s Laboratory.

(iii) A plutonium isotope determination intercomparison exercise (PIDIE) by γ ray spectrometry was performed in the framework of the NDA working group. The aim of the exercise was to test the X ray and γ ray methods used for measuring plutonium over a wide range of isotopic composition, to give the opportunity to improve them, to investigate sources of error and, if possible, to improve the knowledge of emission probabilities. Seven sets of samples with different isotopic composition were circulated by the UKAEA to ten laboratories. The measurement results analysed were extensively discussed in the working group and are being used for the determination of the NDA performance values.

(iv) Several years ago, the NDA working group promoted an action for the procurement of certified reference materials (CRMs) (five different ²³⁵U enrichments of U₃O₈ powder) for NDA enrichment measurements by γ ray spectrometry. A large number of sets of these CRMs were prepared by the JRC (IRMM), Geel, and characterized by the IRMM, the National Bureau of Standards (USA) and several NDA laboratories. An intercomparison exercise was also conducted using these U₃O₈ reference materials. This action led to the evaluation of the ultimate performance that can be achieved by this technique on bulk powder material, and a detailed measurement procedure was also developed.

(v) Neutron coincidence counting is one of the basic techniques used in the assay of plutonium. Much experience has been gained for the application of this technique in laboratories, by safeguards inspectors and by plant operators. In April 1993 the NDA working group convened at the PERLA laboratory in Ispra an international workshop on passive neutron counting applied to the assay of plutonium bearing
materials. The basic issues discussed were: the evaluation of the applications and performances of well established shift register based instruments and the evaluation and future prospects of the new generation of passive neutron instruments, based on neutron multiplicity analysis. Demonstration exercises were performed at PERLA, and a report with the recommendations of the group was published.

(vi) A number of measurement methods for volume and mass determination have been implemented routinely by plant operators and safeguards inspectors in pilot facilities of reprocessing plants. The reprocessing input verification (RIV) working group participated in an important exercise (managed by DWK/GNS) for intercomparing different calibration and measurement systems on a large size (12 m\(^3\)) tank in cold conditions, the so-called CALDEX exercise. Eight different measurement devices and five methods were evaluated. Twelve organizations from Europe and the USA, and the IAEA participated in this exercise. The results were extensively discussed by the working group, and some of them have been published. An enlarged exercise is being organized at the TAME laboratory at the JRC, Ispra, on a variety of tanks, including the CALDEX tank, which was transferred from Karlsruhe to Ispra.

(vii) An isotopic correlation technique benchmark exercise has been conducted by the RIV working group of ESARDA. This exercise consisted of the intercomparison of the performances of different isotopic correlation techniques for the verification of the input inventory of a nuclear fuel reprocessing plant. To the seven participants of the exercise, Cogéma supplied data (chemical and isotopic analysis and Pu/U ratio) from 53 routine reprocessing input batches made of 110 irradiated fuel assemblies. The results have been published.

(c) Harmonization of procedures

One of the results of interlaboratory exercises and discussions in the ESARDA working groups is the development of detailed measurement and data evaluation procedures. A few examples are given below:

(i) An exercise was originated by the LEU working group in order to establish general and easy procedures of weighing that would enable users to control the accuracy and precision of their scale system. Standard weights, in the range of 1–40 kg, were circulated among the plant operators in Europe and the results were analysed by the JRC, Ispra, and then discussed in the working group. On the basis of two measurement campaigns a procedure has been prepared and made available to plant operators.

(ii) The MOX working group studied the practices applied in the different facilities in Europe for the calculation and reporting of nuclear transformation, and a paper on this study is to be issued soon.
(iii) Both the LEU and MOX working groups continue to express their interest in discussions on nuclear material statistical accountancy systems for the evaluation of material unaccounted for, and on reference and real plant data. The goal is to define common calculation procedures and possibly computer packages and to make a correct interpretation of the different error components in the material balance evaluation of fuel fabrication plants.

(d) Technical needs and state of development

The working groups have the permanent task to keep up to date regarding the latest developments of different techniques and the experience gained in their practical applications. This task is performed by means of general discussions of the R&D activities of the participants at working group meetings and by organizing topical meetings for the purpose of establishing the state of the art for a particular subject.

The C/S working group organized a specialists meeting on Optical Surveillance Data Reduction Techniques, where the implications of front end versus back end optical data reduction in video surveillance for safeguards were discussed in detail. At the internal ESARDA meeting in Salamanca on C/S Safeguards Techniques Applicable to the Intermediate and Long Term Storage of Irradiated Fuel, the C/S working group discussed safeguards relevant facility design features and basic safeguards concepts for spent fuel storage facilities, established criteria for the development of C/S equipment, and identified relevant C/S devices and techniques. The top priority areas of development proposed were: integrated system techniques, remote data transmission and remote monitoring, data reduction and evaluation techniques, and authentication techniques.

The C/S working group has also decided to issue a compendium of C/S devices, a compilation of outline information on a range of products and particular devices that could meet the requirements of specific applications.

The LEU and MOX working groups discussed the specific measurement issues of scrap and waste material. The different categories of scrap and waste material encountered in such facilities were defined, and the existing measurement capabilities and needs were noted. This information was used by the NDA working group for discussions on performance values and development needs. As mentioned earlier, the NDA working group concentrated its discussions at the 1992 internal meeting in Salamanca on the issue of NDA Techniques Applicable to Safeguarding Nuclear Material in Waste, in the different parts of the nuclear fuel cycle.

In general, the plant oriented working groups are interested in regular information on the ongoing developments by the technique oriented working groups in order to evaluate, at an early stage, the possible impact of these developments on the nuclear material management and verification practices in their facilities. Several combined meetings were organized between working groups, such as between the
MOX and NDA groups and the DA and LEU groups, to promote this type of dialogue.

Because of the need to develop more and more instruments using a combination of different techniques, which have to be integrated, the NDA and C/S working groups have decided to organize joint meetings. Similarly, the DA and NDA working groups also meet periodically to discuss common approaches regarding, for instance, traceability or the definition of target and performance values.

4. CHALLENGES AND FUTURE TRENDS OF ESARDA

In December 1992, the Steering Committee decided to begin to consider the future orientation of ESARDA and the activities of the working groups. A reflection group was established for this purpose and a number of subjects were discussed; these are briefly described below.

The need to further study in a working group the technical aspects related to the application of safeguards in the intermediate storage of spent fuel and waste was assessed. The group also exchanged some ideas on the possible technical consequences and on new development work required for the application of the New Partnership Approach between Euratom and the IAEA.

The R&D requirements for implementing the proposals for the strengthening of safeguards have been considered (for instance in the area of environmental monitoring). The need to involve ESARDA in studies related to verification techniques required for safeguarding ex-military nuclear material, coming from weapons dismantlement, has been discussed.

The possible role of ESARDA has been examined in the field of communication and dialogue with non-safeguards experts on the features of nuclear safeguards and the experience gained in its implementation in a supranational and international inspection regime. This last point is considered important because non-proliferation and safeguards issues are more and more becoming key elements, in addition to safety, waste management and radiation protection, for the acceptance and safe development of nuclear energy.

The reflection group considered the possible enlargement of ESARDA, to include other Parties, and co-operation with the former Soviet Union and Eastern European countries.

Finally, the Steering Committee asked the reflection group to pay special attention to analysing the organizational structure of ESARDA and to make proposals to streamline the internal decision making process and working structure.

The reflection group will present its final report to the Steering Committee at the next internal ESARDA meeting in May 1994 in Ghent.
REAL COSTS OF SAFEGUARDS INSTRUMENTS

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Abstract

REAL COSTS OF SAFEGUARDS INSTRUMENTS.

In the last decade there has been a large increase in the use of instruments for international safeguards. The number of different types of instruments as well as the quantities of the same instruments have increased substantially. At the same time the cost per unit of nuclear material safeguarded has decreased (by approximately a factor of three). Clearly, the use of safeguards instruments by the IAEA has contributed to improvements in efficiency. However, the costs of the instruments and the instrument programme have also increased. The paper examines some of the cost components of the instrument programme in an attempt to introduce a more global perspective on the use of instruments for international safeguards.

1. INTRODUCTION

Budgets and costs have always been important in the IAEA, but, in recent years, budgets have attained an even higher level of significance. Equipment purchase is an important item of each budget and, because of the uncertainties over the last three years, imaginative planning and flexibility have been required to accommodate the basic inspection equipment needs. The purchase of equipment is, however, only one factor of a number of cost components associated with the use of instruments for international safeguards purposes. The total costs, or resource requirements, need to be identified and quantified to enable a comprehensive management review (and optimization).

IAEA resources related to the support of safeguards instruments are approaching the saturation point. This is due primarily to the large number of instrumental systems authorized for safeguards use (76) and the rapid increase in the use of instruments in recent years. The total inventory of inspection related equipment is approximately 5000 items, worth roughly US $20 000 000. These numbers reflect the understandable tendency of the IAEA and of its Member States to rely increasingly on instruments with which other costs can be saved while maintaining credible safeguards.

It must be noted that the applications of IAEA safeguards instruments are varied, since the instruments are utilized for all chemical and physical forms included in the peaceful nuclear fuel cycle. (The number of different applications may further
FIG. 1. Repetitive or operating costs (left hand side) and one-time costs (right hand side) for development and implementation of safeguards instruments.
increase in the future if international safeguards are applied to dismantled weapons material.) The geographical distribution of the utilization of instruments also contributes to the need for a large inventory of equipment types and quantities.

It is our intent to put into perspective the overall costs of safeguards instruments in order to direct more effectively the resources required for the instrumental programme in the future. The following questions have to be considered: "Are all the direct and indirect equipment costs known and are there changes or improvements to the overall approach that can be made in order to realize effectiveness and efficiency? If this is the case, at which point and to what degree can these improvements be made?"

2. INSTRUMENT COST COMPONENTS

Generally, instrument activities can be classified into three categories:

- Development,
- One-time implementation activities for a new instrument,
- Ongoing activities during the operating lifetime of an instrument.

Development costs, as defined in this classification (i.e. from the IAEA perspective), are the costs incurred by the developer of instruments in the course of designing, constructing prototype units, testing these units, documenting the operation and transferring the technology to the IAEA. In almost all actual situations the developer has a contract with a Member State or receives financial support through a Member State Support Programme (MSSP). Historically, the developer has most often been a national laboratory; however, more recently, there has been some competition from private contractors. In either case, the IAEA may have limited control or knowledge of the development costs that are incurred. All of the costs are incurred by the developer, except for a small percentage of costs for liaison incurred by the IAEA. The IAEA provides the user requirements and works together with the developer on the functional specifications of instruments. The adequacy of the final product is determined in relation to the functional specifications.

When the development phase of a new instrument has reached an appropriately defined point, it is necessary to start activities associated with the integration of the instrument into the IAEA instrument programme. Many of the costs associated with this second phase are incurred by the IAEA. The activities listed on the right hand side of Fig. 1, i.e. the one-time costs, are of a generic nature. For a specific instrument the one-time implementation costs will vary somewhat.

A major activity covered by the one-time implementation costs is the instrument acceptance test. The instrument is tested and evaluated relative to the functional specifications — generally at a site that represents typical, or actual, field conditions. The capability and operational procedures of the instrument are tested and evaluated
following a predefined acceptance test plan. Documentation can also be a major cost factor, although the costs are generally shared between the developer and the IAEA. The costs for development of a training programme may also be shared. Investigation and documentation of the procurement specifications and shipping procedures are normally primarily an IAEA effort.

The third category in this classification scheme (left hand side of Fig. 1) represents the recurring or operating costs that are incurred over the implementation lifetime of an instrument. Nearly all of these costs have to be paid by the IAEA. These costs are also of a generic nature, and their impact and relative importance will vary substantially from instrument to instrument. Major components include the training of inspectors and support staff, shipping and handling, development of facility specific procedures, and inventory control and maintenance. The latter include the development and implementation of performance monitoring procedures and a database. The operating and maintenance data are maintained in this database for periodic review and subsequent upgrading of the hardware and software components as well as for the instrument application operating procedures.

3. ACTUAL INSTRUMENT COST COMPONENTS

Detailed consideration of the generic cost components of IAEA safeguards instruments relative to actual situations indicates that no ‘average’ safeguards instrument exists. Each instrument on the IAEA inventory has its own peculiarities.

As an example, consider the modular integrated video system (MIVS) for optical surveillance and the Cerenkov viewing devices. Our experience with MIVS is that the costs for installation, maintenance and performance monitoring dominate, in relative terms, compared with other MIVS operating costs. On the other hand, for the Cerenkov viewing devices there are no installation costs and very low maintenance costs. However, the training costs for these devices (two weeks, including training on the physics of Cerenkov radiation and hands-on training at a fuel storage site with special fuel assemblies) are substantial, since the effectiveness of the instrument is directly related to the inspector’s capability of obtaining and interpreting the appropriate image patterns.

Even for a real situation, it is practically impossible to precisely quantify each of the cost components for the total life-cycle of a safeguards instrument. This is due to the combined work of different organizations, the activities that are done in parallel and the different methods of accounting for resources. Some data do exist, however, and for an understanding of the activities it is useful to extrapolate from these admittedly incomplete data the orders of magnitude of the effort involved for the more important components.

First we consider the one-time implementation costs for a safeguards instrument. The acceptance test provides experimental evidence that the newly developed
instrument is capable of meeting its functional specifications under realistic conditions. Normally, this test is performed in a typical facility, using the proposed inspection procedures. Preparation, implementation and evaluation of the test may require two to three person-months of effort. (If the technique is totally new to the IAEA, the testing period could be considerably longer.) Documentation is a major activity. The operating manual will normally be produced by the developer, with minimum involvement of the IAEA. However, the application procedure will be produced with some involvement of the IAEA and tested separately, probably under an MSSP task. The total documentation, testing and final editing effort may exceed six person-months. Requirements for other activities are as follows: development of a training programme: two to six person-months; procurement specifications and determination of a manufacturer: two person-months; shipping procedures, including the development of an appropriate shipping container: up to one person-month. Finally, for the maintenance staff some training on preventive maintenance and repair of the instrument is required — typically one week for four staff members, which adds up to one person-month. Thus, for the one-time activities, some of which may be performed under the auspices of the developing MSSP, a total of 15 to 17 person-months may be required.

Ongoing activities, which continue over the instrument’s operating lifetime, are even more complex and variable. Operating costs are directly dependent on the quantity and quality of a specific instrument deployed.

One of the more mature instruments in the IAEA safeguards inventory is the family of neutron coincidence counters (NCC). At present, the IAEA has approximately 51 systems, with 11 different detector heads (an additional ten other types are used by the IAEA, but are owned by facility operators). An operating system consists of an electronic package, which the inspector uses to control the measurement, and a detector head, which contains neutron detectors and preamplifiers in a specified geometry. The electronic package has a serial port that can be connected to a computer for automated data analysis. The first versions of this type of NCC were developed in Los Alamos and subsequently deployed by the IAEA in the 1970s. This type of NCC is the primary quantification verification instrument for plutonium bearing material.

For this mature instrument system the initial (one-time) requirements have already been met, and repairs have settled down to random breakdowns, i.e. design deficiencies have been rectified. In 1992, the estimated support resources were:

- 1.8 person-years for software maintenance and software training;
- 2.0 person-years for maintenance, calibration, inventory control, training instructors, and shipment and handling;
- 1.3 person-years for formal training courses for inspectors regarding operation of the instrument;
- 0.4 person-year for self-training/familiarity training and consultations.
Thus, the estimated total for this major operational instrumental system is 5.5 person-years. These figures for the ongoing activities in 1992 seem to be representative for this specific instrumental system, except perhaps for the extraordinary software maintenance costs.

There are situations where the ownership and maintenance of the instrumental system are retained by the State or the facility. Of course, there are still certain costs which must be covered by the State (through the MSSP). An example of a State owned instrumental system is the Consulha equipment. This consists of optical surveillance units and radiation detectors in a spent fuel storage area of a reprocessing plant. The system has only recently been authorized for inspection use and therefore it is not at the same stage of maturity as the NCCs. The 1992 resource requirement for this one-of-a-kind system was US $40 000 for maintenance alone (for an on-call maintenance contract with a private contractor). For the Consulha equipment, special training is also necessary, since it must be done on the site (requiring inspector travel and per diem costs). For 1992, the training consisted of two sessions (seven inspectors per session) with three professional instructors. It should also be noted that the IAEA may have to recheck the functionality and integrity of an in situ system that has undergone major repairs.

Unfortunately, a comprehensive set of cost information for the entire life cycle of an instrumental system is not readily available. However, it seems that, on the basis of extrapolating the available information and experience, a reasonable model can be constructed which provides some insight into the real costs of a safeguards instrument.

4. COMPOSITE INSTRUMENTAL SYSTEM

For our model we shall assume a non-destructive assay (NDA) system for moderate use, with a capital cost of US $25 000 per unit. The numbers quoted cannot be considered as definitive ones, but preciseness is not necessary since the model basically represents a reasonable summary of our present experience.

The operating lifetime of the instrument is assumed to be 10 years (this includes modifications and upgrades). A 'moderate use' system is defined as a system requiring 20 units. The development costs are roughly one to two person-years (this is a highly variable number) over a development period of 1.5 years, with the production of prototype units; thus the total costs are ~US $500 000. The total one-time implementation costs are assumed to include 2.8 person-years (1.5 actual years) of an IAEA professional at the P4 level, i.e. ~US $322 000. The operating costs are obtained by assuming two G6/G7 technical staff per year for the ten year operating period, i.e. total costs of US $1 400 000. The simplifications for the one-time costs and the operating costs are based on the most likely requirements for these periods. Of course, not all work could be performed by a single individual.
FIG. 2. Estimation of costs for twenty units of a safeguards instrument with a lifetime of ten years.

With the above assumptions the total lifetime costs for a composite instrumental system are:

- Capital costs: US $ 500 000
- Development costs: US $ 500 000
- One-time costs: US $ 322 000
- Operating costs: US $ 1 400 000
- Total costs: US $ 2 722 000

The estimated costs of a safeguards instrument, excluding the capital costs, are shown graphically in Fig. 2. We observe that this figure has the typical form of an obsolescence curve. At the beginning the costs are high, in our case for development and initial implementation, and then there is a rather flat part which remains constant for the operating lifetime, with perhaps a slight rise near the end of the instrument lifetime, indicating increasing maintenance problems.

5. CONCLUSIONS

We present some general conclusions that can be drawn from this seemingly arbitrary cost model for a composite safeguards instrument. The immediately
The obvious point is that, even for a modest instrument, of US $25,000 per unit, the total lifetime cost is a substantive sum. Changes in some of the assumptions would alter the final result, but the total cost relative to the capital cost per unit would remain relatively high. Since the total cost is large, any economies that could be realized, particularly in the operating costs, would have a significant financial impact.

For the situation assumed, the capital costs are less than 25% of the total costs. The percentage number is somewhat arbitrary; however, it is difficult to conceive of a situation where the capital costs are not a small part of the total lifetime costs.

The development costs and the one-time implementation costs are comparable. The sum of these two cost components is less than the operational support costs (which are more or less linearly dependent on the operating lifetime of the instrument). It is of interest to note that, by definition, purchasing off-the-shelf instruments would avoid development costs. However, it is not possible to know in advance the effect on the implementation and operating costs. One could argue that using off-the-shelf equipment might increase both the initial implementation costs and the operating costs. The initial implementation costs could increase because the IAEA may not have the benefit of the developer to transfer the technology efficiently. The operating costs could remain high if the IAEA has no possibility to improve the capability of the commercial instrument so that it meets the requirements of the safeguards application.

Finally, it seems obvious that the cost effectiveness of one-of-a-kind instrumental systems requiring substantial operating support should be carefully considered before the instrumental system is introduced into the IAEA instrument programme.
THE IAEA'S ANALYTICAL CAPABILITIES FOR SAFEGUARDS

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Abstract

THE IAEA'S ANALYTICAL CAPABILITIES FOR SAFEGUARDS.

Up to 1991, analytical measurements were used primarily to verify the accountability of nuclear materials placed under IAEA safeguards. The objectives of the analytical services set up in the 1970s by the IAEA for this purpose were to improve the accuracy of the analyses, to reduce the response time of the laboratory services and to increase their cost effectiveness. The major challenge was and still is to provide the IAEA with the capability to confirm with high probability that no significant quantities of safeguarded nuclear materials have been diverted from nuclear plants with even very large inventories or throughputs. The paper describes the progress made in this respect at the Safeguards Analytical Laboratory and in the network of analytical laboratories of the IAEA, with the introduction of robotized chemical procedures, X ray techniques, mass spectrometric methods and advanced spectrum evaluation software in radiometry. The aftermath of the Gulf war in 1991 demonstrated the need to complement the traditional accountability verification measurements with analytical systems that provide the capability of identifying sensitive non-nuclear materials, such as maraging steel used in centrifuge technology, and of detecting radioactive releases in effluents and in the environment, which may be indicators of clandestine nuclear activities. Relevant analytical resources existing at the IAEA or provided by Member States are described.

1. INTRODUCTION

The IAEA has traditionally used the concept of materials accountability in the implementation of Safeguards Agreements with States party to the Non-Proliferation Treaty (NPT) as well as other agreements. Materials accountability requires first that special nuclear materials under safeguards must be inspected in order to independently verify the declared amounts present. Frequently, this inspection takes the form of counting items, such as fuel assemblies, bundles or rods, or drums of powdered compounds of uranium and/or plutonium. This activity is
termed gross defect detection. The next level of verification, called partial defect
detection, may involve the weighing of items or measurements with non-destructive
assay (NDA) techniques such as neutron counting (active or passive) or gamma
spectrometry. These NDA techniques are capable of measuring the amount of
nuclear material with a relative uncertainty in the range of 1–10%, which is
appropriate whenever the throughput or inventory of material is not too large. For
larger amounts of material under safeguards, it is necessary to sample some of the
items and to apply chemical analysis techniques having the lowest possible relative
uncertainty, typically in the range of 0.1–1%. These techniques are termed destruc­tive assay (DA) methods and their use requires that the IAEA have access to labora­
tories which apply such precise and accurate techniques on a routine basis.

A second requirement for laboratory services regarding materials accountabil­
ity is related to the need for reference materials, working standards, and control
materials used in the assurance of the quality of measurements on nuclear materials,
whether by the inspectors (DA and NDA techniques) or by the responsible
laboratories in the State under inspection. This activity may involve the provision of
specialized reference materials for the calibration of measurement systems used by
the inspectors in the field. A complementary measure of control of the quality of
on-site measurements is to take a parallel sample of the item subject to on-site
measurement and to send it to a laboratory applying more accurate DA. Another
measure is the provision of control samples used in the verification of the operator’s
measurement (VOM) system. Commonly applied control samples are also needed
whenever several laboratories are used for the analysis of safeguards samples, as in
the case of the IAEA’s network of analytical laboratories (NWAL).

A third requirement for laboratory services regarding materials accountability
results from the need to train safeguards inspectors in the performance and limita­tions of analytical instrumentation which they may encounter in the field. This may
include installed instrumentation for use by the IAEA or the facility operator. Such
training will alert the inspector to the critical steps of the analytical measurements
performed with this instrumentation and will provide additional assurance that the
results are sufficiently reliable for the purposes of the IAEA. Of course, training in
the use of these instruments should be under conditions and with samples that are
as realistic as possible.

The points described above have contributed to the justification for the estab­
lishment of the IAEA’s Safeguards Analytical Laboratory (SAL) in Seibersdorf,
which co-ordinates the activities of a network of 19 analytical laboratories in IAEA
Member States for the analysis of traditional safeguards samples and the provision
of reference and control materials for the relevant DA and NDA techniques. The
establishment of SAL and NWAL has been extensively described in the past [1–3].

Starting in 1991, the IAEA entered a transitional period in which the traditional
methods of implementing safeguards have been challenged. The inspections in Iraq,
under United Nations Security Council Resolutions 687, 707 and 715, have been
carried out by the IAEA Action Team with the support of many Member States, especially in the planning of inspection activities, the analysis of samples and the interpretation of the results. Ad hoc inspections in South Africa and the Democratic People’s Republic of Korea (DPRK) have been carried out to verify the correctness and completeness of the initial declarations provided by the State authorities; similar inspections are planned for other States entering into comprehensive Safeguards Agreements in the next few years (Argentina, Brazil, newly independent States of the former USSR). Finally, the IAEA recently took the unprecedented step of requesting a special inspection of two sites in the DPRK, on the basis of the analysis of ad hoc inspection samples as well as on third-party information.

The cases cited above have already resulted in the IAEA’s use of non-traditional sampling and measurement techniques, involving the services of laboratories outside of the established NWAL. IAEA staff have acquired experience in the collection of special types of samples and in the interpretation of non-routine analysis data. The IAEA has begun investigating how to make best use of existing analytical capabilities within the IAEA and how to expand and improve its ability to handle these new types of samples and analyses. In response to recommendations of the Standing Advisory Group on Safeguards Implementation (SAGSI) [4], the Department of Safeguards has initiated the ‘Programme 93+2’ to study a number of ways to strengthen IAEA safeguards and to make them more efficient and cost effective [5]. Two specific areas of this Programme are relevant to the IAEA’s needs for non-routine sampling and analysis methods: the use of environmental monitoring techniques for detecting undeclared nuclear activities, and the construction of a ‘clean’ laboratory for safeguards in Seibersdorf, to perform screening of samples, distribution to outside laboratories and quality assurance functions.

It is possible that the IAEA’s mission will change dramatically in forthcoming years, as a result of the NPT review conference in 1995, the offer of the USA to have the IAEA supervise a cut-off in the production of nuclear-weapons-usable material, and the progress towards a comprehensive test ban treaty (CTBT). Each of these developments represents a possible use of non-routine sampling and analysis techniques, which will necessarily require that the IAEA develop its in-house capabilities and that it make best use of existing expertise in the Member States and at the IAEA laboratories in Seibersdorf and Monaco.

2. ‘TRADITIONAL’ DESTRUCTIVE ANALYSIS CAPABILITIES

2.1. IAEA Safeguards Analytical Laboratory

2.1.1. Currently applied techniques

The basic techniques applied at SAL for the isotopic and elemental assays of fissile materials are similar to the ones that were already used in 1986 [3]. Significant
technological changes have, however, been introduced. All measurements are now automatized and computer controlled. The chemical titration procedures were scaled down to reduce the volume of liquid discards and their content in radioelements. The number of applications of radiometric methods has increased considerably. The power of the laboratory information system has been vastly increased.

(a) The New Brunswick Laboratory (NBL)/Davies and Gray titration is still the basic method for the determination of uranium in gram size samples of all types of unirradiated materials. The Mitsubishi robot arm, intended initially for automating the plutonium titration [3], was in fact used as the sample changer in a new and fully automated uranium titrator [6].

(b) The MacDonald/Savage titration has replaced the AgO/Fe(II) titration used earlier for the determination of the plutonium content in gram size samples of unirradiated nuclear materials. The original procedure was scaled down and was fully automatized with the assistance of experts from Dounreay [7] and Řež [8, 9]. The procedure is considerably more selective than the AgO/Fe(II) titration, but neptunium still interferes quantitatively.

(c) Chemical separations of plutonium and spent fuel samples prior to isotopic or isotope dilution analyses continue to be done by solvent extraction chromatography. The procedure was robotized (Fig. 1) with the assistance of the Transuranium Institute of the CEC [10].

(d) High resolution gamma spectrometry (HRGS) is used to screen all plutonium product samples as they are received. The plutonium content of samples containing neptunium will be analysed by isotope dilution rather than by titration. SAL has thereby acquired a considerable experience in isotopic analyses using the multipurpose gamma spectrometry analysis (MGA) programme [11].

(e) Energy dispersive X ray fluorescence (XRF) has become a very handy tool to determine low concentrations of actinides in scrap samples, insoluble residues or chromatographic eluates.

(f) An LKB α-β liquid scintillation spectrometer usually serves purposes similar to those of XRF. It is also capable of providing very accurate measurements, which have been used, for example, for the characterization of spike solutions applied in isotope dilution alpha spectrometric determination of plutonium in spent fuel solutions [12].

(g) Alpha spectrometry with SiLi or ion implanted detectors is used in parallel with mass spectrometry for the determination of the $^{238}$Pu abundance or for the isotope dilution analysis of plutonium in spent fuel samples [13]. The GRPANL software delivered by the Lawrence Livermore National Laboratory (LLNL) was used to determine $^{240}$Pu/$^{239}$Pu and $^{238}$Pu/$^{239}$Pu isotope ratios with accuracies of 0.5–3%, solely by alpha spectrometry of sources prepared by drop deposition and electrodeposition [14].

(h) Two multidetector thermal ionization spectrometers (FINNIGAN MAT 260-1), each equipped with nine Faraday cups, are operated at full capacity to
measure the isotope composition of all samples of nuclear materials submitted to SAL. A comprehensive new software was developed in co-operation with the Central Bureau of Nuclear Measurements (now the Institute of Reference Materials and Measurements) in Geel. This software includes routines for basic calibration steps, such as the cup linearity test, relative cup efficiency measurements, and a system calibration of the mass fractionation effects. Single-detector or multidetector measurements are possible. Total evaporation is used either for high accuracy measurements or for measuring samples with low or unknown amounts of actinides [15]. The latter method is easier and more robust than the procedure with internal normalization, which involves spiking with well known mixtures of $^{233}\text{U}/^{238}\text{U}$ or $^{242}\text{Pu}/^{244}\text{Pu}$ isotopes.

(i) Isotope dilution mass spectrometry with large size dried (LSD) spikes is now the normal procedure for the determination of the uranium and plutonium concentrations in spent fuel samples [16]. Batches of LSD spikes are prepared from certified reference materials either by SAL or by the Safeguards Analytical Laboratory of the Nuclear Material Control Centre in Tokai. The composition of these batches is therefore known with very good accuracy (Table I) and is verified by independent analyses in collaboration with the Seibersdorf Research Centre [17].
Table I. Certificate of LSD-Batch SAL-9979 (LSD-9)

Best estimates ± confidence limits for $\alpha = 0.05$
(Date of validity: 1994-04-01)

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uranium concentration (wt%)</td>
<td>8.2128±0.0013</td>
</tr>
<tr>
<td>Plutonium concentration (wt%)</td>
<td>0.41364±0.00016</td>
</tr>
<tr>
<td>$^{235}\text{U}$ abundance (wt%)</td>
<td>19.133±0.003</td>
</tr>
<tr>
<td>$^{239}\text{Pu}$ abundance (wt%)</td>
<td>97.924±0.002</td>
</tr>
<tr>
<td>$^{235}\text{U}/^{238}\text{U}$ atom ratio</td>
<td>0.24047±0.00006</td>
</tr>
<tr>
<td>$^{240}\text{Pu}/^{239}\text{Pu}$ atom ratio</td>
<td>0.020987±0.000020</td>
</tr>
</tbody>
</table>

(j) The stability of dried plutonium nitrate samples with time and their resistance to shocks during transport are of great concern because a large number of spikes and inspection samples containing 2–4 mg of plutonium per vial are handled every year. We have recently verified again that the drying procedure defined by Krtíl [17] yields dried salts of suitable resistance [18].

2.1.2. Techniques under development

Other techniques are investigated or being introduced at SAL, usually in the frame of Member States Support Programmes (MSSP), with the aim to improve or complement DA procedures already in use at SAL or to acquire expertise in techniques applicable for on-site measurements of high accuracy.

(a) A controlled potential coulometer is being built for SAL by the Savannah River Laboratory as a task of the United States Support Programme. The experimental aspects of coulometric analysis of 1–2 mg of plutonium are being investigated in Harwell at the request of SAL. The United Kingdom Support Programme also plans to develop for SAL a robot system which automatizes a procedure involving an anion exchange separation of 1–2 mg plutonium followed by controlled potential coulometry. The objective is to set up a procedure which would be free from neptunium or other interferences and which would yield accuracies of the order of 0.05 wt% with samples of small size.

(b) $\text{Pu(IV)}$ spectrophotometry is applicable to the determination of milligram amounts of plutonium in small samples of products with accuracies similar to those of titration [7]. This method is also used at reprocessing plants to measure the concentration of plutonium in high active liquid waste concentrates. It is a candidate procedure for use at an on-site laboratory. Therefore, it is considered to install it at SAL, in order to gain hands-on experience.
FIG. 2. Calibration curve of the XRF analysis of plutonium in U, Pu oxide.

(c) A hybrid K-edge densitometer (HKED) was installed at SAL in 1989 by Ottmar and his co-workers of the Joint Research Centre at Karlsruhe. Some initial problems with the fast pulse processors were resolved by increasing the size of the ground-lines. K-edge absorption and XRF measurements can be done with reproducibilities and accuracies of the order of 0.1% [19].

(d) A Philips XRF spectrometer is used in conjunction with a commercial high frequency furnace to develop a fast procedure for the analysis of plutonium and mixed uranium/plutonium oxides; such a device may be installed at inspected facilities or at a laboratory operated by the IAEA on an inspection site. About 0.3 g of samples is melted in a lithium borate flux. The melt is cast into a platinum dish. A borate disk of very homogeneous composition is produced. The concentrations of Th, U, Np, Pu and Am can be determined simultaneously by measuring the fluorescence of the Lα lines. The calibration curves are linear over at least a tenfold change in concentration (Fig. 2). A complete analysis can be performed in about 15 minutes, with a reproducibility of about 0.3% for the concentration of the major heavy elements [20].

2.2. The IAEA network of analytical laboratories

The Chalk River Nuclear Laboratories (Atomic Energy of Canada Ltd) were the only addition to the IAEA NWAL since 1986 (Fig. 3). They are used like the Physics Research Centre of Budapest (PRPI) for the analysis of heavy water samples.
The isotopic abundance of these samples can be measured by density or mass spectrometry.

The laboratories in Berlin, Dresden, Harwell, Petten, Prague, Seibersdorf and St. Petersburg provide or provided mostly isotope dilution analyses of spent fuel samples, mass spectrometric analyses, titration or coulometric analyses of uranium samples. With the reduction of the national nuclear programmes, only Harwell, Prague and St. Petersburg have retained the capability to analyse samples of plutonium products. After the reunification of Germany, Dresden decided to terminate its participation in NWAL.

Oak Ridge National Laboratory (ORNL) is called upon occasionally to perform mass spectrometric analyses of resin bead samples containing nanogram amounts (or less) of actinide using ion counting detection.

A close co-operation has developed between IAEA-SAL and the Safeguards Analytical Laboratory of the Nuclear Material Control Centre (NMCC-SAL) in Tokai-mura, in the preparation, validation and implementation of LSD spikes for isotope dilution analyses of spent fuel solution.

Interlaboratory experiments have been carried out with the Centre d'études de l'énergie nucléaire, Mol, Eurex (Saluggia) and ITREC (Bari) for checking the conditions of transport and analysis of plutonium product samples. The Bhabha Atomic Research Centre and the laboratories in Harwell, Petten, Seibersdorf and St. Petersburg took part in a test of isotope dilution alpha spectrometry of plutonium in spent fuel solutions [21].
The laboratory at Harwell, the NBL, the Institute for Reference Materials and Measurement (IRMM) in Geel, the CEA in Fontenay-aux-Roses and the Khlopin Radium Institute (KRI) in St. Petersburg supply SAL and NWAL with essentially all reference materials needed for the calibration or the testing of DA procedures.

2.3. Analytical capacity and performance

In 1986, it was still anticipated that the number of samples of nuclear material submitted annually could grow to more than 2000 in 1990. The demand remained below this level (Fig. 4) and could be accommodated up to now, despite the complications resulting from the large annual and seasonal fluctuations in the load. The

FIG. 4. Samples of nuclear materials received annually from 1983 to 1992 (excluding samples taken by the IAEA Action Team).

FIG. 5. Total verification times for DA (based on median values).
budgetary freeze during the last ten years and the 12% deferment of the financial resources since the formation of the newly independent States of the former USSR have further complicated the utilization of the services of NWAL.

Manpower resources have been strained even more by the aftermath of the Gulf war. As a consequence, the response times, after reaching a minimum in 1989–1990, have started to rise again (Fig. 5).

**TABLE II. REPEATABILITY OF DUPLICATE ANALYSES**

<table>
<thead>
<tr>
<th>Measurement</th>
<th>Material</th>
<th>Number of samples</th>
<th>SD (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{235}$U abundance</td>
<td>U oxide</td>
<td>22</td>
<td>0.113</td>
</tr>
<tr>
<td></td>
<td>U, Pu oxide</td>
<td>2</td>
<td>0.030</td>
</tr>
<tr>
<td>$^{239}$Pu abundance</td>
<td>Pu oxide</td>
<td>9</td>
<td>0.027</td>
</tr>
<tr>
<td></td>
<td>U, Pu oxide</td>
<td>3</td>
<td>0.017</td>
</tr>
<tr>
<td></td>
<td>Spent fuel solution</td>
<td>23</td>
<td>0.005</td>
</tr>
<tr>
<td>U titration</td>
<td>U oxide</td>
<td>37</td>
<td>0.028</td>
</tr>
<tr>
<td></td>
<td>U, Pu oxide</td>
<td>3</td>
<td>0.040</td>
</tr>
<tr>
<td>Pu titration</td>
<td>Pu oxide</td>
<td>9</td>
<td>0.053</td>
</tr>
<tr>
<td></td>
<td>U, Pu oxide</td>
<td>3</td>
<td>0.029</td>
</tr>
<tr>
<td>U IDMS</td>
<td>Spent fuel</td>
<td>5</td>
<td>0.039</td>
</tr>
<tr>
<td>Pu IDMS</td>
<td>Spent fuel</td>
<td>21</td>
<td>0.043</td>
</tr>
</tbody>
</table>

**TABLE III. DETERMINATION OF $^{238}$Pu ABUNDANCES**

Relative differences between alpha spectrometry and mass spectrometry

<table>
<thead>
<tr>
<th>Material</th>
<th>Number of samples</th>
<th>Mean of relative difference (%)</th>
<th>Standard deviation of relative difference (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\text{PuO}_2$</td>
<td>18</td>
<td>0.059</td>
<td>0.35</td>
</tr>
<tr>
<td>U, $\text{PuO}_2$</td>
<td>6</td>
<td>−1.80</td>
<td>2.2</td>
</tr>
<tr>
<td>Spent fuel solution</td>
<td>27</td>
<td>−0.28</td>
<td>0.32</td>
</tr>
</tbody>
</table>
TABLE IV. ISOTOPE DILUTION MASS SPECTROMETRY WITH LSD SPIKES
Results of procedure control measurements in 1991

<table>
<thead>
<tr>
<th>Elements</th>
<th>U</th>
<th>Pu</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of Measurements</td>
<td>24</td>
<td>24</td>
</tr>
<tr>
<td>Treatments</td>
<td>20</td>
<td>20</td>
</tr>
<tr>
<td>Mixtures</td>
<td>12</td>
<td>12</td>
</tr>
<tr>
<td>Control batches</td>
<td>3</td>
<td>3</td>
</tr>
</tbody>
</table>

Coefficient of variation (%)

<table>
<thead>
<tr>
<th></th>
<th>Measurement</th>
<th>Treatment</th>
<th>Mixture</th>
<th>Control batch</th>
<th>Total uncertainties</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>0.19</td>
<td>a</td>
<td>a</td>
<td>a</td>
<td>0.19</td>
</tr>
<tr>
<td>Treatment</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.10</td>
</tr>
<tr>
<td>Mixture</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.31</td>
</tr>
<tr>
<td>Control batch</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.19</td>
</tr>
<tr>
<td>Total uncertainties</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.32</td>
</tr>
</tbody>
</table>

Average difference (%) 0.04 -0.10

a Negative variance estimate.

On the other hand, the quality of the analyses could be maintained or even improved. Examples of the repeatability of duplicate analyses done in January and February 1994 are given in Table II. The good agreement between the values obtained by alpha spectrometry and mass spectrometry for the $^{238}$Pu abundances (Table III) documents the good quality of the separation by solvent extraction chromatography. Control samples of known composition are run in parallel with the inspection samples. Table IV summarizes the results of the analyses of control samples in isotope dilution mass spectrometry of spent fuel samples carried out by SAL in 1991. In the determination of the uranium concentration, the final mass spectrometric measurement was the only significant source of uncertainties. For plutonium, the major uncertainty appeared to be in the preparation or characterization of individual batches of control samples. A well characterized plutonium oxide with low $^{239}$Pu abundance would be needed for preparing the control samples.

Figure 6 shows how the introduction of the LSD spikes after the 14th campaign greatly decreased the fluctuations of the operator-inspector differences from one
TABLE V. STANDARD DEVIATIONS OF MUF (SDMUF)
International Standard of Accountancy (S(e)) and actual performances

<table>
<thead>
<tr>
<th>Type of plant</th>
<th>International Standard S(e) (%)</th>
<th>SDMUF/throughput (%)</th>
<th>M/SQ$^a$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reprocessing plant</td>
<td>1.0</td>
<td>0.45</td>
<td>0.13</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.55</td>
<td>0.72</td>
</tr>
<tr>
<td>Plutonium fuel fabrication plant</td>
<td>0.50</td>
<td>0.54</td>
<td>0.59</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.13</td>
<td>0.49</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.20</td>
<td>2.04</td>
</tr>
</tbody>
</table>

$^a$ M is the limit of confidence in the estimate of the MUF difference with a 90% probability level.

reprocessing campaign to the next one. It is also rewarding to observe that the standard deviation of the material unaccounted for (MUF) difference at sensitive plants remains consistent with the international standard of accountability and that the verification system is capable to detect the missing of less than one to two significant quantities (SQ) with a high probability, even at relatively large throughputs (Table V).
3. EXPANDED ANALYTICAL CAPABILITIES FOR SAFEGUARDS

3.1. IAEA laboratories

The IAEA operates a number of laboratories under the Department of Research and Isotopes, with different programmatic missions and sources of funding. At present, only SAL is directly funded by the Department of Safeguards for routine analytical services. However, in the case of expanded use of environmental or other non-routine samples for safeguards, there exist relevant expertise and analytical capacity in the following IAEA laboratories: the Physics, Chemistry and Instrumentation (PCI) laboratory units in Seibersdorf [22]; the Isotope Hydrology Laboratory of the PCI in the Vienna International Centre [22], and the Marine Environment Laboratory in Monaco [23].

In addition, the IAEA is currently planning a clean laboratory for safeguards in Seibersdorf, for which a Member State made an extra-budgetary contribution. The analytical techniques relevant to non-routine samples which are employed (or planned) in each of these laboratories are discussed below.

3.2. Activities at SAL

SAL employs a number of analytical techniques for routine inspection samples; many of these techniques can be adapted to specialized measurements on non-routine samples. For example, wavelength dispersive XRF analysis has been used successfully for a range of samples from the IAEA Action Team. This technique has been used to identify specific alloys of steel (350 maraging steel) that can be used to construct centrifuge rotors for uranium isotopic enrichment [24, 25]. This method could also be used to measure the trace elements in uranium compounds such as yellow cake or UO₂ as a means of tracing the origin of the material. The concentration of uranium in phosphate ores or by-products of phosphate fertilizer production has also been measured by XRF at SAL. Impurities in nuclear fuel or structural materials can be analysed by emission spectroscopy.

The techniques of gamma spectrometry and alpha spectrometry are routinely used at SAL for nuclear material samples, but they can also be used to detect the presence of fission or activation products such as ¹³⁴Cs, ¹³⁷Cs or ⁶⁰Co in environmental samples. These signatures may reveal the existence of undeclared reactor or reprocessing operations. Alpha spectrometry was used to detect ²¹⁰Po on smear samples from Iraq, suggesting the existence of research on neutron generators for initiating a nuclear explosion [24, 25]. Low level emissions of alpha emitting actinides can be measured with good selectivity by liquid scintillation spectrometry with time resolution. Thermal ionization mass spectrometry (TIMS) was used to measure the isotopic composition of U and Pu on smear samples which contained only nanogram to microgram amounts of these elements. The isotopic analyses of...
U and Pu give highly valuable information about their origin and the processes which produced them.

3.3. Planned activities for a clean laboratory for safeguards

Experience with samples from Iraq alerted the IAEA to the need to ensure that environmental samples are not contaminated during sampling, transport or handling at the receiving laboratory. It was also seen that the achievable blank levels at SAL for the measurement of U and Pu were above those expected for certain environmental sample types. Therefore, on the basis of the recommendations of outside experts [26, 27] it was agreed that the IAEA should have a clean laboratory for handling and screening of environmental samples. The planning for such a facility has started and funds have been made available from one Member State to cover a significant part of the projected cost. The technical specifications for this laboratory are listed below:

- 230 m² of laboratory space inside a building of approximately 500 m².
- Modular clean-room installations with approximately 100 m² at Class-100;
- Laminar flow fume cupboards and work surfaces at Class-10 conditions;
- Four separate chemistry laboratories for treatment of different environmental sample types (smears, vegetation, water, soil and sediment);
- Sample pre-screening with gamma spectrometry, energy dispersive XRF spectrometry and total alpha counting;
- TIMS with ion counting detection, with detection limits of about \(1 \times 10^7\) atoms of U or Pu (4 fg);
- Clean areas for preparation of sampling materials or splitting of samples.
- Possible future expansion of the clean laboratory's capabilities to install a scanning electron microscope with electron microprobe attachment and an inductively coupled plasma mass spectrometer for ultra-sensitive analysis of small samples.

The tentative schedule for this laboratory calls for construction of the outer building to be complete in 1995, with installation and commissioning of the clean-room components and analytical equipment to be finished in late 1995/early 1996.

3.4. Activities at the PCI laboratory units

The PCI laboratory at Seibersdorf supports a wide range of activities related to the technical co-operation programmes of the IAEA, including co-ordinated research projects, analytical quality control services and training of visiting scientists from developing countries. PCI contributes also to environmental monitoring projects sponsored by other international organizations such as WMO and WHO.
3.4.1. Chemistry and Instrumentation Units

The Chemistry and Instrumentation Units of PCI at Seibersdorf employ a variety of analytical methods, which are summarized below:

— Neutron activation analysis (NAA) for sensitive and accurate determination of a large number of elements in complicated matrices, such as soil, vegetation or animal tissue;
— Gamma and alpha spectrometry for the measurement of actinides, fission or activation products or other radionuclides in environmental samples, as demonstrated in the International Chernobyl Project [28];
— Electrical conductivity and pH measurements of natural water samples;
— Inductively coupled plasma atomic emission spectrometry for the determination of trace elements in water or other environmental samples, as demonstrated in studies on heavy metal pollution in the food chain;
— Laser induced fluorimetry for the determination of U with high selectivity and sensitivity;
— Energy dispersive XRF analysis using total reflectance for higher sensitivity or a microfluorescence probe for high spatial resolution in the determination of trace elements in geological or metallurgical samples.

3.4.2. Isotope Hydrology Laboratory

The Isotope Hydrology Laboratory in the Vienna International Centre (VIC) measures naturally occurring isotopes of hydrogen, carbon, oxygen and sulphur, and trace components in water or geological samples. The techniques used in this laboratory involve:

— Liquid chromatography and atomic absorption spectrophotometry to measure the major ions or trace elements in water samples derived from surface water or groundwater, from precipitation or from water entrained in geological matrices;
— Gas proportional counting of $^3$H (following electrolytic enrichment) or of $^{14}$C in water or geological samples;
— Mass spectrometry to measure the isotopic composition of hydrogen, carbon, oxygen or sulphur in water or geological samples (the isotope ratios measured are $^2$H/$^1$H, $^{13}$C/$^{12}$C, $^{18}$O/$^{16}$O and $^{34}$S/$^{32}$S);

The isotopes $^{12}$C, $^{13}$C and $^{14}$C can also be measured at an accelerator mass spectrometry facility in a Member State, following sample preparation by the IAEA.
3.5. Activities at the Marine Environment Laboratory in Monaco

The Marine Environment Laboratory (MEL) measures a variety of radioisotopes and trace elements in samples of marine or estuarine origin. The main techniques employed at MEL are described below:

(a) Alpha spectrometry using Si surface barrier detectors is used to measure alpha particle emitting nuclides such as $^{238}\text{Pu}$ and $^{210}\text{Po}$;
(b) Gamma spectrometry using high purity Ge, Ge(Li) and NaI(Tl) detectors is applied to the measurement of fission and activation products such as $^{60}\text{Co}$, $^{95}\text{Zr}$, $^{134}\text{Cs}$, $^{137}\text{Cs}$, $^{106}\text{Ru}$;
(c) Beta spectrometry using Si(Li) and Si(Au) detectors can measure beta emitting isotopes such as $^{90}\text{Sr}$, $^{99}\text{Tc}$ and $^{147}\text{Pm}$. Liquid scintillation counting is employed to measure both alpha and beta emitters and is especially useful for measuring $^3\text{H}$ and $^{14}\text{C}$ in liquid samples;
(d) Gas proportional counters are used to measure the isotopes $^3\text{H}$, $^{14}\text{C}$, $^{85}\text{Kr}$ and $^{133}\text{Xe}$ in gaseous samples;
(e) Gas source mass spectrometry is also employed to measure the $^3\text{He}$ daughter of $^3\text{H}$ at very low levels;
(f) Inductively coupled plasma mass spectrometry with electro-thermal vaporization of the sample provides a highly sensitive measurement capability for most elements of the periodic table and especially for the actinide elements.

3.6. Activities at Member States laboratories

In the three years since the first IAEA Action Team inspections in Iraq, the IAEA has been offered the services of highly specialized laboratories in several Member States for the measurement of environmental and special nuclear samples resulting from the Action Team inspections as well as from ad hoc inspections. Examples of such laboratories are given below:

(a) ORNL co-ordinates a network of laboratories in the US Department of Energy (USDOE) complex. These laboratories are capable of measuring the content and isotopic composition of actinide elements in environmental samples down to the sub-picogram level, using chemical concentration techniques and isotope dilution mass spectrometry.
(b) The CEC Institute for Transuranium Elements in Karlsruhe applies a number of sensitive analysis techniques to the measurement of actinide elements and other radionuclides. These techniques include thermal ionization mass spectrometry and inductively coupled plasma mass spectrometry together with high pressure liquid chromatography.
(c) The CEC IRMM in Geel uses a combination of gas source mass spectrometry and thermal ionization mass spectrometry to measure the minor isotopes of
uranium with extremely high precision and accuracy. This information can be used to trace the origin of natural uranium.

Similar analytical capabilities have been made available by other organizations and Member States.

As the Programme 93+2 proceeds with field trials of environmental monitoring for safeguards, the IAEA will continue to seek the participation of Member States laboratories in an expanded NWAL. Procedures are being implemented to provide quality assurance audits of candidate laboratories for inclusion in the system. The IAEA will also need reference materials and realistic control samples, which could be used to check the performance of a given laboratory.

4. CONCLUSIONS

A very large fraction of the independent measurements needed for the verification of the accountability of over 100 000 t of nuclear materials currently under IAEA safeguards is nowadays made directly in the field using NDA techniques. Samples must still be taken and subjected to DA if on-site measurements are not yet possible. Otherwise, DA is performed in combination with NDA measurements in order to achieve the required level of probability for the detection of smaller potential bias defects, to verify the closing of material balances, to calibrate [29] or evaluate field measurements and to document their accuracy, and to verify the quality of operators' measurement systems.

SAL and NWAL receive and analyse annually up to 1600 samples of nuclear or source materials originating from up to 90 different facilities or material balance areas in 29 countries [30]. These laboratories provide the IAEA with the capability to determine the concentration and isotopic composition of uranium, plutonium or thorium with high accuracies in any nuclear material encountered in the various nuclear fuel cycles. Americium-241 is measured in all plutonium compounds to support NDA or to determine their 'age'. Analytical methods are available for the analyses of heavy water, light water and graphite moderators. Impurities and corrosion or fission products may be determined in any of the above materials or cooling waters.

The accuracy of DA determines how sensitive the safeguards verifications can be in detecting the potential missing of SQs for safeguards purposes. This is why great efforts are made to monitor closely the quality of the analytical measurements [31, 32] and to strive to achieve even better accuracies in co-operation with the plant analysts, particularly in connection with the verification of large flows and inventories of plutonium and highly enriched uranium materials. The level of agreement between the results of the operator and the inspector can be of the order of 0.05% or less for the concentrations of fissile elements in spent fuel input solutions, \( \text{PuO}_2 \)
and uranium products [33]. The use of LSD spikes prepared from certified reference materials has greatly simplified the preparation of spent fuel samples before their shipment to SAL, with a drastic improvement in the verification of their plutonium concentration [16]. It is pointed out to inspectors and plant operators that very careful attention to sampling [34] and sample handling [18] is needed. The IAEA receives an invaluable assistance from its Member States in improving its DA systems, in particular through some 50 support programme tasks in recent years [35-37]. Automatic devices and robots were installed, and they were taught to handle and analyse DA samples as reliably and accurately as the most experienced analysts, with lower amounts of analytes and lower volumes of effluents [6, 9, 10]. Expertise in the capabilities and use of very accurate X ray techniques has been growing since they were introduced at SAL [19, 20] and applied in the field. Dependable sources of supply of highly accurately certified reference materials are developing, again with the help of the MSSPs, the NBL (USDOE), the IRMM (Geel) [38, 39], AEA Technology (Harwell), KRI (St. Petersburg) and the Commission d’établissement des méthodes d’analyses (Paris).

Although success has been achieved in keeping acceptable response times in the above analytical services, the IAEA has recognized, as has Euratom, the need and benefit of on-site analytical laboratories at bulk handling facilities processing large amounts of plutonium [40-42], to complement precise NDA measurement capabilities [43].

Up to now, the traditional accountability verification systems have ensured an effective control over nuclear and other materials placed under IAEA safeguards. Recent experiences, however, have highlighted the limits of the existing systems. The IAEA and its Member States are therefore stepping up their efforts to enhance the ability of the IAEA to confirm that States having entered a comprehensive Safeguards Agreement do not engage in undeclared nuclear activities. It has been pointed out that the IAEA should develop means of detecting minute amounts of specific isotopes, elements or compounds that would be released by undeclared activities into declared materials or processes, in effluents or in the environment [26].

Existing capabilities at the IAEA laboratories at Seibersdorf have already been used to confirm the suspected existence of clandestine uranium enrichment and plutonium separation experiments, using low level alpha and gamma spectrometry, neutron activation analyses, XRF and isotope dilution mass spectrometry [24, 25]. Thus, the IAEA was able to corroborate the results of analyses made in specialized laboratories of its Member States on smears, air filters, wastes, vegetation, waters and other special samples taken during ad-hoc inspections. Also participating in the analysis of special samples coming from the UNSC 687 Action Team or ad hoc inspection activities were the Isotope Hydrology Unit [44] and the Chemistry Unit of the PCI of the IAEA laboratories. Additional analytical resources
for future applications of environmental monitoring for safeguards exist at the MEL in Monaco.

This experience indicated what strict precautions must be followed when taking, handling and analysing such samples for sub-nanogram to femtogram levels of actinides and other isotopic signatures [27]. At the planned clean laboratory in Seibersdorf, special samples would be received, screened and analysed without risk of contamination. This ‘clean laboratory for safeguards’ will play a role similar to that of SAL in the traditional DA verification of nuclear materials accountability. Special samples, taken to detect potential undeclared activities, will be redispached from this laboratory to specialized laboratories of the IAEA and its Member States. Thus, the IAEA will have the necessary independence to ensure the quality of the numerous and delicate steps involved in the sophisticated analytical services which will be requested from an expanded NWAL.

ACKNOWLEDGEMENTS

The authors wish to acknowledge the valuable contribution to the work described in this paper by their colleagues of SAL, NWAL and other IAEA and Member States laboratories, by the personnel who co-ordinate the Member States Support Programmes involved and by the inspectors of the IAEA. The collaboration of all of these persons remains essential to ensure the success of the follow-up activities and to attain the objectives summarized in the conclusion.

REFERENCES


INTERNATIONAL STANDARDS, TRACEABILITY AND QUALITY OF SAFEGUARDS ACCOUNTABILITY MEASUREMENTS

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Abstract

INTERNATIONAL STANDARDS, TRACEABILITY AND QUALITY OF SAFEGUARDS ACCOUNTABILITY MEASUREMENTS.

In order to be able to compare operators’ and inspectors’ measurements, it is necessary to demonstrate that they are performed in compliance with the International System of Units (SI). Several links still need to be developed for tracing effectively all these measurements back to the relevant base units and the derived SI units. The certified uranium isotopic reference materials recently issued by the Institute for Reference Materials and Measurements (IRMM) are an example of how organizations such as the IRMM assist measurement specialists in achieving international comparability of measurements for international safeguards. The paper describes briefly the “1993 International Target Values (ITVs) for Uncertainty Components in Fissile Isotope and Element Accountancy for Effective Safeguards of Nuclear Materials”, based on the model of the 1987 and 1988 ESARDA Target Values, and issued by the IAEA. The IAEA intends to use the 1993 ITVs as the current international standards of measurements foreseen in the Safeguards Agreements concluded in the frame of the Treaty on the Non-Proliferation of Nuclear Weapons. The quality currently achievable in safeguards measurements is illustrated by the results reported in the Regular European Inter-laboratory Measurement Evaluation Programme of the IRMM and by the results of statistical evaluations of several tens of thousands of actual safeguards inspection data at the IAEA. The progress made in recent years and the need to enforce effective procedures of measurement quality control in order to obtain the full benefit of present analytical capabilities are pointed out.
1. INTRODUCTION

Verification of the accountability of fissile materials is and will remain a fundamental element of national and international safeguards systems. Therefore, safeguards inspectors perform independent analyses of the safeguarded materials and compare their results with the values declared by the State. The latter values are derived from the results of measurements of the same materials by the plant laboratories. It is important to remember that the operators’ and inspectors’ measurements are comparable only if they are made in a common and consistent system of measurements. In addition, Safeguards Agreements with the IAEA in the frame of the Non-Proliferation Treaty (NPT) prescribe that “the system of measurements on which the records used for the preparation of the ‘fissile material accountability’ reports are based shall either conform to the latest standards or be equivalent in quality to such standards” [1]. There must also be means to assess how well the actual operators’ and inspectors’ accountability measurements conform to the expected standards. The purpose of the present paper is to draw attention to the importance of the above requirements, to illustrate their correlations and to examine how far they are fulfilled.

2. TRACEABILITY OF ACCOUNTABILITY MEASUREMENTS

The measurements of the mass or volume of the material, the concentrations of the fissile elements and the abundance of the fissile isotopes performed by operators and inspectors are comparable only if they are done in a common and consistent measurement system, which must be the International System of Units (SI). These measurements must be ‘traceable’ to the base SI quantities and the relevant SI units of measurement. However, most chemists continue to measure element concentrations in g/kg, in weight % or mass %, although the relevant SI units are mol/kg or amount % [2], while international fissile material control is actually concerned with the number of atoms of fissile isotopes rather than with their mass.

Several links required to trace our analytical measurements back to the SI system are already well established; others still need to be developed [3, 4]. Certified reference materials (CRMs) of highly pure metals are available from the New Brunswick Laboratory (NBL, now at Argonne National Laboratory, IL) and from the Commission d’établissement des méthodes d’analyse (CETAMA), Fontenay-aux-Roses, for the calibration of methods of chemical assay for uranium and plutonium, but the atomic weight of the elements is not certified to better than \( \pm 0.5 \times 10^{-4} \) in these materials, thus putting a limit to the conversion of mass (kg) to amount (mol). Also, unfortunately, these materials are not readily available to all analysts throughout the world. Alternative reference materials may be more accessible, but they are certified with larger uncertainties and there is not yet an international system of intercomparison of these materials with the higher quality CRMs.
FIG. 1. Traceability of IRMM uranium reference materials.
Part of the tasks of the Institute for Reference Materials and Measurements (IRMM) is to provide the missing links with the SI system, using SI units. The preparation and certification of the IRMM uranium isotopic CRMs is a significant example. Three separate sets of CRMs have been made available for the calibration of ‘field’ isotopic analyses (Fig. 1). One set (EC 171) consists of five units, containing about 200 g of U3O8 oxide with 235U isotope abundances of 0.3, 0.7, 2.0, 3.0 and 4.5%, respectively, each unit being in a sealed aluminium can of calibrated dimensions. This set is intended for the calibration of non-destructive isotope abundance measurements. The same materials are available in smaller units of 0.5 g each (in dissolved form) for calibrating thermal ionization mass spectrometric measurements (IRMM 183–187). The two sets are identical, within the certified uncertainties. Samples from these materials were converted to UF6, and their isotopic composition was characterized by high accuracy gas source mass spectrometry against synthetic mixtures of enriched 235U and 238U isotopes prepared on a gravimetric basis and with a 235U/238U molar ratio known to ±0.01%. In this way, the 235U and 238U isotope ratios in sets EC 171 and IRMM 183–187 were linked directly with molar ratios at an overall uncertainty of 0.03% (2σ). Similarly, low enriched UF6 CRMs (set IRMM 021–027) were characterized against similar synthetic mixtures at an uncertainty of within 0.03% (2σ). With these three CRM sets it is now possible to achieve international comparability of 235U isotopic measurements regardless of the method — non-destructive techniques, thermal ionization or gas source mass spectrometry — and in the most direct manner and with minimum uncertainties.

CRMs of these types are very precious and must be used sparingly. This is why many laboratories use CRMs only for major method calibrations or validations. Working reference materials, also sometimes called in-house or in-plant reference materials, are prepared for use in daily calibrations or quality control measurements. This is wise and perfectly legitimate, provided that these intermediate reference materials are effectively traceable to the CRMs and linked with the SI system. However, each intermediate ‘link’ adds an additional uncertainty to the process.

In summary, the Avogadro constant, the atomic weights, which link mass with amount of substance, the nuclear constants, and the chemical and isotopic reference materials constitute necessary links which allow us to trace the results of analyses back to the SI system. The strength of these links is determined by the uncertainties in the data which characterize them.

3. INTERNATIONAL STANDARDS OF MEASUREMENTS

The uncertainties in the field measurements themselves are another fundamental parameter of the traceability system. This is why NPT Safeguards Agreements stress the need to ensure that these measurements “conform to international standards”.

1993 International target values for uncertainty components in measurements of amount of nuclear material for safeguards purposes (Table 1.11) (% relative standard deviations)

Material type: LWR MOX (<10% Pu)

<table>
<thead>
<tr>
<th>Measurement method</th>
<th>Bulk</th>
<th>Sampling</th>
<th>Pu Conc.</th>
<th>Pu Total</th>
<th>Notes</th>
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<td>BIF</td>
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<table>
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<th>Measurement method</th>
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<th>235U Total</th>
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</tr>
</tbody>
</table>

RAN: Relative standard deviation of the repeatability of measurement of a single laboratory within one inspection;  
BIF: Relative standard deviation of the between-inspection uncertainty component for a single laboratory.

Notes:  
a Equivalent performance expected for coulometric procedure instead of potentiometric titration.  
b Measurement time 300 s; with mass spectrometric isotopic analysis.  
c Better measurement performance to be expected for material in standardized containers.

FIG. 2. 1993 ITV Table for LWR-MOX.

The IAEA defined in the 1970s international standards of nuclear material accountability [5], which describe the uncertainties expected in the annual closing of the balance of fissile materials at five types of bulk handling nuclear plants. The concept of target values for uncertainty components of the analytical measurements, proposed by De Bièvre and accepted by ESARDA after extensive discussions [6, 7], is, however, a more practical tool in the analytical laboratory. The IAEA therefore
conducted extensive consultations in 1991–1993 with experts from four continents and several national and international technical panels, including ESARDA/WGDA, ESARDA/WGNDA, INMM Subcommittee 5.1, ISO/TC85/SC5/WG3 and the Japanese Ad Hoc Technical Group on International Standards of Measurements. This resulted in the "1993 International Target Values (ITVs) for Uncertainty Components in Fissile Isotope and Element Accountancy for Effective Safeguards of Nuclear Materials", issued by the IAEA [8]. The ITVs describe, according to the model of the ESARDA Target Values, the uncertainty components which are expected to be achievable under current industrial and inspection conditions in fissile element and isotope accountability measurements. An important change is, however, worth noting: the ESARDA/WGDA considered only destructive analytical (DA) measurements and characterized their expected quality by separate relative standard deviations for uncertainties of random nature and uncertainties of systematic nature. The 1993 ITVs substitute for these figures the relative standard deviations expected for the within- and between-inspection uncertainty components. This is because the latter can be directly compared with the estimates of the uncertainties derived from the statistical evaluations of the differences observed between the value declared by the operator and the independent measurement obtained by the inspector on the same material. The 1993 ITVs include a separate table for each of the sixteen different materials of greatest significance in safeguards accountability verifications, namely low enriched UF₆, oxide powders and pellets, U scrap, fuel rods and assemblies, U and Pu nitrate solution products, LWR reprocessing spent fuel solutions, Pu oxide, FBR-MOX and LWR-MOX, MOX scrap, highly enriched uranium (HEU) metal and U/Al alloys.

Separate ITVs are defined for the major measurement steps, i.e. bulk volume or mass measurements, sampling, isotope and element assays. These components are combined and used to define the ITVs for uncertainties in the determinations of the amounts of fissile elements and isotopes, which are the quantities subject to verification. Figure 2 gives as an example Table 1.11 of the ITVs applicable to industrial LWR-MOX. The 1993 ITVs incorporate uncertainties for DA measurements and also for non-destructive analytical (NDA) measurements. Compared with the 1987 and 1988 ESARDA Target Values, the 1993 ITVs illustrate also the progress made particularly in mass spectrometric measurements of isotope abundances or element assays, and in bulk volume measurements.

4. QUALITY OF SAFEGUARDS ACCOUNTABILITY MEASUREMENTS

The quality achievable in safeguards measurements may be illustrated by the results reported in measurement evaluation programmes [9, 10] or by the results of actual accountability data and verification measurements reported to safeguards inspectorates.
4.1. The Regular European Interlaboratory Measurement Evaluation Programme

The Regular European Interlaboratory Measurement Evaluation Programme (REIMEP) is organized and carried out by IRMM in order to assist the Euratom safeguards authorities (CEC/DG XVII-E4) \[10, 11\]. Since its start in 1986, REIMEP has covered the key materials in the nuclear fuel cycle, namely UF$_6$, UO$_2$ powders and pellets, uranyl nitrate solutions, plutonium nitrate solutions, spent fuel input solutions, PuO$_2$ powders and MOX pellets. Participation in this programme is voluntary. Participants receive samples of homogeneous materials, prepared, bottled and characterized by IRMM. They are asked to use the analytical method of their choice to perform isotopic and elemental analyses of uranium, plutonium and americium, as applicable. The reference values and their uncertainties, established by IRMM, are not disclosed before the participants have reported their results. Full confidentiality is guaranteed.

IRMM presents the coded results in graphs, which help to identify significant systematic errors (Fig. 3). The DoD method \[12\] is used to estimate the interlaboratory standard deviation without rejecting any outlier. This estimate can serve as a measure of the state of the practice for this particular nuclear analytical measurement. IRMM has now derived such 'performance values' for all uranium \[10\] and plutonium materials mentioned above.

**FIG. 3. REIMEP results of determinations of $^{235}$U abundance in a UO$_2$ powder.**
### TABLE I. DESTRUCTIVE ANALYSIS VERIFICATION MEASUREMENT PERFORMANCE FOR ELEMENT AND ISOTOPE AMOUNTS

<table>
<thead>
<tr>
<th>Material type</th>
<th>Measurement</th>
<th>Standard deviations of operator–inspector differences (% rel.)</th>
<th>Within inspection</th>
<th>Between inspections</th>
</tr>
</thead>
<tbody>
<tr>
<td>Enriched UF$_6$</td>
<td>$^{235}$U total</td>
<td>0.16</td>
<td>0.45</td>
<td>0.37</td>
</tr>
<tr>
<td>U oxide powder</td>
<td>$^{235}$U total</td>
<td>0.18</td>
<td>4.62</td>
<td>0.71</td>
</tr>
<tr>
<td>U oxide pellets</td>
<td>$^{235}$U total</td>
<td>0.14</td>
<td>0.82</td>
<td>0.44</td>
</tr>
<tr>
<td>U scrap/sludge</td>
<td>$^{235}$U total</td>
<td>0.04</td>
<td>50.93</td>
<td>15.87</td>
</tr>
<tr>
<td>Dissolver solution</td>
<td>Pu total</td>
<td>0.42</td>
<td>0.92</td>
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<td>Pu total</td>
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<td>Pu total</td>
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<tr>
<td>MOX scrap</td>
<td>Pu total</td>
<td>0.42</td>
<td>0.92</td>
<td>0.75</td>
</tr>
</tbody>
</table>
### TABLE II. NON-DESTRUCTIVE ANALYSIS VERIFICATION MEASUREMENT PERFORMANCE FOR ELEMENT AND ISOTOPE AMOUNTS

<table>
<thead>
<tr>
<th>Material type</th>
<th>Instrument</th>
<th>Measurement</th>
<th>Standard deviations of operator–inspector differences (% rel.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>U fuel rods</td>
<td>FRSC</td>
<td>$^{235}$U total</td>
<td>0.90</td>
</tr>
<tr>
<td>U fuel assemblies</td>
<td>UNCL</td>
<td>$^{235}$U total</td>
<td>2.53</td>
</tr>
<tr>
<td>PuO$_2$ powder</td>
<td>HLNC</td>
<td>Pu total</td>
<td>0.65</td>
</tr>
<tr>
<td>MOX $&gt;$10% Pu FBR-MOX</td>
<td>HLNC</td>
<td>Pu total</td>
<td>1.29</td>
</tr>
<tr>
<td>MOX $&lt;$10% Pu LWR-MOX</td>
<td>INVS</td>
<td>Pu total</td>
<td>0.62</td>
</tr>
<tr>
<td>MOX scrap</td>
<td>HLNC</td>
<td>Pu total</td>
<td>3.21</td>
</tr>
<tr>
<td>MOX rods</td>
<td>INVS</td>
<td>Pu total</td>
<td>1.97</td>
</tr>
<tr>
<td>MOX assemblies</td>
<td>HLNC</td>
<td>Pu total</td>
<td>1.74</td>
</tr>
<tr>
<td></td>
<td>INVS</td>
<td>Pu total</td>
<td>3.40</td>
</tr>
<tr>
<td></td>
<td>HLNC</td>
<td>Pu total</td>
<td>1.40</td>
</tr>
<tr>
<td></td>
<td>HLNC</td>
<td>Pu total</td>
<td>0.87</td>
</tr>
</tbody>
</table>

---

* Fuel rod scanner.  
* Uranium neutron coincidence collar.  
* High level neutron coincidence counter.  
* Inventory sampling counter.
4.2. Verification measurement performance evaluations

The IAEA collected in recent years DA and NDA verification measurement results related to more than 60,000 items or batches of nuclear materials. Statistical evaluations of the differences between the operators’ declarations and the inspectors’ measurements are used for verification measurement performance evaluations (VMPEs). These yield estimates of the relative standard deviations of the within-inspection and between-inspection fluctuations for each combination of facility, material and measurement method. These estimates are updated regularly and are used for calculating inspection sampling plans, defining reject limits for NDA, and judging the significance of differences in quantitative verification measurements and material balance evaluations.

It has to be noted that these estimates include errors associated with both operators’ declarations and inspectors’ measurement results. Since they include all sources of uncertainties, they do not purely describe the quality of a given measurement method.

Tables I and II summarize for the major material types the IAEA’s estimates of verification measurement performance, involving DA and NDA, respectively. Ranges and average estimates are given for the two uncertainty parameters, the within-inspection and the between-inspection standard deviations. These parameters are based on data from several facilities, and the numbers of operator-inspector differences for the different material types range from about 100 to 5000.

The data shown apply to total element and isotope amounts. In the case of DA, they therefore include uncertainties due to weighing, sampling, sample preparation and the final analytical measurement. Verification measurements applying NDA techniques generally yield directly results for the total amount of element or isotope in an item. The only exception are measurements with the inventory sample counter; these measurements involve the same uncertainties in weighing, sampling and measurement as those obtained when DA measurements are used.

It had been advocated for some time [13] that the use of solid spikes of large size would improve the input analysis at spent fuel reprocessing plants and make these measurements easier also for the operators. The introduction of very accurately certified large size dried (LSD) spikes in the form of dried U, Pu nitrates has reduced by a factor of two the uncertainties in the verification of the plutonium concentration and amounts in spent fuel input solutions [14, 15].

It is impressive to see that the best minimum VMPEs for NDA come close to the mean values or even the best values achieved with DA measurements. A good example is the best result obtained with high level neutron coincidence counters (HLNCC) for PuO₂ and MOX. This will be the case when the industrial process yields oxides of uniform characteristics, with low and stable moisture content, and when the HLNCC results are combined with isotopic data derived from high accuracy mass spectrometric measurements.
### TABLE III. COMPARISON OF VERIFICATION MEASUREMENT PERFORMANCE EVALUATIONS (VMPEs) VERSUS STANDARD DEVIATIONS OF OPERATOR-INSPECTOR DIFFERENCES DERIVED FROM 1993 ITVs AND REIMEP (in % rel.)

<table>
<thead>
<tr>
<th>Measurement</th>
<th>Material</th>
<th>VMPEs</th>
<th>ITVs</th>
<th>REIMEP</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu concentration</td>
<td>Input solution</td>
<td>0.70</td>
<td>0.58</td>
<td>0.64</td>
</tr>
<tr>
<td></td>
<td>Pu nitrate</td>
<td>0.58</td>
<td>0.41</td>
<td>1.50</td>
</tr>
<tr>
<td></td>
<td>LWR-MOX</td>
<td>1.24</td>
<td>0.81</td>
<td>0.55</td>
</tr>
<tr>
<td>U concentration</td>
<td>Input solution</td>
<td>0.61</td>
<td>0.58</td>
<td>0.58</td>
</tr>
<tr>
<td></td>
<td>LWR-MOX</td>
<td>0.30</td>
<td>0.81</td>
<td>0.28</td>
</tr>
<tr>
<td></td>
<td>U oxide powders</td>
<td>0.46</td>
<td>0.35</td>
<td>0.51</td>
</tr>
<tr>
<td></td>
<td>U oxide pellets</td>
<td>0.21</td>
<td>0.17</td>
<td>0.11</td>
</tr>
<tr>
<td>$^{235}$U abundance</td>
<td>Input solution</td>
<td>0.31</td>
<td>0.40</td>
<td>0.81</td>
</tr>
<tr>
<td></td>
<td>Low enriched UF$_6$</td>
<td>0.46</td>
<td>0.29</td>
<td>0.06</td>
</tr>
<tr>
<td></td>
<td>U oxide powders</td>
<td>0.58</td>
<td>0.40</td>
<td>0.45</td>
</tr>
<tr>
<td></td>
<td>U oxide pellets</td>
<td>0.49</td>
<td>0.40</td>
<td>0.31</td>
</tr>
<tr>
<td>$^{239}$Pu abundance</td>
<td>Input solution</td>
<td>0.12</td>
<td>0.07</td>
<td>0.38</td>
</tr>
<tr>
<td></td>
<td>LWR-MOX</td>
<td>0.17</td>
<td>0.07</td>
<td>0.23</td>
</tr>
<tr>
<td></td>
<td>Pu nitrate</td>
<td>0.15</td>
<td>0.07</td>
<td>0.04</td>
</tr>
<tr>
<td>Pu amount</td>
<td>Input solution</td>
<td>0.83</td>
<td>0.76</td>
<td></td>
</tr>
<tr>
<td></td>
<td>LWR-MOX (DA)</td>
<td>1.97</td>
<td>0.83</td>
<td></td>
</tr>
<tr>
<td></td>
<td>LWR-MOX (NDA)</td>
<td>4.95</td>
<td>4.16</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Pu nitrate</td>
<td>0.79</td>
<td>0.67</td>
<td></td>
</tr>
</tbody>
</table>

#### 4.3. Comparison with the 1993 ITVs

The REIMEP 'performance values' include the combined contributions of the random and systematic uncertainties in the measurement of a single laboratory. For the purpose of comparison, the third column of Table III lists examples of the VMPE estimates of the combined effects of the within- and between-inspection fluctuations in the differences between the measurements of two laboratories, as they are observed during IAEA inspections. The figures in the last column correspond to the REIMEP 'performance values', multiplied by $\sqrt{2}$ to give the estimate of the average standard deviation of the differences between two laboratories. The ITV
column shows the standard deviations obtained when it is assumed that both the operator's uncertainties and the inspector's uncertainties are equal to the 1993 ITVs, and when the within-inspection parameters and the between-inspection parameters of the two parties are combined. In most cases the operator and the inspector are assumed to use the most common and the same combination of methods. The elemental assay of uranium oxide pellets constitutes an exception; in this case the operator is expected to use ignition gravimetry, while the inspector normally uses titration. The determination of the $^{235}$U abundance in UF$_6$ is another exception; the operator's values are normally derived from gas source mass spectrometry measurements, while a thermal ionization mass spectrometer is more often used by the inspector.

In most cases the mean VMPE values are close to or smaller than the values derived from the 1993 ITVs. This is not surprising because VMPE values were taken into particular account when selecting the 1993 ITVs at a level that is achievable under present industrial and inspection conditions. However, the mean VMPE values in Table III are noticeably larger than the values derived from the ITVs in the case of plutonium assay in spent fuel input solutions, LWR-MOX, and in the case of the determination of $^{235}$U abundances in low enriched UF$_6$. These values, in particular their between-inspection components, are still sufficiently low to permit the detection of one safeguards significant quantity (SQ) [16] with a 90% probability, as long as the inventory or the throughput of the plant does not exceed 55, 20 and 80 SQs, respectively. When the minimum VMPE values given in Table I are achieved for the same materials, it becomes possible to detect with high probability the missing of one SQ for inventories or annual throughputs of 130, 140 and 400 SQs, respectively.

An analysis of the experience gained in safeguards verifications shows that such low uncertainties or even lower ones are achievable under current conditions if both the operator and the inspector use analytical techniques of suitable performance, which are calibrated against well established and certified reference materials, and provided that careful and efficient quality control measures are enforced. A good example is the determination of the plutonium concentration in spent fuel input solutions by isotope dilution mass spectrometry using common LSD spikes with a 0.01% mean relative difference and standard deviation for between-laboratory fluctuations [12]. Wagner et al. [17] gave another excellent demonstration of the benefits of strict measurement quality control, with the evaluation of eight years of application of a K-edge densitometer (KED) for the verification of spent fuel input solutions. According to IAEA and Euratom experience reported in Ref. [17], the mean relative difference between operator and inspector over several years can be lower than 0.1%, with a total standard deviation of the difference of 0.20% or less when the fissile element concentration in plutonium and uranium nitrate product solutions is measured by DA or KED.

The REIMEP values are in most cases equivalent to or better than the ITVs and the mean VMPE values. This may reflect the fact that the sampling errors are probably negligible in REIMEP, as long as the homogeneity and the stability of the
intercomparison samples have been well ensured. Sample heterogeneity may indeed be a major uncertainty parameter in the plutonium assay of LWR-MOX under actual industrial and inspection conditions. In three cases — the plutonium assay in product solution, and the $^{239}\text{Pu}$ abundance measurement in input and product solutions — the ITVs and the average inspection data (VMPE values) were much better than the average REIMEP values. The question is whether this was so because most of the participants in the relevant REIMEP runs were not industrial or safeguards verification laboratories having the responsibility to perform analyses of these materials for accountability purposes as a matter of routine. On the other hand, the REIMEP values for the determination of the $^{235}\text{U}$ abundance in UF$_6$ indicate that the inspection verification results could probably be greatly improved, if this were needed, if the verification laboratories were to use gas source mass spectrometry.

5. CONCLUSION

The improvements in DA and NDA technology in the past years and recent results of inspection or measurement evaluation programmes indicate that the possibility of detecting with high probability the missing of safeguards SQs at large nuclear plants is not at all utopic. Yet, strict measurement control measures, well validated sampling methods and improved measurements of the volume of industrial solutions are prerequisites for achieving such detection. External measurement evaluation programmes such as EQRAIN [9] or REIMEP [10] are powerful tools to verify that quality control measures in laboratories are effective.

The within-laboratory and between-laboratory uncertainties or the between-inspection uncertainties in the analytical measurements performed by a single laboratory under normal operating conditions are now frequently close to 0.05%. The uncertainties of the certified values of the reference materials used for the calibration of these measurements become a significant parameter unless they are smaller than 0.03–0.02%. It would therefore be warranted that several commonly used reference materials which are still in large supply should be recertified with improved accuracy. Alternatively, new reference materials may need to be produced and certified with an accuracy within 0.02%, as are the plutonium metals NBL 126 and MP2, the uranium metals NBL 112A and MU2, the uranium isotopic reference materials IRMM 299 and 072, and the plutonium isotopic reference materials NBL 128 and IRMM 290. An internationally endorsed programme of intercomparisons of reference materials of this quality but of different origin would be needed to demonstrate their consistency and to make them traceable to the SI. As long as reference materials of this quality are not intercompared or are not available, it would be desirable that the operator and the inspector adopt the same and the best available reference material, traceable to the SI, for calibrating their analyses for all key measurement points of the plant. Such steps would eliminate the effect of uncertain-
ties in the reference materials as a source of systematic errors in the measurement of material balances and their verifications. These steps would also strengthen significantly the links between the field measurements and the SI for measurements of amounts of substance. Ultimately, the day may come when the IAEA will determine its safeguards detection goals in terms of ‘safeguards significant amounts’, rather than in terms of SQs, and when it will measure these in moles, the established SI units, rather than in kilograms. It is indeed the number of fissile atoms which is basically important for the make-up of an ‘SQ’ and not their ‘mass’. What better way is there of demonstrating that the international system of measurements has become a reality for chemical accountability and safeguards measurements of ‘amounts of substance’?

REFERENCES


ROBOTIZED EQUIPMENT FOR THE ON-SITE ANALYSIS OF FISSION MATERIAL

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Abstract

ROBOTIZED EQUIPMENT FOR THE ON-SITE ANALYSIS OF FISSION MATERIAL.

A robotized system for the preparation of safeguards samples and mounted in a glove box is described. The advantages of this approach lie in the high sample throughput, the associated low operator dose and the secure transmission of measured values. A standard laboratory robot and peripherals were used with minimum changes. The glove box was designed to prevent the robot mechanism being degraded by acidic atmosphere. The integration of the robot with an expert system allows the measured U and Pu concentrations from alpha spectrometry to be utilized to optimize the amount deposited on the mass spectrometry filaments for subsequent analysis by isotopic dilution mass spectrometry. Product U and Pu oxides and nitrates are handled also with a similar robot in a second glove box. In this box non-destructive methods are used for all samples, with a robot assisted automatic titrator for the potentiometric determination of a subset of the samples.

1. INTRODUCTION

The advantages of analysing samples containing fission material in on-site laboratories compared with transporting and analysing in central safeguards analytical laboratories have been discussed previously [1].

This paper describes instrumentation which has been developed for this purpose and adapted to on-site operation. Robotization has been applied as far as possible to ensure high sample throughput, low occupational dose and reliability of data transmission.
FIG. 1. Glove box for input sample analysis. Left compartment: heater block, mixer, reagent waste bottles; central compartment: robot with bar code reader, balance and exchangeable hands; right compartment: alpha counter and filament preparation.

2. METHODS

2.1. Analysis of reprocessing input solutions

A standard laboratory robot (ZYMARK) is mounted in a glove box. Peripherals, including exchangeable hands for specific tasks (e.g. grasping test tubes or pipetting solutions), racks for samples and spike solutions, analytical balance, mixer, heater block and a bar code reader are positioned within reach of the robot, as shown in Fig. 1. This glove box is designed for the preparation of reprocessing plant input samples by subsequent measurement by isotopic dilution mass spectrometry (IDMS). The glove box is divided into three compartments: a large one where the chemical manipulations are carried out, including sample and spike aliquoting by weight; a small compartment for dispensing of chemicals and a mixer and hot-plate; and a further one where alpha planchettes are prepared and measured and mass spectrometry filaments are prepared, all by the robot.

The main purpose of the compartments is to shield the relatively delicate mechanism of the robot from corrosive fumes engendered by warming acidic solutions during the chemical conditioning steps and during the preparation of alpha planchettes and mass spectrometry filaments. To allow access to the compartments, the robot is mounted on rails and under programme control can move from one end of the glove box to the other. The robot also opens the doors between the compartments for access and closes them afterwards.
Data are passed to and from the robot via a serial interface to a laboratory minicomputer. This computer is also connected via serial lines to a VAX computer where the measured data are stored and quality control tests made using an expert system, and to a PC on which the input specifications of the samples to be analysed are collected. This computer provides the starting information (e.g. number and size of samples and spikes) which is filtered over another expert system before being passed via the laboratory central computer to the robot.

The advantages of robot operation for such routine work are clear: increased throughput of samples and decreased personnel dose. Further advantages accrue from the reliability and reproducibility of the robot actions and particularly from the close coupling of the robot with peripheral devices. Thus, data are transferred to the robot to control the optimal amount of samples and spikes to be weighed, and the weight values are sent back directly to the central laboratory computer, minimizing transmission errors.

The solution concentrations are calculated by the central computer immediately after a short count of the separated solutions on the alpha counters and from this the amount of each sample solution to be deposited on the mass spectrometer filaments is transmitted back to the robot. The multichamber alpha counter and the filament preparation are controlled via robot programme commands. This approach necessitated designing and building a hardware extension to the robot input/output controller. Apart from this, however, the standard robot peripherals have been used as far as possible, with the exception of some alterations to the exchangeable hands of the robot.

The deposition of the sub-microlitre amounts of solution on the mass spectrometer filaments is carried out using a special rack which is centred over the filament. This then supports the tip of the Eppendorf pipette which the robot employs for the deposition. By this means a very accurate deposition is obtained without recourse to optical methods of control. The mass spectrometer turret is mounted on a turntable designed for this purpose: after the first solution is deposited, the turret is turned, the drop partially dried with a small resistor heater and the process repeated for the subsequent filaments. When all the filaments are finished, an electronic programmed controller is started which bakes the filaments following a preset programme. The turret is then ready to be mounted in the mass spectrometer, after affixing by hand the necessary blend plates.

The setting up of the filament deposition by the robot was checked for many turrets by a video recording. In this way, unexpected results could quickly be controlled to see if they resulted from failures in the deposition process. The robot programme for the solution deposition was also optimized in this way.
FIG. 2. Robotized glove box for analysis of product samples. Left compartment: balance and microwave oven (under box: neutron coincidence/gamma detector); central compartment: robot with balance and exchangeable hands, (under box: K-edge absorptiometer and gamma spectrometer); right compartment: radiometer titrators.

2.2. Analysis of product samples

Product samples, U and Pu oxides and nitrate solutions are determined in a separate glove box (Fig. 2). Potentiometric titration is the analytical method of choice here, backed up by a number of non-destructive methods (K-edge absorptiometry, neutron coincidence, gamma spectrometry).

The apparatus for these techniques is situated under the glove box, as shown in the figure, and the samples are inserted into wells or fingers projecting below the glove box for determination by these methods.

Dissolution of samples is planned to be carried out in a microwave oven situated in a separate glove box compartment for the reasons given above. The robot prepares solution aliquots for titration, adding reagents and positioning the aliquot beakers ready for the titration, which is carried out in two separate radiometer automatic titration devices. The robot signals via the central computer to the correct device to start the titration; the results are transmitted directly to the central computer, which subsequently signals to the robot to remove and rinse out the beaker and prepare the next solution. The close coupling of the active (robot and titration) devices promises a highly efficient and reproducible processing of product material.
The coupling of the robot with the non-destructive measurement techniques is feasible, but because of the relatively long measurement time, infrequent sample changing is needed and a robot does not bring any advantages in this case. The dose received by the operator from product samples is also much lower than that from input samples.

3. CONCLUSION

A robotized glove box for input samples has been constructed and has prepared a large number of safeguards samples for IDMS analysis. This has yielded much valuable information about the reliability and viability of the robot application and will be used in constructing a final version for installation on the site at Sellafield. A glove box for product samples, building on this experience, is under construction and will be brought into routine work in the near future.

REFERENCE

DEVELOPMENT OF A DIGITAL-ANALOGUE ROBOTIC SYSTEM FOR INPUT SOLUTION AND MIXED OXIDE SAMPLES

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Abstract

DEVELOPMENT OF A DIGITAL-ANALOGUE ROBOTIC SYSTEM FOR INPUT SOLUTION AND MIXED OXIDE SAMPLES.

Automated analytical systems for uranium and plutonium samples have been under development at the Nuclear Material Control Center (NMCC) since 1983. These include two systems for a large scale reprocessing plant for measurement of input solution samples and for measurement of mixed oxide (MOX) samples using automated destructive analysis procedures. Diverse tasks, such as designing the enclosures, developing the control systems and performing preparatory experiments, were combined. On the basis of NMCC's routine manual analysis, input solution sample analysis employing the isotope dilution method has been automated. A new method has been applied to automate the analysis of MOX powder samples. This method uses glass bead samples prepared by the system. Each sample may have different properties; therefore each is processed in different enclosures. The methods for the two analyses were quite different. The technical considerations were based on NMCC's ten years of expertise in nuclear material analysis.

1. INTRODUCTION

Automated analytical systems for nuclear materials have been developed by the Nuclear Material Control Center (NMCC). These systems will increase the analytical capability and improve the throughput of analysis measurements in a large scale reprocessing plant. Although the design criteria for these systems did not include accountancy measurements or process control, they are suitable for use as a safeguards measure.

Basically both systems perform automated chemical procedures. The preparation times required by the systems are almost the same as for manual analysis operations because the chemical reaction times are the same. However, these systems are superior to those operated manually because they are capable of unattended continuous operation. As a result, the analytical throughputs are significantly higher.
addition, these systems will reduce personnel exposure to radiation from radioactive material, avoid human errors in these very labour intensive tasks, and improve the quality of the analytical results.

Both systems developed are composed of various component devices and robots. After the initial data are inputted manually, the samples are treated and analysed automatically. Two types of robots have been applied by NMCC to the input solution and mixed oxide (MOX) samples. The automated analytical system for input solution samples [1, 2] uses stainless steel Cartesian robots. These robots operate in an acid mist environment. The MOX system [3, 4] uses commercially available dust-proof, multi-joint arm robots to handle the powder samples being analysed.

It is believed that these robotic systems with some control/surveillance (C/S) devices incorporated will have high tamper resistance. This should make them suitable for safeguards analysis, where they would contribute significantly to safeguards implementation.

2. SYSTEMIZATION TECHNIQUE

2.1. Enclosures

An operator can access only a semicircular area having a 60 cm radius centred at each glove port in the glove box. Operations requiring both hands to be used simultaneously are not feasible beyond the overlapping semicircular areas. The weight of devices to be lifted is limited to 10 kg.

The efficiency of the automated hot cell for nuclear material handling is lower than that of the automated glove box because the hot cell relies on remote operations using manipulators or tongs. A compact manipulator has a limited operational area. Therefore, mock-up tests for system arrangement are necessary for installation of devices into a hot cell. Devices in a hot cell must be designed to be disassembled for repair or removal by a manipulator and bag-out. This is a very important detail in systemization of an analysis hot cell.

2.2. Robots

Robots are usually classified into revolvable, cylindrical, and Cartesian types, depending on the arm mechanism design. The first two are popular for automated analysis. However, both types have similar limitations. Other devices cannot be installed near them because of the dead space under the arm, and the mechanisms needed to support the arm are very heavy. The system for analysis of input solution samples requires operation in a hot cell. Cartesian robots are especially useful for this application because component devices may be located under their arms and the
robots are easy to repair using manipulators. Robots are driven by hydraulic or pneumatic pistons or electric motors, etc. An electric motor for the arm movement mechanism and a pneumatic drive for the hand opening and closing mechanism were selected.

A robot used in an enclosure requires ruggedness and resistance to a harsh environment, since highly radioactive and acidic sample solutions are often handled for nuclear analysis. Commercially available robots were unable to meet these requirements. Thus, it was necessary either to modify a commercial robot or to design and fabricate a new robot.

When robots are used as the primary material handling devices in a system, the accuracy and repeatability of the stop positions of the robot hands are very important for reliable system motion. Manufacturers of most commercial robots intended for analysis applications specify the accuracy and reproducibility of their robot’s movement to be within 1 mm. Therefore, most of these robots meet the requirements for normal operation. In addition, the structure of the robot hand and the devices for sample receipt should be designed to provide infallible sample handling in spite of some shift in stop positions. This requirement enhances the reliability and safety of the system.

2.3. Component devices

Besides the robots, the analytical system contains devices that perform specific operations, such as a balance, an oven, etc. These are called component devices. Component devices should be resistant to a harsh environment, as are the robots. Especially for acid solution handling, the devices frequently have more exposure to acid fumes than the robot. Prudent countermeasures against corrosion are necessary. Metal ball bearings at exposed parts of the devices may be plagued by bearing adhesion that will lead to binding and malfunctioning of the mechanism. Like the robots, the component devices require some moving parts. Proper selection of electric motors and pneumatic drives for component devices should simplify both the means of control and the motor or drive reliability.

2.4. Electrical components

These systems require various sensors to determine the condition of a reaction or the position of the robot’s hand. Sensors must be selected that have resistance to the working environment. Devices in the enclosure should be designed with the premise that a sensor will be exchanged when its performance is impaired. However, a device designed without robust sensors will be short lived because sensor exchange is nearly impossible in an enclosure. If miniaturization is pursued at the expense of maintainability when designing the device, very difficult maintenance work will be encountered. Therefore, preference should be given to efficient exchange of sensors
instead of miniaturization when designing these systems. Miniaturization of a device installed in an enclosure is of secondary importance. Furthermore, for a positioning sensor, it is essential that the former stop positions of the robot remain unchanged after sensor exchange.

Electrical components having moving elements, such as motors and solenoids, may be vulnerable to the harsh environment. Therefore, sealed configurations of these components are usually selected to cope with the environment. However, a sealed device may be unable to be adequately cooled and may fail in certain situations. Thus, careful attention to the arrangement of the electrical components and the driven devices of a system is required. This consideration must address motor loads, heat from the devices, etc.

2.5. Piping and wiring

As with the glove box and the hot cell, reagents and compressed air necessary for system operation are supplied to component devices through tubing from outside the enclosure. For this application, the tubing materials should be corrosion resistant to acid fumes. The use of specially designed tube fittings that simplify remote manipulation is also important. When auto-piston burettes deliver reagents to the component devices, small bubbles may form in the delivery tubing. These bubbles will cause errors in quantifying volumes and should be removed.

Electrical cables routed in an enclosure are afforded protection from the harsh environment through their sheath materials or routing conduits. Electrical cables connected to moving devices must be flexible and durable because of the repeated bending and movement of the cables. Such movement often introduces failures in wires and connections where a failure may be difficult to find, especially since such failures are often intermittent. The data communication distance from the enclosure to the controller is large. Therefore, it is important to use well shielded coaxial cable so that noise is not superimposed on the signals. In addition, special electrical connectors suitable for remote maintenance should be used if the device is to be installed in a hot cell.

2.6. System control

Control devices for these systems are a programmable controller and a central processing unit (CPU), a personal computer in this application. The CPU approach has been applied to the input solution sample analysis system and the programmable controller approach has been used for the MOX powder sample analysis system. In the light of NMCC’s experience, the characteristics of both approaches are discussed below.

The programmable controller is designed to maintain its performance in harsh surroundings, such as in a factory. As a result, the programmable controller has the
advantage of resistance to dust and electrical noise. The programmable controller allows easy programming for data communication between devices and itself, since it does not require the program to monitor continuously the I/O port for data polling. Though a ladder chart program is familiar to a relay engineer, a chemist finds it awkward. Especially bothersome are the variables. Although undefinable in programmable controller language, variables are easily definable functions in high level languages such as C and Fortran. A ladder chart shows only relay numbers, so debugging work using such a chart is very tedious. Furthermore, a programmable controller program requires an interlock description for every system motion. This greatly complicates programming.

For a chemist, a CPU is easier to use than a programmable controller because a CPU allows the use of a familiar language. Its program only describes system motions without tiresome interlocks. Subroutines can easily describe the various unit movements. Also, it permits easy rearrangement of the analytical process. Data acquisition also should be noted; the CPU processes the program top down. This requires the return process after the interrupting process for data acquisition to be carefully described so as to avoid a polling error.

The characteristics of programmable controllers and CPUs are described above. The differences are the result of simultaneous processing of data as a word in a CPU versus the simultaneous processing of a line of data as a sentence in a programmable controller. In summary, the programmable controller method easily addresses the fixed analysis processes while the CPU method lends itself to situations where rearrangement of an analytical process for optimization is often necessary.

2.7. Fault monitoring

The developed automated systems have simple fault monitors consisting of hardware sensors and software. When abnormal movement of a component device is detected by monitoring the time necessary for unit operation during automatic operation, the computer usually outputs error messages on the screen and terminates sample processing. Abnormal movement of a robot, for example contact with a component device, can also be detected by monitoring a robot's motor current. However, it is not practical to provide complete fault monitoring and automatic recovery from the fault for unattended operation. Such monitoring is too complex, both in hardware and software. It is more practical to develop a less complex monitor that requires manual recovery from a fault. An absolute encoder for the robot drive mechanism's position is very useful to determine the system status after a fault or interruption of electric power. The use of an uninterruptible power supply (UPS) will enhance the reliability of the automated system.
2.8. System reliability

To maintain the efficiency of the electronic and mechanical devices, it is very important that the analytical system be designed to avoid the problem of enlargement of robotic movements. The alignment position for exchanged devices depends on the remote teaching playback method. Before performing analytical operations, these systems initiate positioning tests to determine if the movements of the robotic systems in each axis are normal. These tests are very effective in confirming correct operation of the systems.

The robots are expected to perform many tasks, for example vessel transfer, stirring and pushing. They also simplify the mechanisms of other devices, so these mechanisms are simpler and less costly. Complex robots are thus required and these are the most expensive components in the systems. To make the selection of expensive robots cost effective, the idle time of the robots was minimized. However, this high duty cycle complicated both the robot hand mechanism and the control software. The complicated hardware and software caused the system reliability to deteriorate and the robotic motion error to increase for the various functions. This required the addition of many sensors and safety mechanisms to the system to confirm all robotic motions. This evolutionary process proved to be a valuable lesson.

The reliability of these systems is inversely proportional to the complexity of the systems. It is important to have backup devices available for any possible accident and to cope with system fragility. Further, satisfactory operation of the system requires skill in device exchange.

3. DISCUSSION

With careful maintenance, the automated analytical systems for nuclear material can be reliably operated by the robotic systems in the enclosures. Image processing techniques using charge coupled device (CCD) cameras are employed and are particularly useful for movement control for the mass spectrometer filament sample loading process because precise positioning is necessary. These techniques do not require sensors to monitor directly movement of the robots. In the future, these techniques will be used to eliminate some sensors located in harsh surroundings.

If a robot installed in an enclosure can be completely isolated from its harsh surroundings in a way that conforms to nuclear regulations, a breakthrough will be realized for automated analysis in an enclosure. The robot of the future may be a new type of manipulator that has been robotized and uses playback functions. For the near future, optimized analytical processing may be accomplished with system control performed by artificial intelligence techniques using an analytical database.
ACKNOWLEDGEMENTS

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A FIELD STUDY IN THE USE OF LOW BURNUP PLUTONIUM AS SPIKE TRACER

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Abstract

A FIELD STUDY IN THE USE OF LOW BURNUP PLUTONIUM AS SPIKE TRACER.

Accuracy of analytical measurement methods plays a vital role in nuclear material accounting in a reprocessing plant. Input, product and waste discards of the plant are the strata where special nuclear materials are measured using different analytical techniques. Among them, input measurement, namely the analysis of fuel dissolved solution, is of great importance because it is the first place in the fuel cycle where plutonium produced in the reactor is physically measured. This measurement also helps in validating and refining the computer codes used for calculating the plutonium yields in a reactor fuel. Isotope dilution mass spectrometry (IDMS) has come to be regarded as the method best suited for this purpose because of its high precision, sensitivity and specificity. Since only a ratio measurement is involved, this method does not require a quantitative recovery of plutonium in the separation procedure preceding the determination step. This is not the case with other methods. In this method, highly enriched $^{242}\text{Pu}$ is normally used as the spike material. Owing to the difficulties encountered in the procurement of $^{242}\text{Pu}$ spike, use of low burnup plutonium (LBuPu) as an alternative spike material was tried and established in Bhabha Atomic Research Centre for the determination of plutonium in power reactor spent fuel dissolved solution. The paper describes a field study conducted in the power reactor fuel reprocessing plant in Tarapur to assess the performance of the method in a plant environment. Field study indicates that LBuPu can be used as spike in place of $^{242}\text{Pu}$ and the method has the added advantage of less isotopic fractionation during ratio measurement. LBuPu spike was used for the determination of the plutonium concentration in the input solution during one of the reprocessing campaigns. Comparison of the results obtained by the facility laboratory using LBuPu and those obtained by a referee laboratory using conventional $^{242}\text{Pu}$ spike are presented and discussed in the paper.

1. INTRODUCTION

Accurate assay of plutonium in the input solution is essential for nuclear material accounting in a reprocessing plant. This is the first place in the fuel cycle where the plutonium produced in a reactor is physically measured for comparison with the yields predicted using reactor burnup codes.

Isotope dilution mass spectrometry (IDMS), with its proven performance of sensitivity, reliability and amenability for automation, has come to be regarded as
the most appropriate technique for this purpose [1–3]. In this technique, the plutonium in the sample is isotopically equilibrated with a known amount of $^{242}\text{Pu}$ tracer added as spike material, and the ratio of $^{242}\text{Pu}$ to $^{239}\text{Pu}$ is measured by mass analysis. From the change in this ratio in the sample due to the addition of the spike, the $^{239}\text{Pu}$ content of the sample can be estimated. Knowing the relative abundances of other plutonium isotopes in the sample, the total plutonium content in the sample can be computed.

In spite of the advantages mentioned, the technique suffers from the shortcoming of its dependence on the availability of a highly enriched $^{242}\text{Pu}$ tracer, which is quite expensive.

A search for an alternative tracer revealed that it is possible to make use of the significant difference in the isotopics of plutonium (LBuPu) separated from low burnup (< 1000 MW·d/t) and high burnup (> 5000 MW·d/t) uranium so that one can be used as the spike tracer in the determination of the other by IDMS. This has become possible with the commercial availability of high precision thermal ionization isotope ratio measurement mass spectrometers that can determine small changes in abundances with extremely high precision. This method was reported for the first time from our research centre where it was used on simulated samples of power reactor fuel dissolved solutions [4]. The advantage of this spike material (less isotopic fractionation) over the conventional spike has also been demonstrated [5]. Since the LBuPu spike method involves measurement of only slight changes in the isotopic ratio of $^{240}\text{Pu}/^{239}\text{Pu}$ in the sample due to spiking, its suitability in a reprocessing plant environment needs to be investigated.

Hence a field experiment was conducted to evaluate the precision and accuracy of plutonium concentration determination in the input solutions. The experiment was conducted in two stages:

1. Five input batches in the power reactor fuel reprocessing plant (PREFRE), Tarapur, during one of the reprocessing campaigns involving PHWR fuel with burnup of about 7000 MW·d/t, were analysed by the facility laboratory, using $^{242}\text{Pu}$ as well as LBuPu as spike materials. Thus ten samples were analysed.

2. In a subsequent campaign, 27 input batches were analysed by the facility using an LBuPu spike. The same samples were also analysed by a referee laboratory using a $^{242}\text{Pu}$ spike.

2. EXPERIMENTAL

2.1. Instrument used

The mass spectrometer used in these experiments was a completely automated thermal ionization mass spectrometer (TIMS) system with double filament geometry and magnetic analyser with stigmatic focusing. This makes use of a multicollector
TABLE I. ISOTOPIC COMPOSITION OF SPIKE TRACERS USED

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Atom % in $^{242}$Pu spike</th>
<th>LBuPu spike</th>
</tr>
</thead>
<tbody>
<tr>
<td>238</td>
<td>—</td>
<td>0.018</td>
</tr>
<tr>
<td>239</td>
<td>1.52</td>
<td>93.42</td>
</tr>
<tr>
<td>240</td>
<td>2.94</td>
<td>6.23</td>
</tr>
<tr>
<td>241</td>
<td>1.68</td>
<td>0.31</td>
</tr>
<tr>
<td>242</td>
<td>93.86</td>
<td>0.025</td>
</tr>
</tbody>
</table>

Average atomic weight 241.94 239.12

242/239 ratio 61.78

240/239 ratio — 0.06668

with an eight Faraday cup configuration adjusted for simultaneous collection of masses 233, 234, 235, 236 and 238 (for uranium) and masses 239, 240, 241, 242 and 244 (for plutonium).

2.2. Spikes used

Plutonium separated from a fuel irradiated to about 1000 MW·d/t and purified on an ion exchanger was used as spike tracer (LBU Pu). The $^{239}$Pu and $^{240}$Pu contents of this tracer were about 94% and 6%, respectively. Highly enriched (> 90%) $^{242}$Pu spike tracer was also used. Both tracers were calibrated mass spectrometrically for isotopic composition as well as plutonium assay using a chemical standard. The isotopic composition of the two spikes used is shown in Table I.

2.3. Sampling

Triplicate samples were drawn from each batch of spent fuel dissolved solution. All three samples were diluted to nearly 150 times on a weight basis. The first sample dilution was preserved as a reference sample; the remaining two were aliquoted for mass spectrometric analysis by the PREFRE laboratory as well as by the referee laboratory.

2.4. Chemical treatment of samples

The aliquots of sample solution and the mixture of sample and spike tracer solution were first evaporated with concentrated nitric acid (to depolymerize the
polymerized species, if any). This was followed by a redox treatment with ferrous sulphate and sodium nitrite (to ensure the isotopic equilibration of the plutonium in the spike tracer and in the sample solution).

The dry spike technique of preparation of small samples of spent fuel dissolved solution was followed before sending samples to the referee laboratory. Diluted aliquot of dissolver solution was added to a dry mixed spike of $^{242}$Pu and $^{233}$U contained in a vial. Sample aliquots were treated chemically by evaporating them with a mixture of HNO$_3$ (8M), HClO$_4$ (1M) and HF (0.01M) to dryness.

2.5. Ion exchange separation

Plutonium in the chemically treated aliquots was purified until it was free of associated uranium and fission products by an anion exchange separation. This consisted of percolating the treated sample solutions and spiked mixtures in 8M nitric acid through a Dowex 1 × 4 50-100 mesh resin bed in nitrate form, washing the resin with about 8M and 3.5M nitric acid solutions and eluting the plutonium with 0.35M nitric acid.

2.6. Mass analysis

The concentrated eluate solution containing purified plutonium was deposited on the rhenium filament and the mass analysis was carried out as follows:

— In the case of sample aliquots not spiked with the tracer the complete mass spectrum was scanned.
— In the case of sample aliquots spiked with LBuPu tracer the isotopic ratio of $^{240}$Pu to $^{239}$Pu was measured.
— In the case of sample aliquots spiked with $^{242}$Pu tracer the isotopic ratio of $^{242}$Pu to $^{239}$Pu was measured.

2.7. Calculations

The concentration of plutonium in the sample was computed using the following equations.

*Equation to calculate the Pu concentration using LBuPu tracer:*

$$C_s = \frac{W_{sp}}{W_s} C_{sp} \frac{AW_s}{AW_{sp}} \frac{(AF_{239})_{sp}}{(AF_{239})_{sp}} \frac{(R_{0/9} - R_{sp0/9})}{(R_{m0/9} - R_{sp0/9})} DF$$

*Equation to calculate the Pu concentration using $^{242}$Pu tracer:*

$$C_s = \frac{W_{sp}}{W_s} C_{sp} \frac{AW_s}{AW_{sp}} \frac{(AF_{239})_{sp}}{(AF_{239})_{sp}} \frac{(R_{sp2/9} - R_{m2/9})}{(R_{sp2/9} - R_{m2/9})} R_{sp2/9}$$
TABLE II. DETERMINATION OF Pu CONCENTRATION IN REPROCESSING PLANT INPUT SOLUTIONS BY IDMS

<table>
<thead>
<tr>
<th>Sample</th>
<th>Concentration of Pu in sample (mg/kg of solution)</th>
<th>Ratio of values of the two methods</th>
</tr>
</thead>
<tbody>
<tr>
<td>No.</td>
<td>LBU Pu spike</td>
<td>$^{242}$Pu spike</td>
</tr>
<tr>
<td>1.</td>
<td>777.56 ± 1.83</td>
<td>778.13 ± 2.65</td>
</tr>
<tr>
<td>2.</td>
<td>418.78 ± 0.85</td>
<td>418.02 ± 0.63</td>
</tr>
<tr>
<td>3.</td>
<td>626.47 ± 0.30</td>
<td>627.65 ± 0.30</td>
</tr>
<tr>
<td>4.</td>
<td>587.78 ± 0.49</td>
<td>588.70 ± 0.73</td>
</tr>
<tr>
<td>5.</td>
<td>382.58 ± 0.06</td>
<td>383.31 ± 0.91</td>
</tr>
<tr>
<td></td>
<td>Average</td>
<td>0.9992 ± 0.0016</td>
</tr>
</tbody>
</table>

Mean difference between laboratories: - 0.25%
Standard deviation between mean difference: ± 0.44%

- Facility laboratory values (LBU Pu)
The centre line represents referee laboratory values ($^{242}$Pu spike)

FIG. 1. Interlaboratory comparison of IDMS (Pu) by two different spikes.
where \( W \) = weight of solution, \( C \) = concentration of solution, \( AW \) = average atomic weight, \( AF \) = atomic fraction, \( R \) = isotopic ratio (subscript 0/9 indicates \( ^{240}\text{Pu}/^{239}\text{Pu} \) and subscript 2/9 indicates \( ^{242}\text{Pu}/^{239}\text{Pu} \), \( DF \) = dilution factor, and \( s, sp, m \) denote sample, spike and mixture respectively.

3. RESULTS AND DISCUSSION

Results of plutonium assays carried out in the plant for five input samples with IDMS using both LBuPu and \( ^{242}\text{Pu} \) tracers as spikes are given in Table II. It can be seen that the inter-method difference is less than 0.2%. Figure 1 represents the inter-laboratory comparison of results obtained in 27 batches of input solutions by the facility laboratory using LBuPu as spike and the referee laboratory using \( ^{242}\text{Pu} \) as spike. It is evident from Fig. 1 that the average inter-laboratory difference is 0.25% with a standard deviation of \( \pm 0.44\% \). These results show that low burnup plutonium can be used for the estimation of plutonium in the power reactor fuel dissolved solutions, in place of \( ^{242}\text{Pu} \), with comparable accuracy and precision. The method has the additional advantage of less fractionation during isotope ratio measurement.

REFERENCES

1. INTRODUCTION

The Safeguards Analytical Laboratory (SAL) of the IAEA analyses Pu-bearing safeguards samples employing, inter alia, procedures after McDonald and Savage [1] and Davies and Gray [2]. The resulting aqueous discards amount to ≈ 300 L/a, with a specific alpha activity of ≈ 500 MBq/L, and contain 0.4 g Unat, 0.04 g reactor grade Pu and 0.004 g ²⁴¹Am per litre discard solution.

According to agreements concluded with the IAEA’s host country Austria, the nuclear material contained in the discards should be concentrated and returned to the countries of origin as the preferred option. Alternatively, the materials are to be converted into solid state and transferred to the Austrian Research Centre Seibersdorf (ÖFZS) for further management.

2. THE APPROACH

Either of the two alternatives is strongly impeded by the nature of the discard medium which contains strong, non-volatile mineral acids (H₂SO₄, H₃PO₄, NH₄HSO₄; 0.1M < [H⁺] < 1M) and the titration reagents (Ce³⁺, Fe³⁺, Al³⁺, Cr³⁺, VO²⁺, AsO₅³⁻; 10⁻³M < c < 10⁻²M). Therefore, the IAEA has called upon the United Kingdom and the German support programmes to suggest and develop a treatment process and to install, test and commission a treatment unit at SAL. A batchwise and sequential process was selected and investigated. It consists of five stages (Fig. 1), as described below.
(1) **Feed supply and adjustment**

Six litres of discard containing a defined number of analyses are collected to form a process batch, thus ensuring an identical batch composition. They are transferred to the process unit and mixed with 0.6 L 65% HNO$_3$ and 8.5 g Fe$^{2+}$ to adjust the solution to optimal separation conditions (Am(III)/Pu(III), [H$^+$] $\approx$ 2.5). A feed assay is carried out on-line by solid and off-line by liquid scintillation counting prior to the further treatment.

(2) **Chromatographic separation**

Solvent extraction chromatography is employed for the decontamination of the feed applying the breakthrough technique. The twin column concept is used, the
scintillation detector being placed between the two columns [3]. The feed is loaded onto the columns packed with 15 wt% CMPO (octyl(phenyl)-N,N-diisobutyl-carbamoyl-methyl phosphine oxide) and 35% TBP (tri-n-butyl phosphate) on dried Amberlite XAD 7. Pu, Am and U are retained on the column, the decontaminated column effluent is collected, posted out and transferred to the ÖFZS. The columns are scrubbed and stripped with 0.1M citric acid containing 0.025M Fe²⁺ and adjusted to a pH of 3. The scrub is collected in the feed, the strip in the eluate tank.

(3) **Evaporation**

The eluate tank also serves as evaporation vessel. It is placed on a hot plate. The solution is heated up to 80°C and slowly evaporated. The concentrated actinide solution is then collected for export to the countries of origin.

(4) **Immobilization**

As an alternative to evaporation, provision has been made to immobilize the actinides using ordinary Portland cement (OPC) [4]. The eluate is transferred to an immobilization vessel, the solution stirred, 40 g OPC/L are added and the suspension is adjusted to a pH of 10. After five hours, stirring is stopped and the OPC is allowed to settle into the receptor bottle attached to the immobilization vessel. After sedimentation, the decontaminated supernatant solution is removed and posted out as low level non-alpha waste. The receptor bottle containing the cured OPC is detached and also prepared for transfer to the ÖFZS.

(5) **Waste and product transfer**

The concentrated eluate from the evaporation, the OPC product from the immobilization and the decontaminated waste from chromatography and immobilization are subjected to quality control measures. The concentrated eluate and the OPC product are removed in double sealed containments and shipped in approved transport containers to their destination. The decontaminated waste is pumped from the box into a waste transfer vessel and conveyed as low level non-alpha waste to the ÖFZS.

3. **TREATMENT UNIT AND MODE OF OPERATION**

The entire treatment unit is housed in a four port glove box. A schematic diagram is shown in Fig. 2. The pumps (P1–P5), stirrers (S1–S4) and three port ball valves (A1–A10) are remotely operated. The unit accommodates the following loops:
FIG. 2. Schematic diagram of treatment unit.
3.1. Feed assay loop (concurrent flow)

*Feed tank—P2—A4—A5—A2—A3—A10—detector loop—A9—A7—feed tank*

The detector loop provides for a gross alpha determination (A10—detector—A9) and an Am determination (A10—column 3—detector—A9). Column 3 (diameter = 3 cm, length = 5 cm) is filled with tri-N-octylphosphine oxide (TOPO)/TBP on Amberlite XAD 7 and retains Pu and U, while Am is not extracted.
FIG. 4. Scrub/strip chromatogram of Am(III)/Pu(III) on CMPO/TBP.
3.2. Loading loop (concurrent flow)

Feed tank-P2-A4-A6-column 1-A5-A3-A10-detector-A9-column 2-A2-A1-waste tank

Loading is monitored in the first column effluent. The first breakthrough of the alpha emitters is observed when the breakoff volume V-BO 1 has passed the column (Fig. 3). This value is extrapolated for the second column and loading is discontinued when the feed volume V-BO T (2 V-BO 1) has passed the system.

3.3. Scrub and strip loop (countercurrent flow)

Scrubing and stripping are carried out with the same elutriant, but at different flows (Fig. 4). Scrubbing and stripping are monitored in the second column effluent. The scrub flow is directed back to the feed tank and removes the non-volatile acids and most of the analytical reagents from the column. The beginning of stripping is indicated by a drastic increase of activity in the effluent. Again, the corresponding effluent volume is extrapolated, this time to the first column. When this volume has passed the system, the first column effluent is directed to the eluate tank and the flow velocity is reduced for the stripping. The actinide recovery from the second column is completed when the detector reaches background level. Stripping is discontinued when twice this volume has passed the system.

3.4. Product/immobilization loop

Eluate tank-P3-immobilization vessel-receptor bottle

The eluate is concentrated by evaporation below the boiling point down to a defined liquid level at which the hot plate is switched off automatically. The evaporation is repeated for four additional runs, before the concentrated eluate is pumped with P3 into the receptor bottle of the immobilization vessel.

Immobilization vessel-P4-F-waste tank

Alternatively, the eluates of two runs are pumped without prior evaporation into the immobilization vessel and the actinides are fixed in OPC. The supernatant solution is pumped with P4 into the waste tank through a filter F to retain uncured OPC particles.
3.5. Waste transfer loop

*Waste tank-P5-box outlet QC7-waste transfer tank*

The decontaminated liquid waste is pumped with P5 into the waste transfer vessel located outside the box.

4. RESULTS

Eight runs have been processed with the treatment unit so far. The first one was a cold run for investigating the behaviour of the inactive components of the discards during scrubbing and stripping. During scrubbing, the non-volatile mineral acids and most of the other components (VO$_2^+$, AsO$_2^-$, Al$^{3+}$, Cr$^{3+}$) are completely removed; Ce$^{3+}$ and Fe$^{3+}$ are extracted and stripped together with the actinides.

The results of the active runs are compiled in Table I. Runs 6 and 7 were used to process aged, artificial discards with a low acid concentration. The formation of polymer, non-extractable Pu compounds led to the rather poor decontamination factors (DF). The eluates of runs 2 and 3 were immobilized. The process yielded 3 L liquid waste with a specific alpha activity of $2.5 \times 10^5$ Bq/L and a cured concrete block of ~750 mL. The eluates of the other runs were concentrated by evaporation. Twelve litres of eluate yielded a concentrated product of ~0.8 L.

### TABLE I. HOT OPERATIONS WITH THE DISCARD TREATMENT UNIT

<table>
<thead>
<tr>
<th>Run</th>
<th>$V_{\text{processed}}$ (L)</th>
<th>$A_{\text{r Feed}}$ (Bq/L)</th>
<th>$A_{\text{r Waste}}$ (Bq/L)</th>
<th>DF</th>
<th>$V_{\text{Scrub}}$ (L)</th>
<th>$V_{\text{Strip}}$ (L)</th>
<th>$A_{\text{r Strip end}}$ (Bq/L)</th>
<th>$T_{\text{total}}$ (h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2</td>
<td>6.1</td>
<td>$6.7 \times 10^6$</td>
<td>$&lt;5 \times 10^3$</td>
<td>&gt;1.3 $\times 10^3$</td>
<td>1.4</td>
<td>1.5</td>
<td>$1.0 \times 10^4$</td>
<td>5.5</td>
</tr>
<tr>
<td>3</td>
<td>6.7</td>
<td>$6.5 \times 10^6$</td>
<td>$&lt;5 \times 10^3$</td>
<td>&gt;1.3 $\times 10^3$</td>
<td>1.4</td>
<td>1.9</td>
<td>$1.0 \times 10^4$</td>
<td>6.2</td>
</tr>
<tr>
<td>4</td>
<td>6.1</td>
<td>$8.9 \times 10^7$</td>
<td>$2.0 \times 10^4$</td>
<td>4.3 $\times 10^3$</td>
<td>1.3</td>
<td>2.0</td>
<td>$2.7 \times 10^6$</td>
<td>5.8</td>
</tr>
<tr>
<td>5</td>
<td>6.5</td>
<td>$7.7 \times 10^7$</td>
<td>$3.5 \times 10^4$</td>
<td>2.2 $\times 10^3$</td>
<td>1.6</td>
<td>2.6</td>
<td>$6.0 \times 10^5$</td>
<td>6.7</td>
</tr>
<tr>
<td>6</td>
<td>6.5</td>
<td>$1.8 \times 10^8$</td>
<td>$3.0 \times 10^5$</td>
<td>6.0 $\times 10^2$</td>
<td>1.3</td>
<td>1.9</td>
<td>$2.0 \times 10^6$</td>
<td>6.0</td>
</tr>
<tr>
<td>7</td>
<td>6.8</td>
<td>$1.7 \times 10^8$</td>
<td>$1.3 \times 10^5$</td>
<td>1.3 $\times 10^3$</td>
<td>1.3</td>
<td>2.6</td>
<td>$4.0 \times 10^5$</td>
<td>6.7</td>
</tr>
<tr>
<td>8</td>
<td>6.7</td>
<td>$2.9 \times 10^8$</td>
<td>$4.5 \times 10^4$</td>
<td>6.3 $\times 10^3$</td>
<td>1.4</td>
<td>2.6</td>
<td>$1.0 \times 10^6$</td>
<td>6.5</td>
</tr>
</tbody>
</table>
In summary, the following can be concluded:

— For the evaporation variant, 1 L feed yields 1.4 L non-alpha waste \((10^5 \text{ Bq/L})\) and 0.03 L concentrated product.
— For the immobilization variant, 1 L feed yields 1.4 L non-alpha waste \((2.5 \times 10^5 \text{ Bq/L})\) and 0.03 L solid alpha waste. (An improvement of the decontamination is, however, expected with a better pH adjustment.)

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USING REFERENCE MATERIALS FOR WITHIN-LABORATORY QUALITY CONTROL

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The degree of reliability that is required of analytical results depends on the importance of the decisions that are based on these results. When measurement
results are used for nuclear accountability in general and for verification purposes in particular, a high degree of confidence needs to be put on these figures. Consequently, the analytical laboratory 'producing' and releasing such figures has to take adequate measures to provide the confidence required.

One of these measures is to establish a quality assurance system in the laboratory. This quality assurance system has to cover all the steps of the procedure to arrive at results and related uncertainties. Thus, not only the actual measurement steps have to be considered but also all the other parts of the analytical procedure that may, directly or indirectly, influence the results.

Well-documented procedures, training of staff, use of reference materials, regular use of quality control materials, participation in external quality control programmes such as REIMEP (Regular European Interlaboratory Measurement Evaluation Programme) [1] are essential components of a QA scheme. Using the same procedures for processing control samples as the materials provides essential information on the precision of the analytical technique under investigation. Reference materials are used to determine the bias correction factor of an analytical procedure. These expensive materials may of course also be used for internal quality control, in which case not only information on the precision but also information on the accuracy of the analytical procedure will be obtained.

A set of reference materials, prepared for calibrating K-edge instrumentation, was also used for internal quality control. The materials, solutions of the same acidity but different in uranium concentration, were used for uranium assay by K-edge densitometry (KED), by potentiometric titration and by isotope dilution mass spectrometry (IDMS). The results obtained by these three methods were used for assessing the quality of a particular measurement series, for intercomparison of methods, for comparing experimental variabilities between different operators and for determining biases in the one or the other measurement system.

The reference solutions were prepared from primary reference material (uranium metal) gravimetrically. The isotopic composition was certified by mass spectrometric measurements using gas source mass spectrometry ($^{235}$U/$^{238}$U ratio, Varian MAT 511) and thermal ionization mass spectrometry (Finnigan MAT 260 and Finnigan MAT 262 RPQ).

The reference solutions were provided in sealed glass ampoules, each containing 10 mL of solution. One part of the solution was directly used for KED, another part was diluted by weight to 40 mg U per gram of solution for titration, and a third aliquot was diluted to 800 $\mu$g U per gram of solution for IDMS.

Isotopic measurements were carried out using a Finnigan MAT 261 multicollector instrument and applying the 'total evaporation' method. IDMS samples were spiked with $^{233}$U spike solution prepared at the Institute for Transuranium Elements. The actual spiking and the chemical preparation steps were carried out using a Zymark robot [2]. The potentiometric titration that was applied is the well known Davies and Gray method. X-ray densitometry measurements were carried out
on a K-edge and a compact K-edge instrument [3, 4]. A comparison of the results obtained by the different methods is given in Table I.

The results obtained by IDMS on the various control samples show in general a standard deviation of 0.15%. The biases observed were usually small (<0.1%) and most often within the measurement uncertainty. However, the biases tended to be positive, possibly indicating a small systematic error.

For the titration, the observed differences between certified and measured concentration are randomly distributed and their absolute value (<0.1%) is usually smaller than the standard deviation of the measurements (0.15%) and much smaller than the overall accuracy of the results. Consequently, one may conclude that the method is well under control. Among the numerous dilutions that had to be made, only one sample was suspect as to a dilution error.

Direct determination of the uranium content in the undiluted samples allowed an instrument to instrument comparison. From these data, differences in the performance of the individual instruments could be studied.

The exercise has certainly demonstrated the power and the usefulness of a permanent quality control system. The use of carefully characterized and certified reference material for this purpose enabled detection and quantification of biases in the measurement system.

### Table I. Summary of the Results Obtained on the Various Solutions

<table>
<thead>
<tr>
<th></th>
<th>Concentration prepared (mg U/g)</th>
<th>Density (g/cm³)</th>
<th>Concentration calculated (g U/L)</th>
<th>K-edge mean (g U/L)</th>
<th>CKEDa mean (g U/L)</th>
<th>Titration mean (mg U/L)</th>
<th>IDMS mean (mg U/g)</th>
<th>Bias KED (%)</th>
<th>Bias CKED (%)</th>
<th>Bias titration (%)</th>
<th>Bias IDMS (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>42.973</td>
<td>1.16610</td>
<td>50.111</td>
<td>50.150</td>
<td>49.920</td>
<td>42.956</td>
<td>43.093</td>
<td>0.08</td>
<td>-0.39</td>
<td>-0.04</td>
<td>0.28</td>
</tr>
<tr>
<td></td>
<td>147.272</td>
<td>1.36170</td>
<td>200.540</td>
<td>200.320</td>
<td>200.520</td>
<td>147.301</td>
<td>147.449</td>
<td></td>
<td></td>
<td></td>
<td>0.10</td>
</tr>
<tr>
<td></td>
<td>201.550</td>
<td>1.48888</td>
<td>299.987</td>
<td>299.650</td>
<td>299.670</td>
<td>201.772</td>
<td>201.792</td>
<td></td>
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<tr>
<td></td>
<td>238.540</td>
<td>1.58119</td>
<td>378.682</td>
<td>377.370</td>
<td></td>
<td>238.610</td>
<td>238.659</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*a Compact K-edge densitometry.*
REFERENCES


IAEA-SM-333/23P

SPECIAL REFERENCE MATERIALS FROM THE INSTITUTE FOR REFERENCE MATERIALS AND MEASUREMENTS

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A. ALONSO, M. BICKEL, P. DE BIEVRE
Institute for Reference Materials and Measurements,
Joint Research Centre,
Commission of the European Communities,
Geel

1. INTRODUCTION

The duties of the Institute for Reference Materials and Measurements, IRMM (formerly called Central Bureau for Nuclear Measurements), were defined in the Treaty of Rome in the late 1950s. Although its tasks have shifted over the years and have been adapted to the changing needs of our society, the main goal will remain unchanged: to provide reference materials and perform reference measurements. For this purpose, IRMM is especially equipped and has developed a recognized competence in its fields of expertise.
Nuclear reference materials have always played a particular role at IRMM, since the 'nuclear' used to be the cradle of the Institute. The present reference materials activities at IRMM — in the context of nuclear safeguards — focus on two types of material: isotopic and elemental reference materials. These materials may be applied to a number of problems: calibration of instrumentation, quality control (see also paper IAEA-SM-333/23P, these Proceedings), method and instrument evaluation, and performance testing.

A variety of materials are produced on a routine basis and sold on a commercial basis. However, for certain applications or new instrumentation, suitable reference materials might not be available from stock or material certified to the desired degree of accuracy might not be available. In such cases, 'special reference materials' may be manufactured according to the customer's specification. The preparation and certification of such special material for special applications are described in the following section.

2. PREPARATION AND CERTIFICATION OF SPECIAL REFERENCE MATERIALS

The fact that materials have to be prepared upon request implies several conclusions: materials available from commercial sources do not fulfil all the needs; there are gaps in the existing series of materials; customers prefer to rely on a specialized laboratory's experience rather than manufacture the material themselves. Amongst the various requests that have been made to IRMM, we would like to highlight the following.

2.1. Uranium hexafluoride isotopic reference material

In 1992 IRMM was approached by a customer who needed UF$_6$ isotopic reference materials. As some of the enrichments requested were not available, it was decided to prepare these materials. UF$_6$ materials having a $^{235}$U abundance of 2.00, 2.50 and 4.00% had to be prepared. Well-characterized batches of natural uranium and enriched uranium (5%) fluoride served as starting materials for the preparation. Both compounds were distilled into a mixture under careful control of the distillation speed. After homogenization a test measurement was performed in order to verify the $^{235}$U/$^{238}$U ratio which was obtained. If the deviation from the desired value was outside the acceptance limit, another distillation step was performed, adding some more of one or the other starting material. Finally, the mixtures had to undergo an elaborate homogenization procedure and verification measurements during the last distillation step of the whole batch.
TABLE I. UF₆ ISOTOPIC REFERENCE MATERIALS AS PREPARED TO CUSTOMERS’ SPECIFICATION

<table>
<thead>
<tr>
<th></th>
<th>⁵²³U mol% (specified)</th>
<th>⁵²³U mol% (certified)</th>
<th>Relative deviation</th>
</tr>
</thead>
<tbody>
<tr>
<td>IRMM-025 (2 SD)</td>
<td>2.000</td>
<td>2.001 19</td>
<td>0.06%</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.000 96</td>
<td></td>
</tr>
<tr>
<td>IRMM-026 (2 SD)</td>
<td>2.500</td>
<td>2.501 2</td>
<td>0.05%</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.001 2</td>
<td></td>
</tr>
<tr>
<td>IRMM-027 (2 SD)</td>
<td>4.000</td>
<td>3.999 6</td>
<td>0.01%</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.001 9</td>
<td></td>
</tr>
</tbody>
</table>

The certification measurements were performed against IRMM isotopic reference materials. The $n$(²³⁵U)/$n$(²³⁸U) ratio was measured on a Varian MAT 511 instrument using the double standard method, while the $n$(²³⁴U)/$n$(²³⁵U) and the $n$(²³⁶U)/$n$(²³⁵U) ratios were measured on a Finnigan MAT 262 RPQ instrument. Independent verification measurements of all these ratios were performed on a Finnigan MAT 260 instrument. Table I compares the certified values with the specified ones. The agreement between nominal and certified value was found to be better than 0.2% relative in all cases.

2.2. PuO₂ isotopic reference material

There is a strong tendency to perform nuclear accountancy measurements by rapid methods which generate only a minimum of secondary waste and do not require highly skilled specialists to operate the measurement instrumentation. Such equipment has become more and more accurate and is now increasingly being used for safeguards verification measurements. At the request of Euratom safeguards authorities a combined neutron/gamma counter was developed by a commercial manufacturer. It is intended to be used for verification measurements in the product streams (PuO₂) of large installations.

An experiment was set up in order to demonstrate the performance of the instrument. For that purpose a series of reference materials were needed, being certified for ²⁴⁰Pu to better than 0.1%. A PuO₂ reference material for element assay (EC-NRM 210) certified for Pu content to better than 0.05% was used as the starting material. It was certified for isotopic composition by mass spectrometric measurements, calibrated against IRMM’s primary Pu isotopic reference materials (IRMM-290). In total, 80 g of PuO₂ were bottled in ten polycarbonate vials and
<table>
<thead>
<tr>
<th>Vial No.</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{240}\text{Pu}$ content (mg)</td>
<td>55.854</td>
<td>111.219</td>
<td>166.190</td>
<td>224.934</td>
<td>277.365</td>
<td>335.637</td>
<td>390.163</td>
</tr>
<tr>
<td>Uncertainty (2 SD)</td>
<td>0.034</td>
<td>0.066</td>
<td>0.098</td>
<td>0.134</td>
<td>0.164</td>
<td>0.198</td>
<td>0.232</td>
</tr>
<tr>
<td>Vial No.</td>
<td>8</td>
<td>9</td>
<td>10</td>
<td>11</td>
<td>12</td>
<td>13</td>
<td>14</td>
</tr>
<tr>
<td>$^{240}\text{Pu}$ content (mg)</td>
<td>445.783</td>
<td>500.498</td>
<td>556.958</td>
<td>56.871</td>
<td>166.596</td>
<td>278.649</td>
<td>390.146</td>
</tr>
<tr>
<td>Uncertainty (2 SD)</td>
<td>0.264</td>
<td>0.296</td>
<td>0.330</td>
<td>0.036</td>
<td>0.098</td>
<td>0.166</td>
<td>0.232</td>
</tr>
</tbody>
</table>
TABLE III. REFERENCE SOLUTIONS FOR U AND Pu ASSAY BY X RAY DENSITOMETRY

<table>
<thead>
<tr>
<th>Solvent</th>
<th>Element</th>
<th>Mass concentration (g·L⁻¹)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>U</td>
<td>378 300 250 200 150 50</td>
</tr>
<tr>
<td></td>
<td>Pu</td>
<td>47 25 10 5 1 0.2</td>
</tr>
<tr>
<td>3M HNO₃</td>
<td>U</td>
<td>25 10 5 1 0.2</td>
</tr>
<tr>
<td>3M HNO₃</td>
<td>Pu</td>
<td>40 20 10</td>
</tr>
<tr>
<td>1M HNO₃</td>
<td>U</td>
<td>0.4 0.2 0.1</td>
</tr>
<tr>
<td>1M HNO₃</td>
<td>Pu</td>
<td>250 100</td>
</tr>
<tr>
<td>3.5M HNO₃</td>
<td>U</td>
<td>2.5 1.0</td>
</tr>
<tr>
<td>3.5M HNO₃</td>
<td>Pu</td>
<td></td>
</tr>
</tbody>
</table>

Solutions are certified for element concentration to 0.1%.

five stainless steel vials. Special care was taken to avoid moisture pick-up during the manipulations, transport and storage. Table II summarizes the certified values. The results of the performance test were published by Menlove et al. and Davidson et al. [1, 2]. In addition, IRMM performed long term stability weight tests on these materials to monitor moisture pick-up.

2.3. U/Pu reference solutions for K-edge densitometry

The non-destructive assay of uranium and plutonium in spent fuel solutions as well as in product solutions is increasingly being carried out by direct methods such as K-edge densitometry or X ray fluorescence (XRF). Calibration of these instruments requires solutions that are similar to reprocessing input solutions, i.e. showing the same acidity, the same uranium concentration and the same U/Pu ratio. Several series of uranium, plutonium and mixed uranium/plutonium solutions with various known acidities were prepared upon request of the Euratom safeguards authorities. The solutions needed to be certified to better than 0.1% for element concentration and isotopic composition. Uranium metal (EC-NRM-101) and PuO₂ powder (EC-NRM-210) were used as starting materials. They were dissolved in nitric acid, the uranyl nitrate solution was evaporated to dryness and redissolved in HNO₃ of the required molarity. The acidity of the plutonium solution was adjusted by adding the required amounts of water or concentrated HNO₃.
As the K-edge instruments deliver a volume concentration, the gravimetrically certified values (mass concentration) needed to be converted into volume concentrations, too. Consequently, the density of the solutions had to be determined as accurately as possible. The solutions were bottled in glass ampoules which were then flame sealed in order to avoid concentration changes. Table III gives an overview of the different series of materials that were prepared and their certified values.

2.4. Uranium hexafluoride for gamma spectrometry

In-field measurements of the $^{235}\text{U}$ enrichment in UF$_6$ are usually performed using (portable) gamma spectrometers. Calibration of these instruments may be done with solid UF$_6$ samples. These have to be conditioned in a way that the material forms a layer of homogeneous thickness (>99.9% infinite thickness). A set of six such samples was prepared [4]. Eighty grams of UF$_6$ were bottled in each container. The thickness of the measurement window (i.e. the Monel wall of the container) was measured with a dial gauge against a thickness standard to be 2.000 ± 0.002 mm. Uranium-235 enrichment was certified by mass specrometry. Some of these samples were provided for the Regular European Interlaboratory Measurement Evaluation Programme, REIMEP, an external quality control programme; several others are still available as calibration sources.

2.5. Spikes for environmental analysis

IRMM has been offering spike solutions for many years. Amongst the actinide elements Th, U and Pu spikes are available in a variety of concentrations, chemical forms (solutions, dried nitrates, metals) and isotopes. However, for some applications highly diluted spike solutions are required. For these purposes a $^{233}\text{U}$ spike solution, a $^{230}\text{Th}$ spike solution and a $^{244}\text{Pu}$ spike solution were prepared on a

<table>
<thead>
<tr>
<th>Spike isotope</th>
<th>Certified concentration (μmol·kg$^{-1}$)</th>
<th>Uncertainty (2 SD) (μmol·kg$^{-1}$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>IRMM-044</td>
<td>$^{242}\text{Pu}$</td>
<td>38.060</td>
</tr>
<tr>
<td>IRMM-050</td>
<td>$^{235}\text{U}$</td>
<td>4.254 3</td>
</tr>
<tr>
<td>IRMM-051</td>
<td>$^{233}\text{U}$</td>
<td>10.101</td>
</tr>
<tr>
<td>IRMM-061</td>
<td>$^{230}\text{Th}$</td>
<td>2.474</td>
</tr>
</tbody>
</table>
gravimetrical basis. The total activity per spike ampoule was kept below 5000 Bq in order to enable all interested laboratories to handle this material (i.e. comply with radiation protection regulations) and facilitate the transport. The transport may then be carried out by normal parcel mail, as the lower limit for radioactive transports is not exceeded. The preparation of these materials was initiated by a request of the Bundesanstalt für Fleischforschung in Kulmbach, Germany. Table IV summarizes the characteristics of the materials prepared and available for distribution.

3. CONCLUSION

In addition to the reference materials which are provided by IRMM, special reference materials have been prepared upon request. Especially the initiatives coming from the Euratom safeguards authorities have led to efforts in the field of reference materials for non-destructive assay instrumentation. Conversely, the well-characterized tailor-made reference materials have helped to demonstrate the performance of certain techniques and to identify weaknesses in the measurement system. The ongoing developments in the instrumentation sector and the availability of well-characterized reference materials generates an atmosphere of mutual stimulation. Safeguards authorities and plant operators should encourage this evolution, as they will certainly profit from the results.

ACKNOWLEDGEMENTS

The financial support of the Euratom safeguards authorities (European Commission, DG XVII, DECS, E4) for some of the projects described in this paper is gratefully acknowledged.

REFERENCES


1. INTRODUCTION

According to IAEA safeguards criteria, nuclear material in solutions at spent fuel reprocessing plants, even with a low concentration of plutonium, should be verified for gross defects during physical inventory verification (PIV).

Laser induced photoacoustic spectroscopy (LIPAS) [1, 2] is a new sensitive technique for chemical analysis. There are only a few studies on the measurement of plutonium in perchloric acid by LIPAS of transuranium (TRU) elements [3]. The authors investigated the availability of LIPAS for the measurement of a low concentration of plutonium in nitric acid solutions from the PUREX reprocessing plant. The photoacoustic waves generated in the solution on absorbing modulated light were detected by matched piezoelectric transducers (PZT), and their amplitude was directly proportional to the plutonium concentration.

Another technique may be proposed to measure a small amount of plutonium in the highly radioactive waste solution. The authors also investigated alpha spectroscopy, combined with the isotope dilution method [4–6]; for the accurate and precise measurement of the plutonium. The difficulty of this method is to estimate the total plutonium concentration without isotopic abundances of all nuclides containing plutonium. The applicability of the isotope correlation technique is investigated to calculate the total plutonium concentration.

2. LASER INDUCED PHOTOACOUSTIC SPECTROSCOPY

2.1. Experimental

2.1.1. Equipment

A photoacoustic cell (Mitsubishi Heavy Industries, Ltd) was used to measure a small amount of plutonium in the reprocessing solutions, using an optical fibre [2] to introduce the excitation laser light to a sample solution in a glove box.
The experimental diagram for the LIPAS system is shown in Fig. 1. A pulsed Nd: YAG laser (BMI; 502DNS77/10) was used as the excitation light source. An optical parametric oscillator (OPO) (BMI; OP-901), which generates a wide range tunable laser light, was chosen to obtain the laser light matched to the specific wavelength of plutonium VI absorption. A 10 m optical fibre (core diameter: 200 μm; cladding diameter: 230 μm) was employed to introduce the laser light into a sample cell of stainless steel. The signal from the PZT was amplified by a spectroscopy amplifier (NF Electronic Instruments; 5305). The signal was accumulated and then averaged using a digital storage oscilloscope (Tektronix; TDS 460). A photodiode was used to synchronize the signal with the laser light and to monitor the laser energy in order to normalize the photoacoustic signal.

2.1.2. Data processing

The LIPAS signals from the PZT fluctuate with the variation of the energy of each laser pulse and with the electronic noise. The problem of random fluctuation is solved by averaging the accumulated data of at least 300 signals with a digital storage oscilloscope. The photodiode monitors the laser energy in order to correct the fluctuation of individual pulse energy.
2.2. Results

Although it is known that the Pu IV ion is the most stable in acidic solution (e.g. 3M HNO₃), it absorbs 475 nm light, which is influenced by other fission products involved and has a low absorption coefficient. The absorption wavelength of the Pu VI ion is 830.6 nm, which is advantageous. Figure 2 shows the Pu VI spectrum prepared at approximately 30 mg/L in a highly radioactive solution of PUREX reprocessing. This indicates that just the addition of Ce IV into the solution of interest provides the significant peak of total plutonium.

![Absorption spectrum of Pu VI in highly radioactive solution. Pu was oxidized by the addition of Ce IV \((\text{NH}_4)_2\text{NO}_3\)₆ (spectrophotometer; HITACHI: U-3410).](image)

![LIPAS calibration plots of Pu VI solutions.](image)
The photoacoustic signal of the Pu VI solution excited by 830 nm radiation from the YAG-OPO laser was observed. Figure 3 shows the photoacoustic signals as a function of Pu VI concentration in nitric acid (3M). The linear calibration curve was obtained between 8.0 and 160 mg/L. It can be estimated from the result with signal/noise information that plutonium is detected down to 1.0 mg/L.

3. ISOTOPE DILUTION ALPHA SPECTROMETRY (IDAS)

3.1. Experimental

Purified plutonium standard solution (38.0 mg/L solution in 3M HNO₃) was prepared to estimate the precision and accuracy of the IDAS technique. Three highly radioactive waste solutions were also prepared by the addition of appropriate amounts of plutonium. The plutonium concentrations in these solutions were adjusted to approximately 15, 50 and 100 mg/L.

A $^{242}$Pu solution, on the other hand, was used as a spike for the isotope dilution method. The $^{242}$Pu concentration was standardized by IDMS using NBL CRM-126 ($^{239}$Pu) as a standard and was found to be 450.7 mg/L solution in 3M HNO₃.

FIG. 4. Typical alpha spectrum from a source prepared by using anion exchange resin and TTA in o-xylene.
Plutonium in the highly radioactive solution with $^{242}$Pu spike solution for IDAS was purified by anion exchange and using solvent extraction methods with thenoyltrifluoroacetone (TTA) in o-xylene. This procedure is essential to obtain a very purified plutonium for alpha spectrometry.

### 3.2. Measurement method

The spectra were recorded after amounting to 500,000 counts in the three peaks corresponding to $^{238}$Pu, $^{239+240}$Pu and $^{242}$Pu. The typical spectrum is shown in Fig. 4.

Alpha spectra data acquisition comprises the following steps:

- calculation of each alpha peak of $^{238}$Pu, $^{239+240}$Pu and $^{242}$Pu
- calculation of the activity ratio of $^{238}$Pu to $^{239+240}$Pu.

The areas of three peak groups must be calculated. The peaks of $^{238}$Pu and $^{239+240}$Pu influence the regions B and C in Fig. 4, respectively, because of the effect of spectrum tailing. This effect can be subtracted by the method given in Ref. [7].

Table I shows a comparison of the plutonium isotopic compositions of plutonium solution determined by alpha spectrometry and mass spectrometry. Good agreement between alpha spectrometry and mass spectrometry is observed, especially on $^{238}$Pu/$^{239+240}$Pu (the average alpha spectrometry/mass spectrometry ratio is 1.002), which is used for the calculation of total plutonium concentration.
3.3. Isotope correlation

An isotope correlation technique was used to determine the total plutonium concentration. The relation between the alpha activity ratio ($^{238}\text{Pu}/(^{239}+^{240}\text{Pu})$) and total specific activity was investigated to determine the plutonium concentration. These values were calculated from the historical plutonium isotope compositions of dissolver solutions, namely data of PWR and BWR type spent fuel measured by mass spectrometry at the Tokai reprocessing plant (TRP). Consequently, it was found that the linear relationship (first order) between the ratio and the total specific activity was as shown in Fig. 5. This implies that discrimination between PWR and BWR is not necessary.

3.4. Calculation of plutonium concentration

The concentration of plutonium was calculated using the following equation:

$$C = C_{sp} \frac{A_{238} + A_{239+240} \lambda_{242}}{\lambda_{T} A_{242}} \frac{V_{sp}}{V_{s}}$$

where $C$ is the concentration of plutonium in the sample (mg/mL solution), $C_{sp}$ is the concentration of $^{242}\text{Pu}$ in the spike solution (mg/mL), $A_{238}$, $A_{239+240}$, $A_{242}$ are the peak areas of $^{238}\text{Pu}$, $^{239}+^{240}\text{Pu}$ and $^{242}\text{Pu}$ in the spiked solution, respectively, $\lambda_{T}$ and $\lambda_{242}$ are the decay constants of total Pu and $^{242}\text{Pu}$, and $V_{sp}$, $V_{s}$ represent the volume of the spike and sample solution, respectively.
### TABLE II. REPRODUCIBILITY OF IDAS FOR PLUTONIUM DETERMINATION

<table>
<thead>
<tr>
<th>Sample No.</th>
<th>Concentration of Pu (mg/mL of solution)</th>
<th>Error (rel. %)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>37.7</td>
<td>0.79</td>
</tr>
<tr>
<td>2</td>
<td>37.1</td>
<td>2.36</td>
</tr>
<tr>
<td>3</td>
<td>38.1</td>
<td>-0.26</td>
</tr>
<tr>
<td>4</td>
<td>37.8</td>
<td>0.53</td>
</tr>
<tr>
<td>5</td>
<td>37.2</td>
<td>2.11</td>
</tr>
<tr>
<td>6</td>
<td>38.4</td>
<td>-1.05</td>
</tr>
<tr>
<td>7</td>
<td>38.0</td>
<td>-</td>
</tr>
<tr>
<td>8</td>
<td>38.0</td>
<td>-</td>
</tr>
<tr>
<td>9</td>
<td>38.3</td>
<td>-0.79</td>
</tr>
<tr>
<td>10</td>
<td>37.5</td>
<td>1.32</td>
</tr>
</tbody>
</table>

Mean: 37.8
SD: 0.43
RSD (%): 1.15

### TABLE III. COMPARISON OF PLUTONIUM CONCENTRATION\(^a\)
IN HIGHLY RADIOACTIVE SOLUTIONS BY IDAS AND IDMS

<table>
<thead>
<tr>
<th>Sample No.</th>
<th>IDAS(^b) (mg/L)</th>
<th>IDMS (mg/L)</th>
<th>Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>15.6</td>
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<td>2.63</td>
</tr>
<tr>
<td>2</td>
<td>52.3</td>
<td>53.5</td>
<td>-2.24</td>
</tr>
<tr>
<td>3</td>
<td>102.4</td>
<td>100.9</td>
<td>1.51</td>
</tr>
</tbody>
</table>

\(^a\) Plutonium concentrations were adjusted by the addition of Pu to highly radioactive solutions.

\(^b\) Three repeated measurements were performed for each sample.
3.5. Plutonium measurement

The analytical results of the plutonium concentration together with the error (in rel. %) are given in Table II. The concentrations of ten samples are all the same. The mean plutonium concentration obtained by IDAS (37.8 ± 0.43 mg/L) agreed well with the adjusted value (38.0 mg/L). The precision of IDAS was also found to be 1.15%.

The highly radioactive solution with a certain amount of plutonium added was measured by IDAS and IDMS. The results for plutonium concentrations obtained by IDAS and IDMS are shown in Table III. Good agreement between IDAS and IDMS is observed in this result as well.

4. CONCLUSIONS

The proposed method, the LIPAS technique with just one stage of sample preparation (the addition of Ce IV), was found to be available for the rapid determination (in a few minutes) of a small amount of plutonium in highly radioactive reprocessing solutions. Use of an optical fibre enables remote operation for determining plutonium in highly radioactive solutions from the reprocessing plant. Further study for the measurement of plutonium will be made to confirm the application of the system using a highly radioactive solution in the hot cell.

We have also studied another measurement method, using IDAS, for the determination of small amounts of plutonium in a highly radioactive solution. It is confirmed that IDAS is also a precise and accurate measurement method for this purpose, although it takes four hours for a determination. It can be concluded that both techniques are quite useful for the determination of plutonium in highly radioactive reprocessing solutions.

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VOLUME AND MASS MEASUREMENT TECHNIQUES
IN LARGE AND SMALL TANKS FOR
RESEARCH AND DEVELOPMENT AND TRAINING
AT THE JOINT RESEARCH CENTRE ISPRA ESTABLISHMENT

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Ispra

1. INTRODUCTION

With new large scale nuclear reprocessing facilities scheduled to be coming on line, the safeguards inspectorates are facing increasing responsibilities at a time when resources are limited. Techniques for the determination of masses of liquids through weighing and volume measurements are used extensively in the fuel cycle at points which have a very high strategic importance and where the measurement uncertainty has to be low. Input and output accountancy measurements are the most important components of a safeguards system, where for a spent fuel reprocessing plant the measurements are invariably carried out in dedicated tanks whose only operational function is to facilitate accurate determinations of the nuclear material involved in transfer through the tank. The performance of the measurement techniques needs to be continually assessed and the inspectorates will need the verification and training capability to maintain an effective safeguards role. The problems are linked to phenomena affecting verification accuracy, instruments and equipment, calibrations, data acquisition, analyses and evaluation. A solution to resolve these problems can be found in establishing a capability for volume mass measurement research, development, testing and analyses on a continuous basis. This includes a tank
calibration facility incorporating tanks representative of those in current and future reprocessing plants. The Joint Research Centre (JRC) Ispra Establishment has taken on the setting-up of a large tank measurement laboratory called TAME which is now operational. In addition, the availability of the ADECO hot cells facility and its mini-scale reprocessing plant (input accountancy tank 45 dm$^3$) permits the use of depleted uranium for calibration purposes and the use of a fully automized K-edge system for analyses and training exercises.

2. DESCRIPTION OF THE TAME LABORATORY SET-UP AND THE INSTALLED EQUIPMENT

The main components of the TAME laboratory (TAME-LAB) are:

- an input accountancy tank of 12.5 m$^3$ resting on three load cells (up to 40 t);
- an output product vessel of 250 dm$^3$;
- a dosing station capable of operating in a continuous or an incremental mode with a maximum flow rate of 2500 dm$^3$/h in the former mode and variable up to 140 dm$^3$ in the latter mode;
- a 'D' shaped output product vessel (harp tank) of volume 400 dm$^3$ suspended by two tie rods connected to a weighing balance machine;
- three storage feed vessels of combined volume 14 m$^3$. These are connected to the circuitry via two pumps and an overhead tank of 250 dm$^3$ capacity, which provides a constant pressure head feed to the tank being calibrated.

A general schematic view of the main components in the TAME-LAB set-up is given in Fig. 1. The whole facility has been installed on an 8 m by 8 m surface area and extends to a height of 15 m, with two working platforms at the +5.5 m and +9 m levels.

2.1. Input accountancy tank

The main input accountancy tank is annular in shape, with a slab thickness of 400 mm, a diameter of 2900 mm and a height of 4190 mm. The bottom has a slope of 150 mm. Generally the vessel includes:

- ports for temperature measurement
- ports for sampling
- openings and windows for observation and inspection
- feed lines and discharge lines
- pneumercator lines (dip tubes)
- air sparging and recycle lines
- an off-gas line.
FIG. 1. Schematic view of the TAME-LAB layout.
The top of the vessel rests at the +5 m working level, giving easy access to all the flanges. The vessel can be seen in Fig. 2.

2.2. Output product vessel

This vessel, shown in Fig. 3, is of similar annular shape to the input accountancy vessel but with a slab thickness of 51 mm, a diameter of 1035 mm and a height of 2090 mm. The bottom of the vessel has a slope of 22.5 mm and the volume when full is 250 dm³.
2.3. Pneumercator lines

Pneumercator lines (or dip tubes), installed in each vessel which includes vapour head lines, are used for level and density measurements. The dip tubes have different cut lengths, covering different height positions in the vessel. These positions have been well characterized. All the dip tubes are connected to mass flowmeters which have an operating range up to 30 dm$^3$/h. The level and density are determined by connecting the lines via an electrovalve operating system to a series of differential pressure measuring devices.

2.4. Off-gas system

All the vessels, including the input accountancy tank, output product and storage tanks, are connected to the off-gas system. A variable extractor fan has been incorporated, up to 30 m$^3$/h, to create an underpressure inside the tanks with
respect to the ambient conditions from 0 to 15 mbar. Measurements taken on the off-gas include:

- temperature (16 sensors at various points of the circuit)
- absolute humidity (entering air and exit off-gas)
- differential pressure close to the tank and at the fan exit
- off-gas flow rate
- off-gas velocity via hot-wire anemometer instruments, both at the tank and the fan exit.

The off-gas line is connected to the tank via bellows, closely followed by a disentrainment column.

2.5. Instruments

The installed measurement equipment consists of:

- weighing systems (ASEA, Sartorius scales)
- rotary piston meter (Aquametro volume flow meter)
- differential pressure measurement gauges
- Paar densitometer.

2.5.1. Differential pressure measurement gauges

The TAME-LAB experimental set-up is equipped with a number of electro-mechanical gauges, namely:

- Diptron 3
- Diptron 4
- Ruska
- Hartmann-Braun.

The instrumentation rack supporting this equipment is shown in Fig. 4.

2.5.2. Diptron

Of the Diptron instruments, the Diptron 4 is the latest version of the series, with a range from 0 to 1000 mbar and an accuracy better than 0.05%. Both these instruments are connected to the main computer.

2.5.3. Ruska

Connected to the experimental set-up are three Ruska series 6000 digital pressure gauges with differential pressure ranges of 500, 750 and 700 mbar. The
manufacturers in their specification give a precision of 0.003%. In addition to the above, a Ruska series 6220 portable pressure gauge with a range of 1310 mbar has been installed and has a precision of 0.01%.

2.5.4. Paar densitometer

The Paar densitometer is utilized to determine the density of the calibration liquid as precisely as possible. The instrument measures the density on-line with an accuracy of 0.01%, according to the manufacturer's information. Samples are drawn continuously from various positions in the tanks, selected by electrovalves, at a flow rate up to a maximum of 55 dm$^3$/h.

3. TAME-LAB WORKPLAN

The current work being carried out in the laboratory concerns performance tests of the measurement systems described above and the definition of their
operating conditions. In addition, parallel work is under way on the characterization of the tanks, in particular:

- The tank geometry is specified, including the built-in equipment, e.g. dip tube differentials (probe separation).
- Homogenization of tank contents. At present there are two options, namely,
  (1) homogenization during filling, i.e. filling from the top and filling through a pipe leading to the bottom of the tank;
  (2) homogenization using air sparging or a recirculation pump.
- Sampling of tank contents to establish the extent of homogenization. The homogeneity of the tank contents is determined by:
  (1) density readings from dip tubes at various heights;
  (2) sample taking via air lift ejector at sampling stations;
  (3) sampling by gravity from ports on the tanks at various predetermined positions;
  (4) on-line density reading by the Paar system.
- Tank calibrations which are carried out batchwise (by increasing/decreasing weight instruments) or in a continuous mode. In this way calibration equations are established and different regions within the tank are identified. Figure 5 gives a view of the dosing station. Effects of the off-gas and the underpressure, together with evaporation, are taken into account. These can be quite appreciable during long calibration runs and for small incremental liquid additions.
4. TAME-LAB WORKSHOP

A second TAME-LAB workshop was held in October 1993 [1] where the representative parties were asked to provide suggestions and proposals for the activities to be carried out in the laboratory. Many proposals were made which are being analysed by the JRC, some of which have already been indicated above, but principally the main items of interest were:

- operational environment conditions, concerning effect of process variables (temperature, vessel vent, sparging, air purge rate, etc.);
- methods and instrument testing (pressure, weighing versus volume measurement techniques, use of tracers, sampling, data analysis techniques, etc.);
- verification procedures (authentication and unattended);
- training.

From the resulting discussions of the workshop there emerged a widespread interest in the future activities of the TAME-LAB, which will eventually result in collaborative actions. The JRC has also made a firm commitment to use the laboratory principally for R&D which may be needed by the safeguards authorities and by industrial operators, in co-operation with them.

5. PROCESS CONTROL, DATA ACQUISITION AND STORAGE

All the field instrumentation is connected to the process control system. The data from the various instruments are registered continuously and stored on hard disk for subsequent elaboration and evaluation. The operator selects which information is required during a particular experiment. Software has been written automatically to monitor and control the process during a continuous calibration or a batch feed mode exercise.

REFERENCE

QUALITY ASSURANCE SYSTEM OF THE SAFEGUARDS ANALYTICAL LABORATORY OF THE IAEA

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Department of Research and Isotopes,
International Atomic Energy Agency,
Vienna

1. INTRODUCTION

The perception of the term ‘quality’ has been greatly extended in recent years. In the case of an analytical laboratory, quality assurance (QA) systems have evolved which cover the entire organization of the laboratory. The assurance of the correctness of the analytical results through measurement quality control is of central importance, but represents only one component of a comprehensive QA system.

The QA system of the Safeguards Analytical Laboratory (SAL) was last presented and reviewed at the 1987 Advisory Group Meeting for Destructive Analysis [1]. Since then, the system has been further developed, documented and implemented.

2. QA MANUAL

One result of these activities is the completion of a QA manual [2], which serves as the central reference document in the implementation and performance of the QA plan. This QA manual has been prepared according to the models described by Delvin [3] and Ratliff [4]. It defines the commitment and the general quality policy of the laboratory’s management, and specifies in detail all procedures with respect to sample handling, instrument calibration, sample treatment and measurement, data handling and validation, and analysis control. Personnel training, procurement of instruments, reagents and reference materials, documentation and the personnel organization of the laboratory are other important aspects of the QA system which are described in the QA manual.
3. PERSONNEL STRUCTURE

One member of the senior technical staff was assigned the function of a Quality Assurance Co-ordinator (QAC). He reports in this function directly to the Head of the Laboratory, who retains the overall responsibility for all quality related issues. The underlying organizational structure of the SAL is shown schematically in Fig. 1.

4. IMPLEMENTATION

The initial implementation of a formal QA plan is a critical phase, because its realization would fail without the positive engagement of each individual staff member. The skeleton of the plan was jointly reviewed in meetings with all concerned technical staff before being adopted. The technical personnel themselves wrote the annexes of the QA manual which contain the detailed descriptions of all working and control procedures. These procedures were then jointly reviewed before their approval by the Head of the Laboratory.

5. MEASUREMENT CONTROL

Although just one of many components, the quality control (QC) of analytical results is a core element within the global QA system. The primary responsibility
All values are in [X bias] according to a reference value of 88.134.

FIG. 2. SAL quality control chart.
for QC rests with the analysts and the heads of the measurement or treatment groups. In a second step, an ‘after-the-fact’ QC is exercised by

(a) the Analytical Services Manager and his assistant, who perform a critical screening of all results before their reporting to the customer (normally the Department of Safeguards), and

(b) the QAC, who evaluates the control charts weekly (see below).

Both data screening and control charts are based on statistical QC and supported by appropriate computer tools.

6. QUALITY CONTROL CHARTS

The QC charts provide a quickly reviewable, graphical presentation of all control measurement results. They are used to judge the precision and accuracy of the results and to detect drifts, shifts and other aberrant behaviour. An example of these QC charts is shown in Fig. 2. Their statistical properties are described in detail in Ref. [5].

7. ANNUAL EVALUATIONS

To facilitate the long term review of the measurement performance, the QAC is responsible for issuing annual quality evaluation reports. These document the stability of the overall quality and also provide a basis for the definition of alarm and control limits to be used in the QC charts in the following period [6, 7].

8. EXTERNAL CONTROL

While the precision and the internal consistency of measurement results can be evaluated directly from the data, there will always remain some residual doubt about the absence of a systematic laboratory bias. The use of some sort of external reference is therefore indispensable. In the case of SAL four sources of comparison for its results are available:

(a) Evaluations of the operator-inspector differences, carried out by the Statistical Analysis Section of the Department of Safeguards’ Division of Concepts and Planning.

(b) ‘Three-Laboratory’ evaluations (in situations where the facility operator, the national safeguards laboratory and the IAEA analyse the same samples and make their results available to the IAEA).
TABLE I. EXTERNAL CONTROL PROGRAMMES. SAL RESULTS OF 1992 SAMPLES

<table>
<thead>
<tr>
<th>Programme: EQRAIN</th>
<th>Material</th>
<th>Sample</th>
<th>Analyte</th>
<th>Bias</th>
<th>CV (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu nitrate solution</td>
<td>B13 Pu</td>
<td>Pu element assay</td>
<td>-0.052</td>
<td>0.057</td>
<td></td>
</tr>
<tr>
<td>Pu nitrate solution</td>
<td>6 Pu</td>
<td>Pu element assay</td>
<td>-0.029</td>
<td>0.031</td>
<td></td>
</tr>
<tr>
<td>Pu nitrate solution</td>
<td>17 Pu</td>
<td>Pu element assay</td>
<td>0.070</td>
<td>0.072</td>
<td></td>
</tr>
<tr>
<td>Pu nitrate solution</td>
<td>33 Pu</td>
<td>Pu element assay</td>
<td>0.077</td>
<td>0.054</td>
<td></td>
</tr>
<tr>
<td>U nitrate solution</td>
<td>38 U</td>
<td>U element assay</td>
<td>-0.004</td>
<td>0.044</td>
<td></td>
</tr>
<tr>
<td>U nitrate solution</td>
<td>47 U</td>
<td>U element assay</td>
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<td>0.020</td>
<td></td>
</tr>
<tr>
<td>U nitrate solution</td>
<td>79 U</td>
<td>U element assay</td>
<td>-0.044</td>
<td>0.025</td>
<td></td>
</tr>
<tr>
<td>U nitrate solution</td>
<td>117 U</td>
<td>U element assay</td>
<td>-0.139</td>
<td>0.037</td>
<td></td>
</tr>
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</table>

<table>
<thead>
<tr>
<th>Programme: REIMEP</th>
<th>Material</th>
<th>Sample</th>
<th>Analyte</th>
<th>Bias</th>
<th>CV (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>MOX Spent fuel synth.</td>
<td>Results not yet published by organizing institute</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

(c) Joint analysis in co-operation with other laboratories from the IAEA’s network of analytical laboratories (NWAL), e.g. on the occasion of material characterizations.

(d) External control programmes, e.g.:
   — EQRAIN (Programme d’évaluation de la qualité du résultat d’analyse dans l’industrie nucléaire [8])
   — REIMEP (Regular European Interlaboratory Measurement Evaluation Programme [9]).

The first three can only provide an indication for an eventual bias; only the latter programmes involve the blind analysis of independently characterized and certified materials. Recent results of the SAL with these programmes are given in Table I.
9. AUDITING

The review of a QA system and the auditing of its implementation and performance by an external, independent expert (or group of experts) is only possible against a fixed and documented target model. The QA manual provides this required reference. Such auditing is highly desirable to ensure the credibility of the correctness of the analytical results reported by the SAL and hence the credibility of the entire safeguards verification system by destructive analysis. The SAL QA manual prescribes an annual auditing to be performed or organized by the Safeguards Analytical Services Officer of the IAEA’s Department of Safeguards.

Recently, considerable efforts have been undertaken by a number of international organizations to develop standardized reference models for QA systems, against which an actual system can be audited and certified. The formal ‘accreditation’ of an analytical laboratory is a matter of lively discussions currently going on. The SAL will strive to harmonize its QA system with these evolving standards, as soon as they become applicable.

REFERENCES


1. INTRODUCTION

In 1988 the Safeguards Analytical Laboratory (SAL) of the IAEA was equipped with a hybrid K-edge X ray fluorescence (XRF) analyser [1] for the measurement of nuclear materials in liquors. Besides the main application of the instrument — the training of inspectors for in-field measurements — in previous years the instrument was also used for off-site analysis of safeguards samples.

Analysis of uranium, thorium and mixed uranium/thorium oxide samples by K-edge densitometry (KED) has been reported [2, 3] as a technique for determining these elements with a precision and an accuracy of better than 0.2%.

We review here the joint work of the IAEA SAL and the Kernforschungszentrum Karlsruhe to determine quantitatively milligram size amounts of plutonium by dissolving them in an internal standard, containing a known amount of standard reference uranium, and by measuring the element ratio of uranium to plutonium.

The potential application of the instrument for this kind of analysis had been envisaged from the beginning. First, theoretical investigations and measurements were based on Pb/U mixtures, with Pb as spike element and U as sample element to allow tests outside a glove box.

Initial measurements on actual U/Pu mixtures, started in 1990, were first evaluated with the existing software developed for the determination of the U/Pu ratio in spent fuel solutions. However, the results thus obtained were found to be biased when the U/Pu ratio or the U concentration varied significantly. Therefore, the evaluation procedure was modified for the present application.
In June 1991 the final modifications of the revised evaluation procedure were implemented, and the system is now calibrated, qualified and used for routine determination of safeguards samples.

2. INSTRUMENTATION

The hybrid K-edge/XRF analyser installed in the SAL is based on the proven design of the 'hybrid instrument' developed for and routinely used by the Euratom Safeguards Inspectorate at the UP3 spent fuel reprocessing plant at La Hague [1]. Modifications were made to adapt the instrument to a standard glove box as used in the SAL. Further, a sample changer holding up to 28 samples was incorporated in order to allow unattended automated measurements. The design of the instrument and its basic components have been described previously [3].
3. MEASUREMENT PROCEDURE

Liquid aliquots containing about 4 mg of Pu are weighed into penicillin vials and evaporated to dryness (if samples are not already provided in the dried PNH form).

Before spiking, the XRF spectrum of the dried original sample material is measured to verify the absence of U or to determine the amount of U present. This is done by scanning the bottom of the penicillin vial until the Pu K\(_{\alpha_1}\) peak contains 100 000 counts. Uranium concentrations are then evaluated using the U K\(_{\beta_{1,3}}\) peak. The detection limit for uranium with this technique is about 0.02 mg, representing 0.5 wt% with respect to Pu. The dried Pu sample is then placed on a balance (Mettler AE163), and a weighed amount of a U reference solution (1 mL containing about 50 mg U/mL) is added as a spike. Standard and control solutions are prepared in the same way, starting with 4 mg Pu of a certified reference material.

Then, the mixture of spike and sample is again evaporated to dryness, redissolved in 0.3 mL 7M HNO\(_3\) and transferred into the special XRF vials (thin walled polyethylene vials with an inner diameter of 0.5 cm). The samples are placed in the sample holder, together with the control and standard solutions, and measured five times for 1000 s. The XRF spectra are accumulated with an operating voltage of 145 kV for the X ray tube.

4. EVALUATION OF THE U/Pu RATIO

Figure 1 shows the interesting part of a typical XRF spectrum from a sample with a U/Pu ratio of 12:1. The U/Pu weight ratio is derived from the measured net peak area ratio of the fluoresced U K\(_{\beta_{1,3}}\) and Pu K\(_{\alpha_1}\) through the relation:

\[
\frac{U}{Pu} = \frac{A(U) P(U K_{\beta_{1,3}}) I(Pu K_{\alpha_1})}{A(Pu) P(Pu K_{\alpha_1}) I(U K_{\beta_{1,3}}) R_{U/Pu}} \tag{1}
\]

where \(A\) is the atomic weight for U and Pu, \(P\) is the net peak area of the K X rays, \(I\) is the relative abundance of the respective K X rays, and \(R_{U/Pu}\) is the calibration factor describing the ratio of the excitation probabilities for emission of U K\(_{\beta_{1,3}}\) and Pu K\(_{\alpha_1}\) X rays in the primary X ray beam and their attenuation in the sample.

The calibration factor U/Pu depends on both the U concentration and the U/Pu ratio. The respective functional relationship has been determined from a parameter study:

\[
R_{U/Pu} = R_{U/Pu} \exp[U] \{a_0 + a_1 \ln \ln(U/Pu) + a_2 [\ln \ln(U/Pu)]^2\} \tag{2}
\]
The factor $R_{U/Pu}$ derived from this equation is further multiplied with a correction factor

$$D = b_0 + b_1 (U/Pu)$$

which corrects for a small residual dependence of the measurement results on the U/Pu ratio.

The U concentration required in Eq. (2) is determined from the measured net peak area of the U $K_{\beta_1}$ X ray as

$$U \ [g/L] = c_0 + c_1 U K_{\beta_1,3} \ [counts/s] + c_2 U K_{\beta_1,3}^2$$

The coefficients in Eqs (2) to (4) have to be established from calibration measurements.

The final U/Pu ratio, determined iteratively from Eqs (1) and (2), is multiplied with the correction factor $D$ from Eq. (3) and is finally used together with the known amount of the U spike to calculate the Pu content in the sample. This ratio is reported together with the measurement uncertainties as well as the mean and the overall uncertainty of repeated measurements.

5. UNCERTAINTY

**Counting precision.** The dominating error component of the overall procedure is the counting error for the Pu $K_{\alpha_1}$ X ray. For a typical Pu concentration of about 10 g/L, as obtained when the above procedure is followed, the counting precision for a single 1000 s run is approximately 0.27%, whereas the precision for the U $K_{\beta_1,3}$ line is about 0.05%.

**Sample preparation.** The weighing uncertainties remain of the order of 0.02%.

**Sample properties.** It is expected that small variations of the chemical composition will have a negligible effect on the U/Pu ratio. The influence of americium on the peak area determination of the Pu $K_{\alpha_1}$ X ray is sufficiently corrected.

**Instrument variability.** The stability of the high voltage applied to the X ray tube is probably the most critical instrumental parameter for the measurement of the U/Pu ratio. Therefore, a feature has been implemented to correct for its fluctuations. Making use of the second detector of the instrument, the program records the K-edge spectrum of the next sample in the sample changer in parallel to the XRF measurement. From these spectra the high voltage is determined from the cut-off energy of the X ray continuum. The correction factor determined from calibration measurements is 1.0016 for 0.1 kV deviation at 145 kV. The observed voltage stability of the high voltage generator is of the order of 0.1%, which is in agreement with the specified value.
**Calibration.** The samples are always measured together with a set of standard samples for measurement control, and the results are corrected accordingly if a statistically significant measurement bias is observed for the control samples. This correction covers systematic uncertainties of the calibration factor $R_{U/Pu}$, errors of the concentration of the spike solution (according to SAL experience, $4 \pm 0.06$) and long term instrument variabilities.

The total uncertainty of the reported result for the Pu concentration is mainly a combination of the uncertainty of the XRF measurement and the calibration error. The accuracy, which is given by the uncertainty of the concentration of the Pu reference material and the uncertainty of the bias correction factor (usually estimated on the basis of at least ten repeat runs at 1000 s), can be calculated as follows:

$$\sqrt{\left(\frac{0.27}{\sqrt{10}}\right)^2 + 0.06^2} = 0.10$$

(5)

The precision for the mean of two independent subsamples is 0.09% for the routine case.

An overall uncertainty of 0.2% at the 1σ level for 4 mg Pu samples can be achieved.

<table>
<thead>
<tr>
<th>Sample identification</th>
<th>Ratio U/Pu</th>
<th>Titration result (% Pu)</th>
<th>XRF result (% Pu)</th>
<th>Difference (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>4323-01</td>
<td>12</td>
<td>86.344</td>
<td>86.404</td>
<td>0.069</td>
</tr>
<tr>
<td>4323-02</td>
<td>12</td>
<td>86.496</td>
<td>86.230</td>
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<td>85.866</td>
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<td>11.2</td>
<td>86.051</td>
<td>86.112</td>
<td>0.071</td>
</tr>
</tbody>
</table>
6. RESULTS

After calibration of the instrument, the procedure was tested on two subsamples of a large dry spike of a known U/Pu ratio to verify the absolute accuracy of the ratio. Finally, actual Pu samples were measured and the results were compared with results obtained through the McDonald-Savage potentiometric titration procedure that is routinely applied for the determination of 4 mg Pu. The results are given in Table I.

7. DISCUSSION

The performance of the method described is comparable with the isotope dilution mass spectrometry method presently used for the analysis of milligram size Pu samples. The advantages of the method are the simplicity of the sample preparation and the minimum of waste produced.

The procedure described allows measurements of five samples per day (including the time needed for the measurement of the respective standard and control solutions). If more time is available, the overall uncertainty of the procedure can be further reduced by increasing the measurement time and the number of repeated measurements.

REFERENCES


1. INTRODUCTION

Since 1972 the New Brunswick Laboratory (NBL) modified Davies and Gray method [1] for potentiometric titration of uranium has been in use at the Safeguards Analytical Laboratory (SAL) [2] as the most selective and accurate destructive analysis method for uranium product samples in order to meet the goal of the Safeguards Verification System [3–6]. Since 1978 SAL has designed and operated three automatic uranium titration systems, MARK I–III [7].

The most recent system, Mark III, had a load capacity of 25 samples which could be processed without operator intervention. It was optimized for a sample size of 100 mg of uranium, resulting in 250 mL of unrecoverable radioactive waste per measurement. Like almost all commercially available instruments, its sample changer module was constructed for a fixed sample sequence. In 1992 Mark III had been in use for more than five years and over 13 000 samples had been processed with it. It was reaching the end of its usable lifetime.

2. OBJECTIVES AND GOALS

On the occasion of the construction of a new instrument, Mark IV, a number of new desirable features were specified in order to meet the modern, more stringent ecological, economic and quality requirements:

— reduction of the amount of waste (volume and radioactive material content)
— increase of the throughput and the loading capacity
— reduction of the demand for manpower during routine operation
— maintenance, or, if possible, improvement of the quality of the results.
FIG. 1. Schematic design of automatic titration system Mark IV.

FIG. 2. Automatic titration system Mark IV (detail).
TABLE I. RESULTS OF THE VALIDATION

<table>
<thead>
<tr>
<th>Level (mg U)</th>
<th>Raw bias (mg U)</th>
<th>SD (CV %)</th>
<th>Item</th>
<th>Value</th>
<th>Target value</th>
</tr>
</thead>
<tbody>
<tr>
<td>5</td>
<td>-0.05</td>
<td>0.1</td>
<td>Number of measurements</td>
<td>33</td>
<td></td>
</tr>
<tr>
<td>10</td>
<td>-0.13</td>
<td>0.11</td>
<td>Number of different dissolutions</td>
<td>4</td>
<td></td>
</tr>
<tr>
<td>21</td>
<td>-0.06</td>
<td>0.04</td>
<td>Number of different titration days</td>
<td>8</td>
<td></td>
</tr>
<tr>
<td>34</td>
<td>-0.06</td>
<td>0.02</td>
<td>Average bias</td>
<td>-0.02</td>
<td>0.00</td>
</tr>
<tr>
<td>45</td>
<td>-0.05</td>
<td>0.02</td>
<td>Random error SD</td>
<td>0.02</td>
<td>0.10</td>
</tr>
<tr>
<td>56</td>
<td>-0.01</td>
<td>0.01</td>
<td>Between dissolutions SD</td>
<td>0.02</td>
<td></td>
</tr>
<tr>
<td>Averages</td>
<td>-0.06</td>
<td>0.03</td>
<td>Between days SD</td>
<td>0.01</td>
<td>0.10</td>
</tr>
<tr>
<td>Target value:</td>
<td></td>
<td></td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
</tbody>
</table>

(a) Average sample size.

(b) Average relative deviation from expected value; uncorrected.

(c) Group's standard deviation.

(d) Estimates obtained from variance analysis.

(e) Estimate for the pure measurement error; values for comparison are from SAL-IR-6/90 (0.04) and SAL-IR-4/93 (0.04).

(f) For the propagated sum of both errors, 'measurement' and 'dissolution'.

(g) Component of the total fluctuation due to the factor 'dissolution' (expressed as standard deviation); values for comparison can also be taken from the reports SAL-IR-6/90 (0.02) and SAL-IR-4/93 (0.01).

(h) Component of the total fluctuation due to the factor 'titration day' (expressed as standard deviation); values for comparison: SAL-IR-6/90 (0.02) and SAL-IR-4/93 (0.02).

(i) Averaged on the basis of the group standard deviations in mg U; values for comparison from other sources: same as Ref. [5].

(j) According to '1993 International Target Values'.

No ready commercially available instrument could be found that met these specifications.

3. MARK IV FEATURES AND DESIGN

To decrease the amount of waste, the conventional Davies and Gray titration equipment was scaled down to approximately a fifth in volume and to a maximum
sample size of 40–50 mg uranium [8]. A new titration cell, electrode and stirring assembly was constructed to accommodate the smaller titration vessels.

To satisfy the requirement of a higher load capacity (and hence throughput) and to allow complete flexibility in the measurement sequence, a robotic system replaces the conventional sample changer. The flexibility in the measurement sequence enables the dynamic modification of the order of measurements in response to actual results and quality parameters. The new system is also able to perform additional operations such as the disposal of the waste into its collection container, an operation which was performed manually with the previous installations.

Bar code reading systems are going to be installed at SAL to enhance the tracking of sample identifications throughout the entire analytical procedure. The new titration system, Mark IV, is also equipped with such a bar code reading system.

The design of the new titration system is shown schematically in Fig. 1. Figure 2 shows a photograph of the system. The installation was completed in October 1993.

4. VALIDATION AND QUALIFICATION

Qualification schemes for new procedures or instrumentation are described in the SAL QA manual [9]. They comprise tests of the linearity, the calibration, the precision and accuracy and the effects of experimental factors and interferences. The latter two were already tested during the set-up of the chemical procedure [8].

The other validation tests were performed using primary standard reference materials, NBL-112a (uranium metal) for the tests of the linearity and the quality of the calibration and EC-110 (UO₂ pellets) for the validation of the precision and accuracy.

The results prove that the Mark IV automatic titrator can be used for the analysis of samples in the range of 5 to 55 mg uranium, although in practice it is not intended to apply it over such a wide range.

The average random error of the titration is of the order of 0.02–0.03 % (CV). The results of this exercise are given in more detail in Table I.

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EFFECTS OF PREPARATION, SHIPMENT AND AGEING ON ELEMENTAL ASSAY RESULTS FOR MILLIGRAM SIZED SAMPLES OF PLUTONIUM

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1. INTRODUCTION

Current safeguards criteria for Pu containing materials such as Pu nitrate, oxide or mixed U/Pu oxide (MOX) require sampling for bias defect detection. For this purpose, samples are sent to the Safeguards Analytical Laboratory (SAL) in
Seibersdorf for destructive analysis. Measurements to be carried out involve primarily elemental assay of Pu and U and their isotopic composition. In States which have currently not approved type B(U) containers for air shipment of gram sized Pu samples, it is necessary to apply a special sample pretreatment procedure [1]. This involves the quantitative dissolution of samples of solid materials, the dilution of solution samples, the taking of accurately measured solution aliquots containing several milligrams of Pu in penicillin vials, followed by evaporation to dryness. These milligram sized Pu samples can then be shipped to SAL in type A containers by air freight [2].

The delicate steps of this procedure are the weighing of the solution aliquots into the penicillin vials and the evaporation to dryness. All parts of this procedure are performed at the facility by operator personnel and witnessed by an IAEA inspector [3].

2. GOALS

The chemical treatment and subsequent assay of Pu milligram sized samples at SAL [4] critically depends on a quantitative redissolution and recovery of the dried deposit in the vial; this, in turn, is dependent upon the care with which the conditioning procedure was followed at the facility, on the conditions of transport (avoiding excessive shocks) and the ageing time between sample preparation and receipt at SAL [5]. It is thus worth while to study the impacts of these factors on the recovery of Pu and to check the performance of the overall verification procedure by comparing the results obtained by two different laboratories on actual samples.

3. EXPERIMENTS PERFORMED AND RESULTS OBTAINED ON SYNTHETIC SAMPLES

This paper describes and discusses the results of experiments at SAL designed to study the physical integrity of dried aliquots coming from synthetic samples of Pu and mixed U + Pu products (with U : Pu ratios of 1 : 1 and 1 : 20) which were aged for 80 days (representing the maximum possible delays from the time of the end of the conditioning at the facility to the time of the starting of the analysis at SAL). The impact of the different conditioning procedures used at one facility and at SAL [3, 6] and of the shipment conditions [7] on the quantitative recovery of the plutonium from these aged milligram sized samples (80 days) was quantified by determining the Pu content in each vial with the SAL standard dissolution and titrimetric procedures [5, 8].

The results indicate that for all types of material, conditioned with two different procedures, aged up to 80 days, and submitted or not to violent shocks simulating heavy conditions of shipments, the normal dissolution procedure applied at SAL
allows an average Pu recovery yield of 99.990 ± 0.059%, each individual difference being completely within the acceptance of the measurement technique: two times the standard deviation calculated on the basis of the international target values [9] (Fig. 1).

4. RESULTS OBTAINED ON ACTUAL SAMPLES

The effectiveness of the chemical procedure for recovering Pu was checked by examining the results of routine inspection samples; for 83 Pu nitrate samples (PNH) and 40 samples of U + Pu solution (MNH) and Pu + U mixed oxide (MOX) samples received at SAL between April 1992 and April 1993 from three different facilities and analysed by isotopic dilution mass spectrometry (IDMS) [10]; the agreement between duplicate subsamples was found to be 0.15% (coefficient of variation, 3% outlier rejection rate). This level of uncertainty can be ascribed purely to analytical (not treatment) errors (Table I).

In addition, a check of the precision and accuracy of the overall procedure, including sample treatment and measurement, was performed by examining the results from 12 Pu nitrate output samples (from September 1992 to January 1993),
### TABLE I. ESTIMATES OF THE COEFFICIENTS OF VARIATION OF THE BETWEEN-ALIQUOT Pu ERROR
(Milligram sized samples)

<table>
<thead>
<tr>
<th>Facility</th>
<th>Pu nitrate</th>
<th>Mixed oxide</th>
<th>Together</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>N</td>
<td>CV</td>
<td>N</td>
</tr>
<tr>
<td>A</td>
<td>23</td>
<td>0.13</td>
<td>37</td>
</tr>
<tr>
<td>B</td>
<td>—</td>
<td>—</td>
<td>2</td>
</tr>
<tr>
<td>C</td>
<td>57</td>
<td>0.14</td>
<td>—</td>
</tr>
<tr>
<td>Together</td>
<td>80</td>
<td>0.14</td>
<td>39</td>
</tr>
</tbody>
</table>

* Number of samples. Four samples rejected as outliers (3 PNH, 1 MNH).

### TABLE II. AVERAGE BIAS FOUND FOR Pu ON INSPECTION SAMPLES ANALYSED BY TWO LABORATORIES BOTH USING IDMS

<table>
<thead>
<tr>
<th>Type of material</th>
<th>Number of batches</th>
<th>Average differences in % ± SD</th>
<th>1993 target values [10]</th>
<th>Systematic error ((\sqrt{2} \times \text{ITV}_{\text{syst}}))</th>
<th>Random error ((\sqrt{2} \times \text{ITV}_{\text{rand}}))</th>
</tr>
</thead>
<tbody>
<tr>
<td>Output PNH</td>
<td>12(^a)</td>
<td>0.01 ± 0.27</td>
<td>0.28</td>
<td>0.28</td>
<td>0.28</td>
</tr>
<tr>
<td>Transfer PNH</td>
<td>21(^a)</td>
<td>0.12 ± 0.65</td>
<td>0.28</td>
<td>0.28</td>
<td>0.28</td>
</tr>
<tr>
<td>MNH + MOX</td>
<td>37(^a)</td>
<td>0.03 ± 0.21</td>
<td>0.28</td>
<td>0.28</td>
<td>0.28</td>
</tr>
</tbody>
</table>

* Three samples rejected as outliers (one for each type of material).

21 transfer samples and 37 MOX samples (from February 1991 to December 1992) which came from three facilities and for which results were reported by SAL and by the National Safeguards Analytical Laboratory (NSAL), both laboratories using IDMS [10]. For these samples, the two-laboratory evaluation showed that the average bias between SAL and the other laboratory ranged from 0.01% to 0.1% relative, with an average weighted standard deviation of 0.40% (Table II).
5. CONCLUSION

The milligram sized verification procedure applied to Pu bearing materials ensures quantitative verification of Pu with an estimate of the standard deviation of the systematic errors of about 0.1% relative and of about 0.40% for the random errors. A better estimate of the systematic error would be obtained if the analytical laboratories had used two different analytical techniques.

REFERENCES

HISTORY, DESCRIPTION AND OPERATION

An important task of safeguards inspectorates is the verification of operators’ declarations of $^{235}$U enrichment in fuel fabrication processes. On-site verification has two main attractions for safeguards inspectorates: (1) the number of samples transported to the laboratories of safeguards authorities is reduced, and (2) the timeliness goals are readily met.

In 1984, a mobile gas mass spectrometer manufactured by Balzers entered into routine service at the Euratom Safeguards Inspectorate (ESD) for the purpose of measuring the enrichment of UF$_6$ samples taken at enrichment and fuel fabrication facilities. In 1986, a Finnigan Mat THQ mobile mass spectrometer entered into routine service, making measurements of the enrichment of uranyl nitrate solutions and uranium oxide pellets and powders on the site. Owing to transport difficulties with equipment that is potentially contaminated with plutonium, both instruments are restricted to uranium measurements. Over sixty on-site measurement campaigns have been carried out with these instruments. The initial development and in-field experience are described in Refs [1-5].

A set of portable titration equipment is normally transported together with the THQ instrument and is used for measurement of the uranium concentration of samples by Davis and Gray titration. Sample preparation for isotope dilution mass spectrometry (IDMS) and the need to measure a greater number of isotopic ratios would considerably reduce measurement throughputs and therefore render the mobile mass spectrometry less attractive.

At the time of purchase of the mass spectrometers, only quadrupole instruments could meet the size, weight and transportability requirements for mobile instruments. At an installation, the sample throughput is governed by the time taken for access formalities, equipment assembly and placing under vacuum, sample preparation, the measurements themselves, the permitted working hours, instrument disassembly and finally radioprotection controls upon leaving the facility. For a one-week visit to an installation there are three useful measurement days. To increase
the number of analyses possible during the time available, only the $^{235}\text{U}$ to $^{238}\text{U}$ ratio is normally measured. However, a check is first carried out with the spectrometer to ensure that the $^{234}\text{U}$ or $^{236}\text{U}$ content will not significantly bias the results. To avoid transport of standard materials, sets of standards are stored at each installation where measurements with the mobile instruments are performed. Each set of standards covers the range of 0.2–5% enrichment.

The instruments are based at ESD headquarters in Luxembourg. Work, maintenance and testing with radioactive materials are performed at laboratories of the European Commission. The instruments are operated by specialists rather than inspectors.

**Technical details of the Balzers gas instrument**

The instrument uses the QM 400 quadrupole system covering the mass range 0–512 amu. It has four sample inlets, a diffusion pump and a rotary pump serving the inlet system, molecular beam injection and an electron impact ion source. The analysis section is kept under vacuum, using an ion getter pump backed by a turbomolecular pump whose exhaust is pumped by a rotary pump. Detection is by a Faraday cup. Instrument control and data handling are through a PC. Sample and reference bottles are connected and measured alternately.

**Technical details of the Finnigan MAT THQ**

The instrument is constructed around a Finnigan MAT quadrupole analyser of mass range 0–300 amu. The turret takes up to 13 sample filaments. The samples are thermally ionized. Vacuum is supplied by a rotary pump and a turbomolecular pump. Detection is by a Faraday cup. Instrument control is via a Hewlett-Packard computer.

**Operating procedures**

The operator of the mass spectrometer normally receives the samples from the inspectors together with the declared enrichments. Blind measurement would entail two measurements — a first measurement to optimize analytical conditions and the measurement proper. Once the results are available, the inspectors are notified and samples are remeasured as necessary.

**Performance of the Balzers mass spectrometer**

The operator–inspector differences for the last 419 measurements of enriched UF$_6$ were plotted and visually inspected. First, two results having relative deviations in excess of 20% were removed. Then the data were replotted and data points in excess of 5 SD (three points) were removed. The result can be seen in Fig. 1.
FIG. 1. Operator-inspector differences for enriched UF₆ samples.
FIG. 2. Operator-inspector differences for enriched U samples measured with the THQ.
The mean RD is $-0.038\%$, indicating a slight negative bias, and the SD is $0.31\%$.

**Performance of the THQ instrument for solid uranium samples**

The operator–inspector differences for the last 132 measurements of solid uranium samples, mostly $\text{UO}_x$ powders and pellets having accurate enrichment declarations, were plotted as shown in Fig. 2. No outlier treatment was made. The mean for these data was $-0.07$ and the SD was 0.35.

These points represent about one third of the samples analysed with the THQ instrument. The majority of samples analysed are physical inventory verification samples having only nominal or estimated values.

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DEVELOPMENTS OF
CONTROLLED-POTENTIAL COULOMETRY
FOR SAFEGUARDS PURPOSES

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The controlled-potential coulometric method is being developed for use at the Safeguards Analytical Laboratory, Seibersdorf, for the analysis of plutonium at the 1–2 mg level.

In the programme of work to assess and optimize this method the aim is to establish a procedure and equipment that will be suitable for eventual use in a fully automated version of the method. The effect of a prior separation of plutonium by ion exchange, to remove interfering elements, is also being investigated.

The precision of the coulometric method [1, 2] based on the oxidation of Pu(III) to Pu(IV) was initially assessed using the Harwell 6000 series computer controlled coulometer and a gas stirred electrolysis cell used routinely for the determination of plutonium at the 10–20 mg level. Replicate determinations were performed on aliquots of a standard plutonium solution and a precision of ±0.177% was obtained (Table I, line 1).

In order to seek an improvement in the precision, a cell of smaller dimensions was constructed to reduce the electrolyte volume and thereby increase the plutonium concentration in the electrolyte solution. Further series of determinations were carried out using this cell, two 6000 series coulometers and a new Harwell coulometer with updated circuitry. The results obtained (Table I, lines 2 to 4) showed that there was a significant increase in precision due to the reduction in electrolyte volume in the electrolysis cell and that the use of the new coulometer brought about a further improvement, to the extent that a precision of better than ±0.1% was achieved.

The effects of terminating the electrolysis at a lower current and the use of a working electrode with a smaller surface area were investigated, but such changes led to a decrease in precision.

The gas stirred cell design is not well suited to automation and therefore a cell was designed for this purpose; the electrodes and mechanical stirring system are
### TABLE I. PRECISION RESULTS FOR THE DETERMINATION OF PLUTONIUM AT THE 1 mg LEVEL IN GAS STIRRED CELLS

<table>
<thead>
<tr>
<th>Cell</th>
<th>Free electrolyte volume (mL)</th>
<th>Coulometer</th>
<th>Number of determinations</th>
<th>RSD</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>8.5</td>
<td>Harwell 6000 series, A</td>
<td>9</td>
<td>±0.177</td>
</tr>
<tr>
<td>2</td>
<td>5</td>
<td>Harwell 6000 Series, A</td>
<td>18</td>
<td>±0.138</td>
</tr>
<tr>
<td>2</td>
<td>5</td>
<td>Harwell 6000 series, A</td>
<td>20</td>
<td>±0.143</td>
</tr>
<tr>
<td>2</td>
<td>5</td>
<td>New Harwell coulometer</td>
<td>13</td>
<td>±0.088</td>
</tr>
</tbody>
</table>

### TABLE II. PRECISION RESULTS OBTAINED USING AN ELECTROLYSIS CELL WITH MECHANICAL STIRRING

<table>
<thead>
<tr>
<th>Pu equivalent level (mg)</th>
<th>Number of determinations</th>
<th>Electrode type</th>
<th>RSD</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>11</td>
<td>Platinum</td>
<td>±0.078</td>
</tr>
<tr>
<td>1</td>
<td>18</td>
<td>Gold</td>
<td>±0.335</td>
</tr>
</tbody>
</table>

### TABLE III. DETERMINATION OF PLUTONIUM RECOVERY FOLLOWING ION EXCHANGE SEPARATION

<table>
<thead>
<tr>
<th>Eluant</th>
<th>Number of determinations</th>
<th>Mean Pu recovery</th>
<th>RSD</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.4M HCl/0.01M HF</td>
<td>6</td>
<td>99.929</td>
<td>±0.218</td>
</tr>
<tr>
<td>1M H₂SO₄/1M HNO₃</td>
<td>9</td>
<td>99.778</td>
<td>±0.107</td>
</tr>
</tbody>
</table>

### TABLE IV. ION EXCHANGE COLUMN BLANKS

<table>
<thead>
<tr>
<th>Eluant</th>
<th>Number of determinations</th>
<th>Net column blank (µC)</th>
<th>SD (µC)</th>
<th>SD as a percentage of Pu</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.4M HCl/0.01M HF</td>
<td>9</td>
<td>5507</td>
<td>±716</td>
<td>±0.09%</td>
</tr>
<tr>
<td>1M H₂SO₄/1M HNO₃</td>
<td>7</td>
<td>2630</td>
<td>±606</td>
<td>±0.08%</td>
</tr>
</tbody>
</table>
mounted in the lid of the cell. The electrolyte is contained in a simple beaker type vessel, which can be moved by a robotic arm from previous operational stages in the analysis to the cell head.

The performance of this cell was examined under ‘inactive’ conditions in which iron was used to simulate plutonium. Series of determinations were carried out in which gold and platinum working electrodes were used and aliquots of a standard iron solution containing the equivalent of 1 mg of plutonium. The results obtained (Table II) show a marked difference in performance between the platinum electrode and the gold electrode. The results obtained with the platinum electrode were very consistent, with a precision of ±0.08%, while the results obtained with the gold electrode were inconsistent because of a progressive decrease in recovery for successive determinations.

The effect of a prior separation by anion exchange on the coulometric method was examined using a separation procedure based on the sorption of a plutonium hexanitrat complexes. The separation procedure was designed for a sample containing 2 mg of plutonium, and two series of plutonium recovery experiments were conducted using different eluting agents. The results of the recovery experiments (Table III) were not conclusive, since inspection of the individual results obtained with the HCl/HF eluting agent indicated that the poor precision could be masking a lower bias. The results of blank separations (Table IV) show that the uncertainty in the measurement of the blank alone limits the precision of the whole method to ±0.1%.

The investigation has established that a precision of ±0.09% is attainable for the determination of 1 mg quantities of plutonium by coulometry, under ideal conditions. From the limited experimental data obtained for the determination of 2 mg quantities of plutonium following ion exchange, the precision obtainable is between ±0.1% and ±0.2%, and it is highly probable that there is a negative bias of between 0.1% and 0.2%.

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