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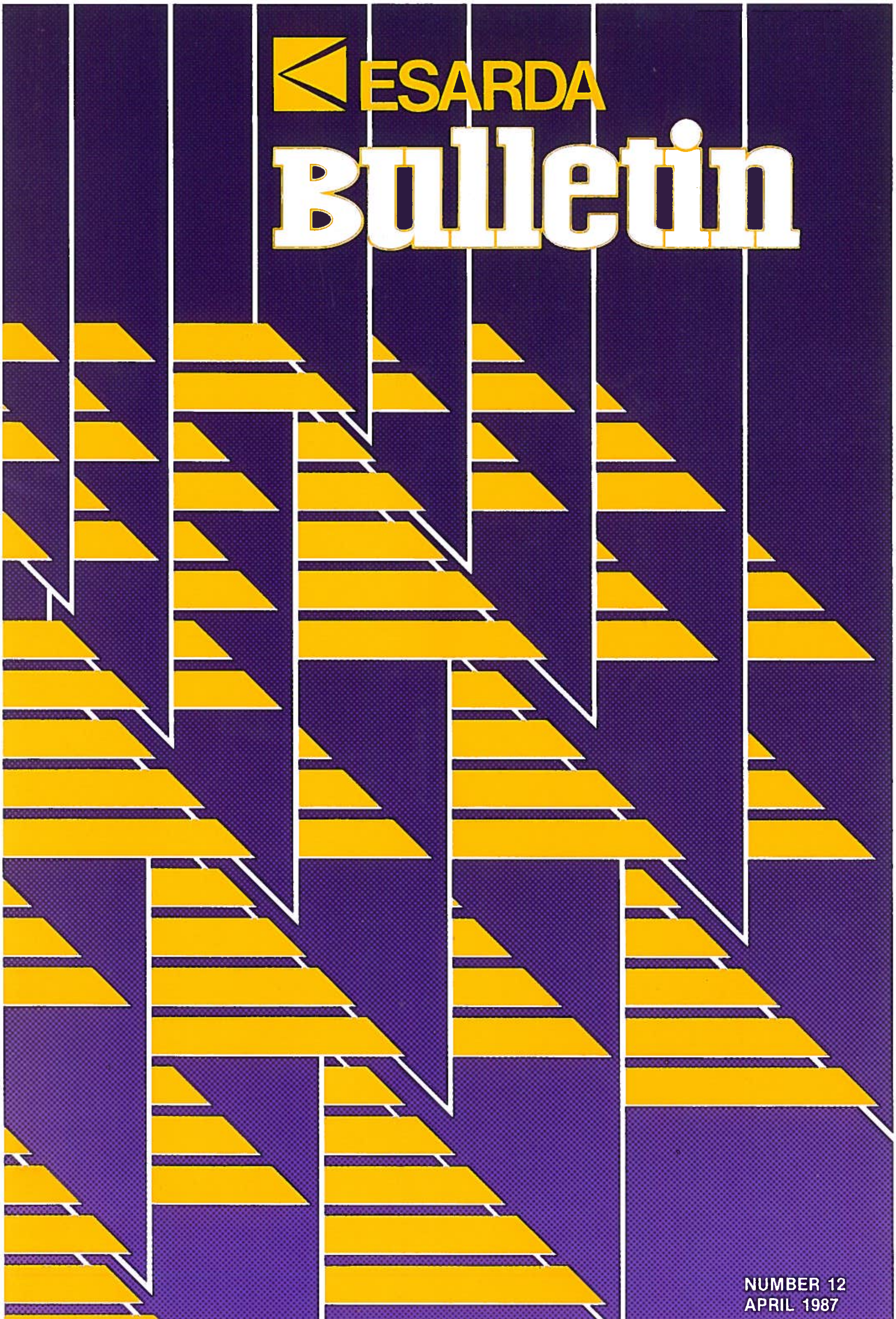

ESARDA BULLETIN, Issue No.12, 1987

Notes on Venue for the

9th ESARDA Symposium,

London, 12-14 May 1987

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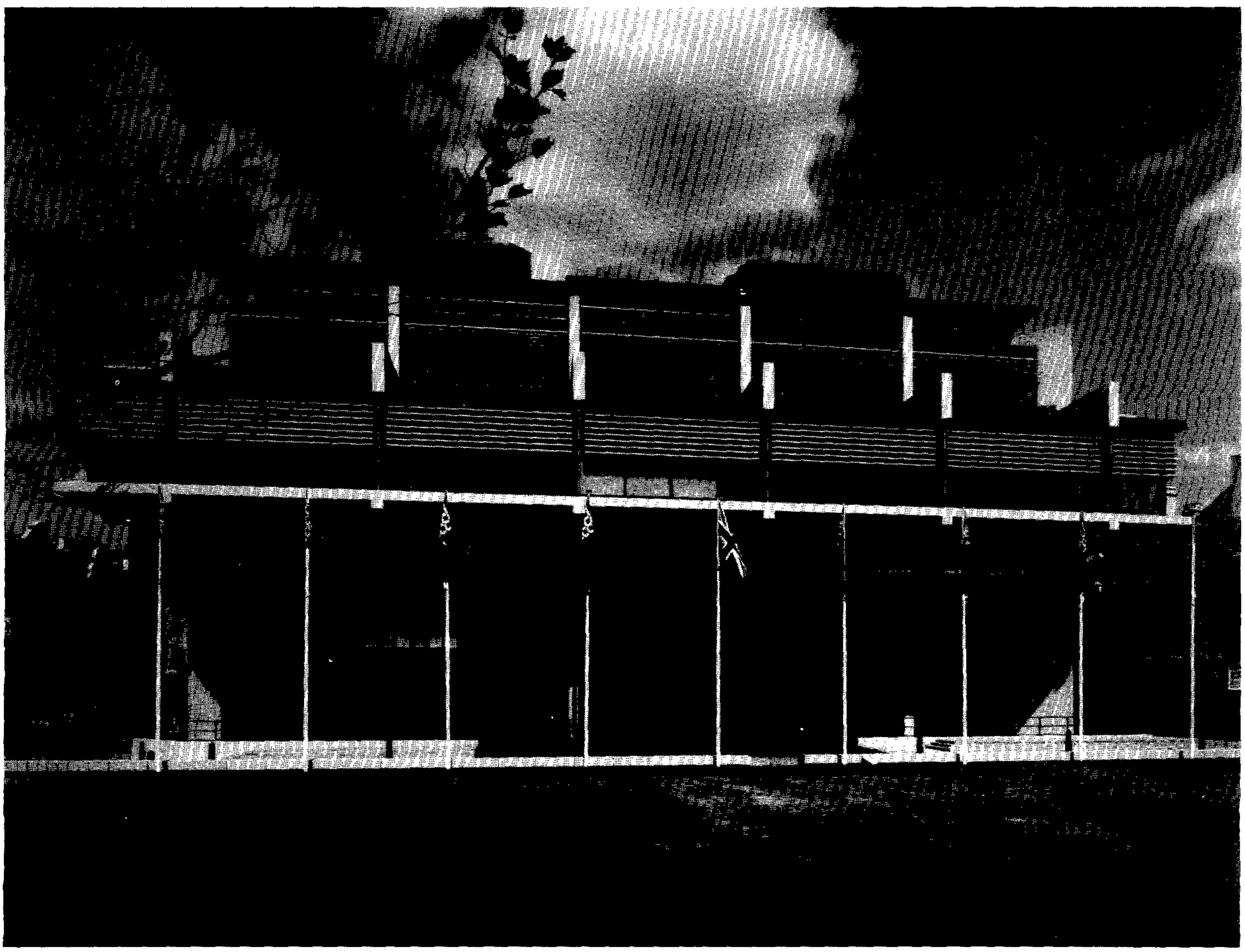
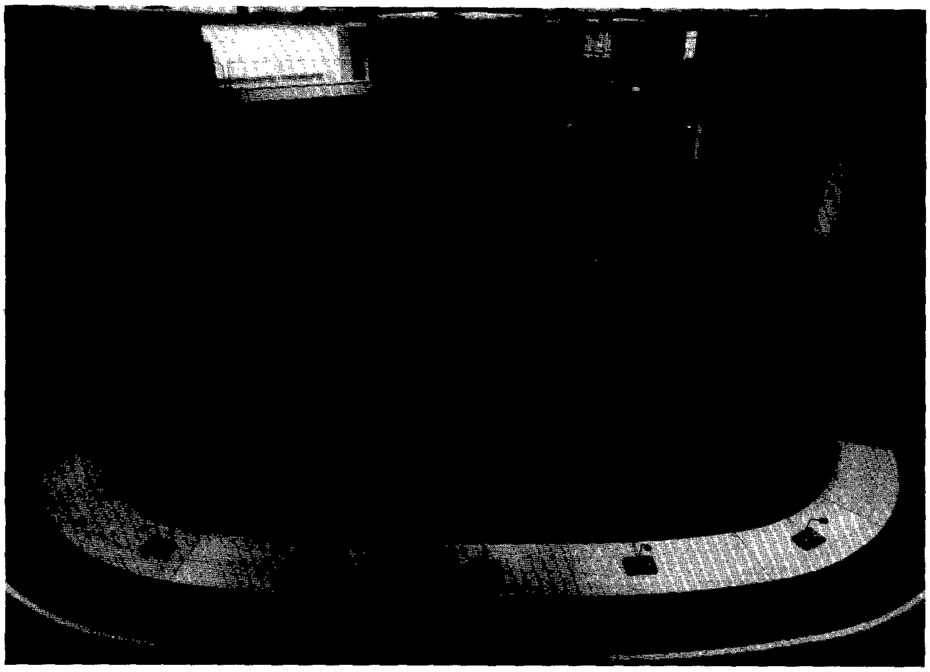
The first ESARDA Symposium on Nuclear Safeguards and Nuclear Material Management was held in Brussels on 25-27 April 1979. An annual meeting has been organized by ESARDA ever since then. This series followed previous ESARDA initiatives such as the Symposium of Rome, 7-8 March 1974, on Practical Applications of R&D in the Field of Safeguards.

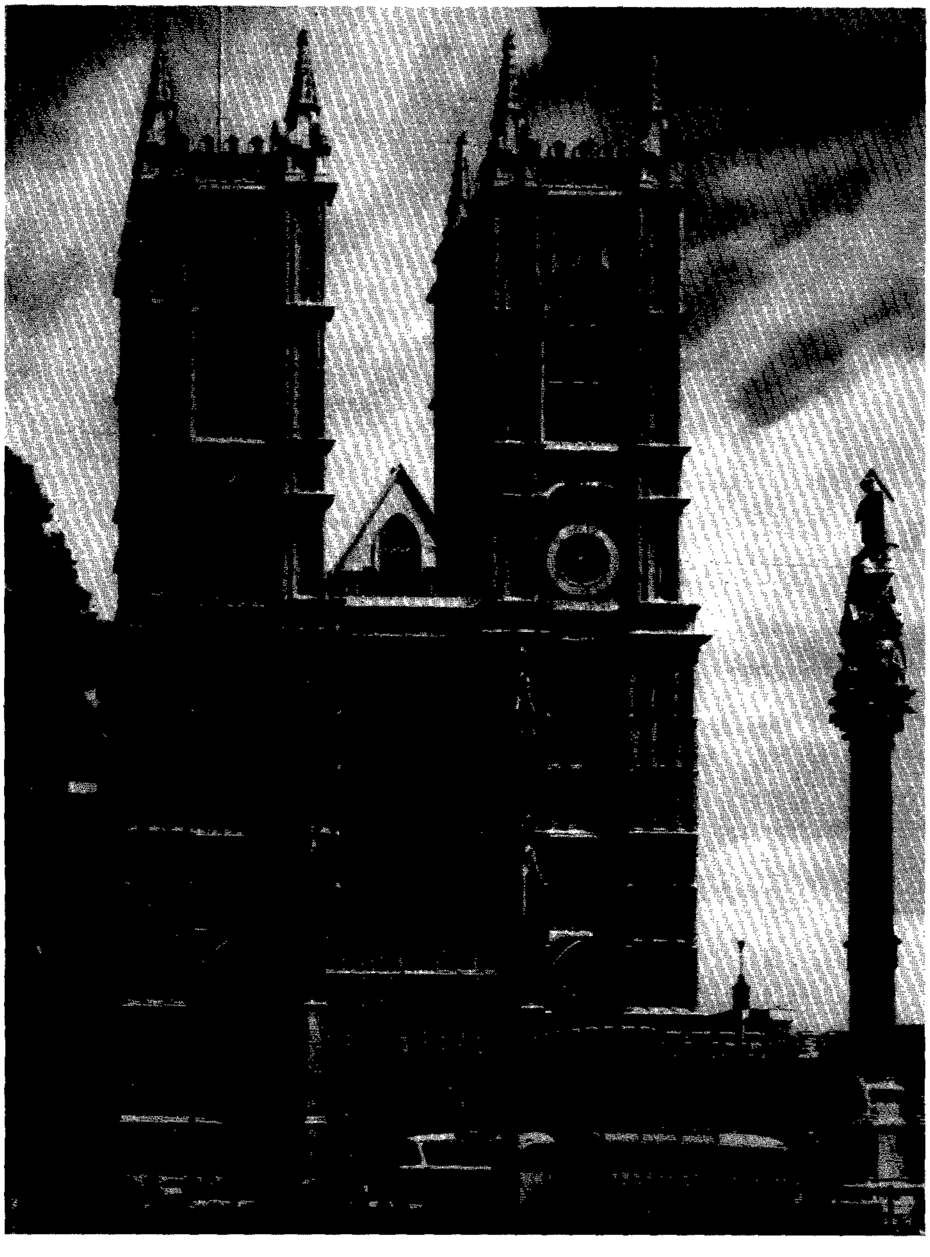
The 2nd Symposium was held in 1980 in Edinburgh and the 3rd Symposium in Karlsruhe in 1981. The 4th annual meeting was organized at the Dutch research centre of Petten in 1982 in the form of a specialists meeting on "Harmonization and Standardization in Nuclear Safeguards". Three full symposia were then organized in Versailles (France) in 1983, in Venice (Italy) in 1984 and in Liège (Belgium) in 1985. The 8th ESARDA meeting was held in Copenhagen in 1986 and was a restricted meeting of ESARDA working groups on the "Capabilities and Objectives of the Use of NDA-DA-C/S Measures in Safeguards".

The 9th ESARDA Symposium will be held in London at the Queen Elizabeth II Conference Centre situated in Broad Sanctuary, Westminster, on 12-14 May 1987. Within sight of the Houses of parliament, the Centre is the result of painstaking research and careful planning. It was formally opened by H.M. the Queen on 24 June 1986. It contains several suites of rooms and is capable of hosting several conferences and meetings simultaneously.

ESARDA will have the use of the Whittle room on the 3rd floor for the oral presentations and adjacent rooms will be provided for the poster sessions and for informal use







during the symposium.

The setting of the Queen Elizabeth II Conference Centre is as impressive as its architecture; looking out from the windows over 13 centuries of ecclesiastical history, represented by Westminster Abbey, one begins to appreciate the panorama of its historical surroundings.

Like all great churches in England, Westminster Abbey has been centuries in the making. The present building was under construction from 1340 to the completion of the towers of the west front in 1738-9. The Abbey has witnessed the coronation of every English monarch from Edward I to Queen Elizabeth II. The Coronation Chair is one of many monuments and memorials on display in the Abbey.

In front of the Centre is the site of one of the world's first publishing houses, founded by Caxton towards the end of the 15th Century.

Close by, across Parliament Square, the Houses of Parliament present a complex of elaborate spires, turrets and towers. Impressive is St. Stephen's Tower, known throughout the world as Big Ben.

In fact, the medieval appearance is spurious. Visitors are often surprised to learn that the majority of the buildings were completed between 1840 and 1850, the exception being the core of the complex, Westminster Hall, which was built in 1097.

On the far side of the Houses of Parliament flows the River Thames, which is still tidal at this point and best seen at full tide when the water fills the channel between the embankment walls. The nearest viewpoint is from Westminster Bridge. Here in 1802, the poet Wordsworth was moved to write his sonnet beginning :

"Earth has not anything to show more fair;
Dull would he be of soul who could pass by
A sight so touching in its majesty"

Perhaps our visitors will feel something of the same spirit !

Measurements of UF_6 Isotopic Ratios by Transportable Quadrupole Mass Spectrometer : Three Years Experience*

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* Work presented at the Poster Session of the IAEA Int. Symp. on Nuclear Material Safeguards (Vienna, 10-14 Nov. 1986) with the code number IAEA-SM-293/148.

Abstract

Since 3 years the DCS Luxembourg is using the mass spectrometry technique in a mobile version for the "in-field" verification of UF_6 enrichment. Even with the severe conditions the transportable mass spectrometer is exposed to, the results obtained remain precise and accurate. Only a limited number of standard reference materials are used to obtain results acceptable for safeguards purposes, even if a slight bias exists due to the difference in ^{235}U content between the analysed samples and the standard.

The experience gained and the results obtained during inspections in different enrichment plants are presented and discussed.

Introduction

The verification of the declared enrichment in the different phases of the nuclear fuel cycle is one of the important tasks in Euratom Safeguards. The taking of samples and their transport to the analytical laboratories suffer, however, from two major drawbacks : transportation problems for both the safeguards authorities and plant operators and lack of timeliness in obtaining results from the laboratories concerned.

To avoid these problems, efforts have been made to adopt in-field measurements with instruments and techniques capable of performing analysis of fissile materials with the required accuracies.

Mass spectrometry is the technique commonly used for isotopic composition measurements. An accuracy of 0.1% is easily attained under laboratory condition. The in-field use of this technique adds many constraints such as frequent transportation of the mass spectrometer by truck and consequent vacuum breaks, furthermore the instrument must be installed and be operative in a minimum of time, must be easy to decontaminate and should require just a minimum of infrastructure services from the visited plant.

In the light of these criteria a quadrupole mass spectrometer was chosen. Characteristics of the instrument, laboratory and short-term in-field tests have already been presented /1,2,3,4,5/, nevertheless long-term performances must be checked for practicability, cost, contamination and stability of results. The results obtained during three years of UF_6 isotopic ratio measurements, allow us to confirm the reliability of the technique, even with the severe conditions the instrument is exposed to.

Experimental

The instrument used by DCS (Fig. 1) has the following basic characteristics :

- inlet system : 4 independent inlets with 4 expansion containers
- analyser : quadrupole filter with 500 amu
- ion source : electron impact
- detector : Faraday cup
- vacuum system : ionic pump - turbo molecular pump boosted by a rotary pump giving a working pressure. with liquid nitrogen cooled trap, of 10^{-7} to 10^{-8} torr
- computer system : PDP11/03 (Digital Equipment Corporation).

The instrument is transported by a small truck with an elevator system and is installed at the visited plant the day of its arrival. It reaches its working conditions during the following night and the measurements can normally start the second day.

Enrichment measurements are performed against standard reference materials. The ratios $^{235}U/^{238}U$ are measured for the sample (R_s) and for the reference material (R_r) respectively.

The enrichment of the sample (E_s) is then

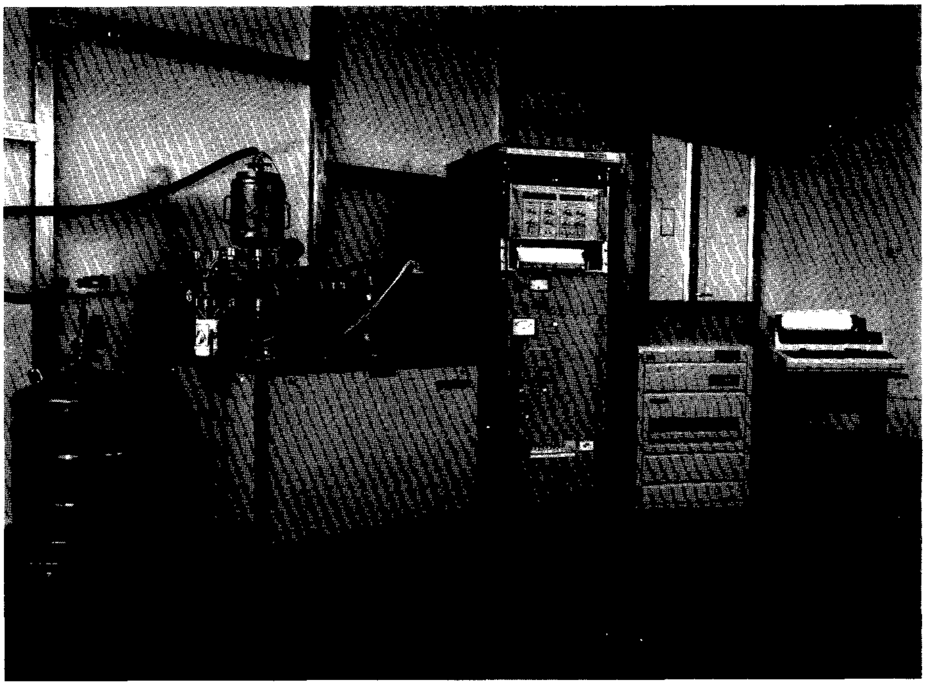

Fig. 1. Transportable quadrupole mass spectrometer for UF_6 isotopic analyses used by DCS Luxembourg

calculated by the relationship

$$E_s = E_r(R_s/R_r).$$

Measurements of sample and reference material are alternatively performed and a linear correction takes account of the sample consumption. A typical run includes nine measurements of the sample and nine measurements of the reference material which are performed in about 40 minutes.

Each measurement result is the arithmetic mean of 450 single measurements corresponding to a true integration time of about 2 seconds.

A complementary program allows the computer to compare the final result of the measurements with the operator declaration. The results of this comparison are printed out in the form of

$$\Delta\% = \frac{\text{found value - declared value}}{\text{declared value}} \times 100$$

Under routine conditions, no account is taken for minor isotopes ^{234}U and ^{236}U. Nevertheless, their presence is checked before any measurement in order to verify that their relative intensities do not affect the validity of the comparison.

In order to avoid the transportation of UF_6 standards used during the measurement campaign, sets of 4 reference samples supplied by JRC-Geel having ^{235}U wt% of 0.4328, 0.7111, 3.2339 and 4.9909 are permanently stored under seal in the different nuclear facilities.

The majority of the UF_6 introduced in the instrument is condensed on the liquid nitrogen cooled traps. These need to be dismounted and cleaned at the end of each measurement campaign.

The rest of the contamination, located in the inlet pipes and in the upper part of the diffusion pump, is easily eliminated during routine maintenance. Between two campaigns and after decontamination, the instrument is maintained under vacuum conditions. This decision was taken after we observed that venting the instrument increases the moisture content on the wall of the vacuum chamber provoking a higher decomposition rate of the first analysed UF_6 sample. The subsequent appearance of HF caused a corrosion of the quadrupole rods.

Results and Discussions

Precision and accuracy of the measurements

After 10,000 km transportation in various countries and more than 200 analysed samples the precision of the measurements does not show any degradation. The relative internal standard deviation after three years is still of the order of 10^{-4} for a 3% range enriched sample.

Figure 2 shows typical differences between declared and measured values obtained in three different enrichment installations over the last three years. A relative external standard deviation of more than 0.5% is considered as significant, and the measurement is repeated in such as case.

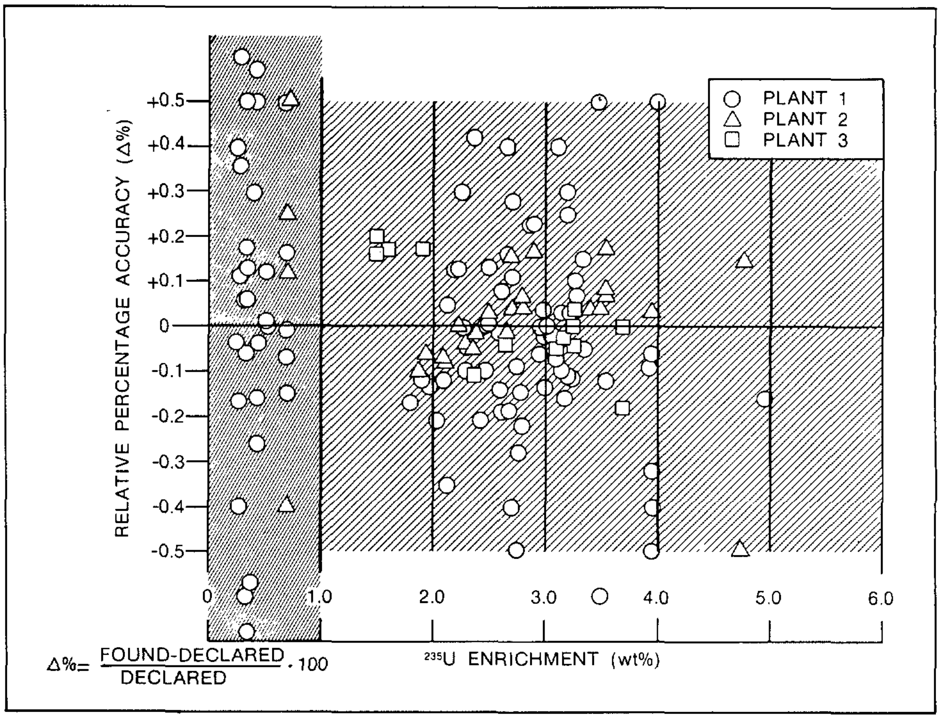

Fig. 2. Typical differences between declared and measured values obtained in three different enrichment plants

For depleted uranium, this variation is higher and is probably due, for a part, to the inhomogeneity of this kind of material.

For "product" material measurements, the reference standard 3.2339 wt% ^{235}U enrichment is normally used. For this order of enrichment, a difference up to 100% between the certified enrichment value and the sample value, affects the accuracy by less than 0.2%. This figure is still acceptable for safeguards purposes.

Figure 3 shows the behaviour of the accuracy against the sample to standard relative differences for the 3.2339 wt% reference sample.

For measurements of "depleted material", the difference between sample and reference value is even more critical.

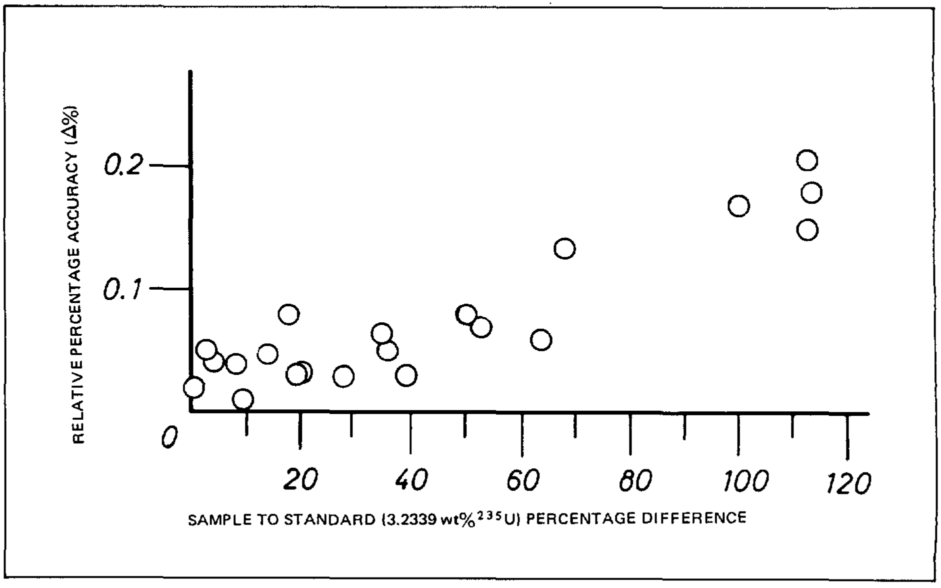

Fig. 3. Behaviour of the accuracy against sample to standard relative difference for the 3.2339 wt% enriched reference sample

Differences of the order of 60% between the sample and the reference material lead to a bias of about 0.7%.

The distribution of the accuracy values against the sample to standard relative differences for the 0.4328 wt% ^{235}U reference sample is given in Fig. 4. The values are scattered over a wide band and increase with the difference more rapidly than those obtained with the 3.2339 wt% ^{235}U standard.

The estimated analysis costs with the transportable mass spectrometer have remarkably confirmed our previsions /5/. For about 60 samples/year the analysis costs are equivalent to those performed in laboratory, i.e. 800 $/sample.

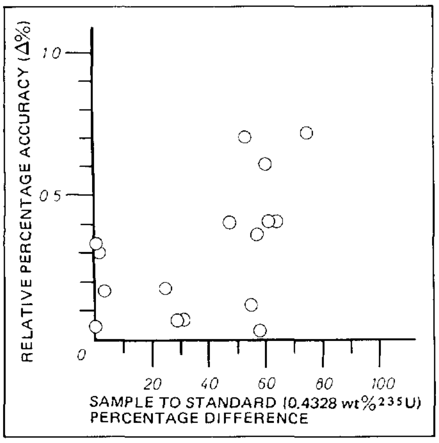

Fig. 4. Distribution of the accuracy against sample to standard relative difference for the 0.4328 wt% reference sample

Future developments

In its present version the instrument is bulky and occupies more than 3 square meters. In order to reduce the instrument size the actual computer will be replaced by a miniature version which will be mounted in the spectrometer own electronics cabinet thus reducing the "foot print" of the instrument by 1 square meter.

In the inlet system, the use of the expansion containers guarantees the flow stability of the source, but is sample consuming. Only a few percent of the sample is utilized for the analysis itself, the rest is wasted. To avoid this useless consumption of sample, a flow controller system is presently under study and may eventually replace the expansion containers.

The positive experience gained with the use of the transportable mass spectrometer for UF_6 analysis is being continued with the in-field utilisation of a thermo-ionic quadrupole used for measurements of solid U and Pu samples. Recent provisional results show an accuracy of better than 0.5% for U enrichment determination.

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Final Disposal of Spent Fuel Safeguards Aspects

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Introduction

IAEA Safeguards under INFCIRC 153 are based on material accountancy as a safeguards measure of fundamental importance, with Containment and Surveillance as important complementary measures (INFCIRC 153 paragraph 29) so far. This approach has provided an optimal solution. Recent developments in the nuclear fuel cycle, especially in connection with safeguards in plants where nuclear material is inaccessible, have given rise to considerations of alternatives to this basic approach, by putting more emphasis on Containment/Surveillance. To illustrate this problem the paper dicusses as a very significant example safeguards aspects of a final disposal for spent fuel.

The possibility of the final disposal of spent fuel elements is considered complementary to the commercial reprocessing of fuel elements. Within the framework of the R&D programme of the Federal Republic of Germany on "Complementary Waste Management Techniques" /1/ the emplacement of spent fuel and its unmonitored non-retrievable storage in a salt dome has been studied under different aspects where commercial, technical and safety problems were of major importance. Another important part of this study covered the analysis of problems with regards to the implementation of international safeguards. In this paper the design of the final repository is described and three safeguards models for the disposal concept for LWR spent fuel are proposed. In connection with a diversion analysis possible solutions for unsolved problems are discussed /2/. In addition, before disposing of spent fuel in a final repository, it has to pass through a conditioning facility. The safeguards relevant features of a conditioning facility for spent fuel are described and a possible safeguards approach is discussed.

General considerations

Since the final disposal of spent fuel has been investigated under the aspect of non-retrieval the question of termination of safeguards for spent fuel arises.

The criteria for terminating IAEA safeguards are laid down in paragraphs 26 (C) of INFCIRC/66 and 11 of INFCIRC/153 corresponding to Art. 11, Verification Agreement (VA) : "... upon determination by the Community and the Agency that the material has been consumed, or has been diluted in such a way that it is no longer usable for any nuclear activity relevant from the point of view of safeguards, or has become practically irrecoverable."

Although the technical concept does not foresee retrieval of the nuclear material a study has not excluded its technical possibility. The conditions of paragraph 11, INFCIRC/153 are thus not applicable to spent nuclear fuel in the described final repository.

For certain types of spent fuel (e.g. AVR, THTR or special types of light-water reactor) reprocessing is either not envisaged or not economical. For such fuel elements direct final disposal is therefore necessary. The question thus arises whether a termination of safeguards is possible for such nuclear material. Although the conditions of paragraph 11 are not fulfilled paragraph 35 of INFCIRC/153 (Art. 35, VA) could come into effect, which says :

"... Where the conditions of that paragraph are not met, but the State considers that the recovery of safeguarded nuclear material from residues is not for the time being practicable or desirable the Agency and the State shall consult on the appropriate safeguards measures to be applied. It should further be provided that safeguards shall terminate on nuclear material subject to safeguards under the Agreement under the conditions set forth in paragraph 13 above, provided that the State and the Agency agree that such nuclear material is practically irrecoverable."

It may be concluded that also for the above mentioned types of fuel a termination of safeguards is not possible, but an application of simplified safeguards measures is conceivable.

In the case of non-retrievable final disposal of spent fuel, where finally no access for verification purposes is possible, safeguards approaches have to rely on C/S-measures only. However, results of recent discussions in the IAEA on the implementation of safeguards in sensitive facilities for reprocessing and enrichment have indicated that safeguards concepts making intensive use of containment/surveillance systems are not acceptable /3/. Since precisely this conception with intensive C/S could be of essential significance in the case of a direct final repository, a conflict might arise. The IAEA would have to make considerable cuts and reorientations in their previous safeguards philosophy in order to solve this issue. Only a few problems can be mentioned here such as design verification, availability and reliability of instruments, verification of nuclear material in the case of instrumentation failure, and internal diversion.

In spite of all these difficulties it must be seen that pursuant to safeguards agreements, e.g. INFCIRC/153, all nuclear facilities, i.e. also a direct final repository for spent fuel, would in principle have to be internationally safeguardable. However, in this case, considerable cuts would have to be made for such a safeguards concept in the "effectiveness" demands as currently discussed in the IAEA. An additional problem is connected with the application of "pure" C/S concepts. There seems to be a need for improving safeguards measures already "up-stream" in the fuel cycle, which means for instance that more intensive safeguards measures are necessary in the reactor power station, intermediate storage and the conditioning facility.

Already in INFCE these problems have been recognized as for instance INFCE Group 7 report /4/ mentioned that in the long term the effectiveness of safeguards measures is questioned since the post-operational phase lasting for centuries will be determined by factors which are hardly foreseeable, such as :

- alterations in the institutional and social system,
- large inventory of fissile material in repositories for spent fuel, decrease of radioactivity and thus better possibilities for

recovering the fissile material,

- development of new technical safeguards measures (i.e. procedures and equipment),
- possible technological developments to accelerate the recovery of highly diluted waste,
- degree of integrity of canisters with spent fuel in shut-down geological repositories and possibilities for recovery,
- later incentives for recovering the fissile material from spent fuel for energy generation purposes.

On most of these factors no detailed predictions can be made. It is therefore not possible from current perspectives to make a decision on the possibility of monitoring a direct final repository in the post-operational phase or terminating safeguards.

Design and safeguards aspects of the conditioning facility

In analysing safeguards features of the different facilities for the final disposal one must take into account that in both facilities for conditioning and final disposal the detailed design elements are still under discussion. The following elaboration is therefore mainly a model strategy.

Although the planned facility is designed purely as a pilot plant for development and demonstration purposes and is not capable of a high throughput operation, all equipment needed for industrial scale operation is included. All operations are carried out entirely under dry conditions. The reference conditioning process for LWR spent fuel is shown in Fig. 1. After passing a storage area for casks spent fuel is unloaded and stored in racks. In a hot-cell fuel assemblies are disassembled into rods. These are filled into trays and loaded together with their compressed structural elements into containers (POLLUX) which meet final disposal requirements. Another option is to cut single rods into pieces of approximately 1 m length, which are then filled into cans, and the structural parts of the fuel assemblies solidified in drums. As a multi-purpose facility, the pilot plant is also designed for preparation of HTR fuel and vitrified high level waste.

The design and operation of the conditioning plant suggest to divide the plant into 2 material balance areas :

- MBA 1 : loading and unloading area
- MBA 2 : process area.

For MBA 1 one can assume that the incoming transport casks carry the electronic sealing system VACOSS. Besides item accountancy, optical surveillance is a major element in this MBA. C/S measures are the main part of the safeguards approach after loading of the POLLUX casks access to the fuel rods for verification is not possible. The design of the plant ensures that only movements with nuclear material are carried out which are in accordance with operational purposes. Sealing of the loaded POLLUX cask with the electronic sealing system VACOSS will finalize the safeguards measures in the conditioning facility.

In MBA 2, after unloading of e.g. LWR fuel elements visual inspection with identification of serial numbers is possible. Also in this MBA the design features of the plant and C/S measures will ensure the continuity of knowledge.

In summary, it can be recognized that in principle no unsolved safeguards problems exist for a pilot conditioning facility with extremely low nuclear material throughput. However, taking into account a clear trend towards more sophisticated and redundant C/S as well as NDA-application, even in facilities where material is difficult to access, safeguards can pose serious impacts on plant design and operation

Design of the final repository

The geological repository is constructed in a virgin salt dome, the reference concept envisaging horizontal, non-retrievable emplacement in tunnels.

Access to the repository is obtained via two shafts. The first shaft serves to transport the salt, material and personnel. The second shaft is envisaged for emplacement and special transports. There will be only one emplacement floor at a depth of between 700 m and 900 m. The schematic view of the storage area for spent fuel is shown in Fig. 2. Access to the emplacement area is obtained by driving two parallel access galleries joined by connection drifts at intervals of 200 m. Starting from the connection drifts, the emplacement tunnels are driven parallel to the access galleries. Before beginning emplacement, emplacement galleries will only be driven starting from the emplacement connection drift furthest from the shaft. Emplacement galleries are driven from the next connection drift at the same time as emplacement is implemented in the first sector of the emplacement field.

It is foreseen that the geological repository will also have an emplacement field for radioactive waste, a material category which is not subjected to international safeguards.

The flow of the final disposal packages (FDP) is shown in Fig. 3. The FDPs come in via public railways. The aboveground facilities of the final repository comprise a buffer zone, which can obtain a maximum of nine FDPs in their transport flasks. In the reloading area the FDPs are separated from the flasks and loaded into a rail-bound internal transport truck. The truck is driven to the shaft, loaded into the hoisting cage and transported to the emplacement level. Underground rail-bound transport is terminated at the junction of the access gallery and the emplacement connection drift. Transport through the emplacement connection drift to the emplacement gallery is railless, effected by an emplacement vehicle. After emplacing the package, the gallery section with the package is backfilled (mechanical or pneumatic stowing). When all the galleries of an emplacement sector are occupied by packages and filled-in the connection drift and ventilation galleries are also back-filled. After terminating

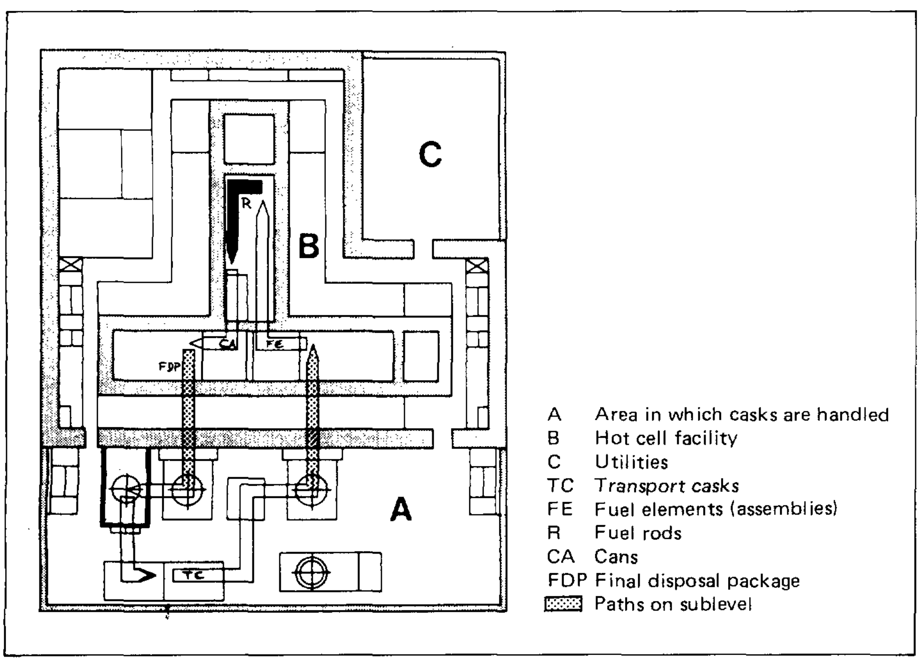

Figure 1. Pilot conditioning facility. Layout of 4.5 m level. Nuclear Material Flow, Process of LWR Spent Fuel

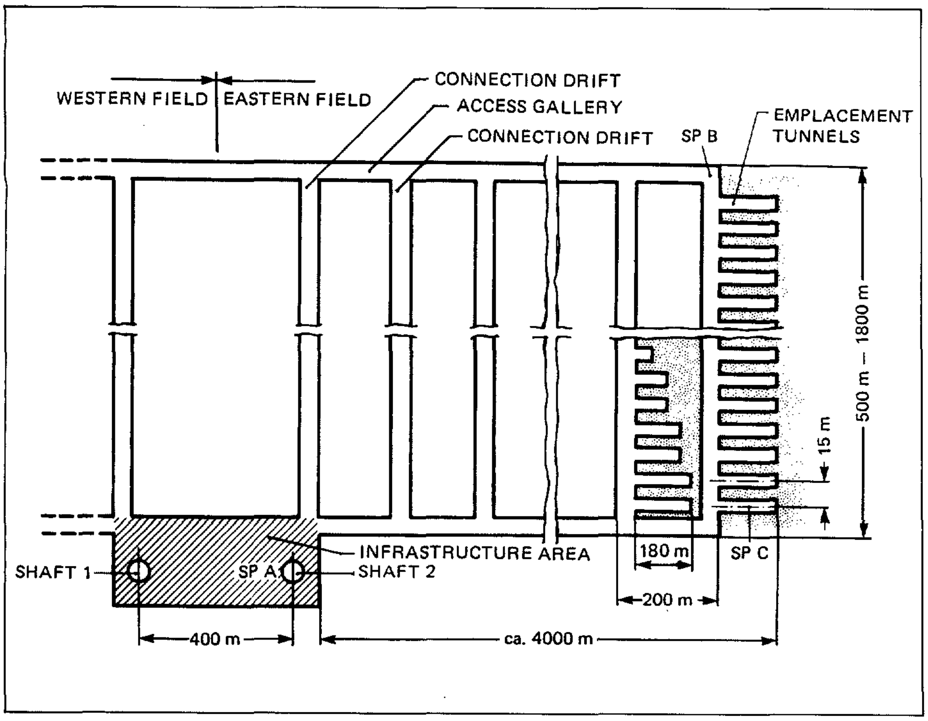

Figure 2. Schematic view of emplacement floor

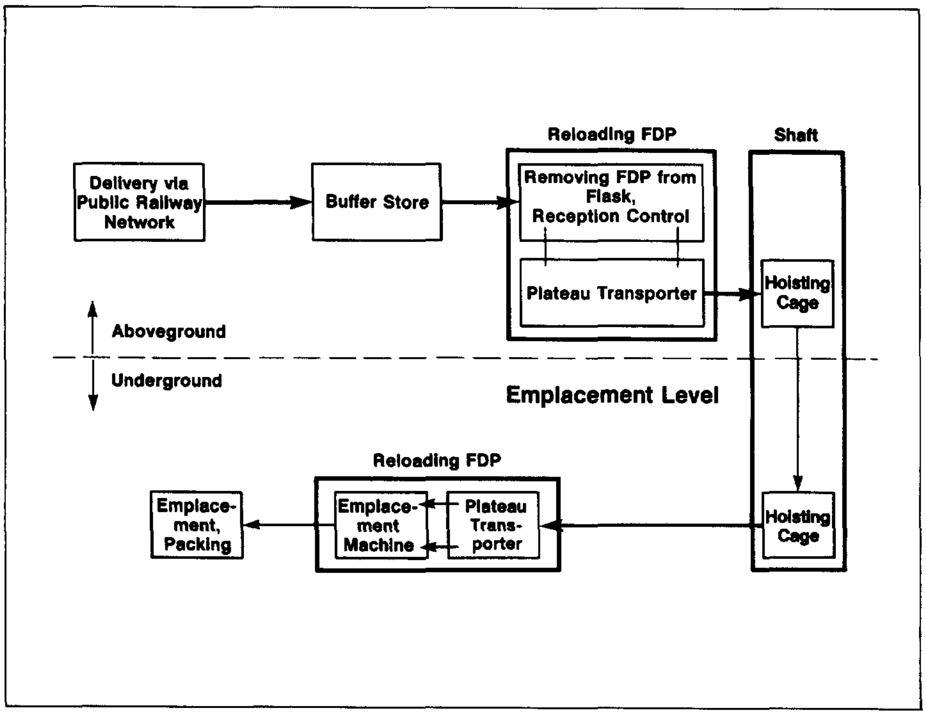

Figure 3. Material flow diagram

emplacement operation all the galleries and cavities are backfilled, in the same way as the shafts. It is intended to operate the mine for 50 years at an emplacement rate of 437 FDPs per year. The fissile content will accumulate to about 480 t, including 270 t ^{235}U, 200 t ^{239}Pu, and 10 t ^{241}Pu.

Safeguards models

The geological repository for final disposal of spent fuel has its nuclear material inventory contained in individual items, the final disposal packages (FDPs). The Agency safeguards concepts for such facilities are based on items accounting. Since the nuclear material contained in FDPs cannot be directly verified, applicable accounting measures are

- item counting
- item identification
- verification of the integrity of the item.

The precondition for the applicability of such a concept is the verification of nuclear material contents of the FDPs in the conditioning facility and application of appropriate containment/surveillance measures to the FDPs in order to allow for an unlimited extension of the validity of the final material verification by verifying the identity/integrity of the item. The shipper data from the conditioning facility are taken over unaltered as data on material quantity and composition. Apart from the possibility of exempting nuclear material from safeguards (INFCIRC/153, Art. 11 and Art. 35) and apart from material being returned for rework, the inventory changes to be recorded only consist of additions. The repository has only one material balance area.

In studying the safeguards problems related to a final repository for spent fuel it seemed to be advantageous to establish three safeguards models. They are differentiated by the degree of authorized access for IAEA inspectors. Thus in Model 1 access is restricted to aboveground facilities, Model 2 envisages limited access to the underground facilities and Model 3 unrestricted access to all underground facilities, including the waste disposal area.

In Model 1 the inspector's access is limited to strategic points above ground. These strategic points are the key measurement points, the reloading facility as well as the bank eyes of the mine shafts. The essential element of this model is that, after the material has been taken under ground, recovery or an internal diversion within the mine is ruled out. However, before the material can be released from safeguards, proof of non-recoverability must be presented. If this cannot be presumed then routine inspections of the site will be required during the post-operational phase in order to monitor activities which could indicate a reopening of the mine or other measures for recovering the material.

Model 2 comprises Model 1 and the following additional underground strategic points: pit bottom of both shafts, intersections of the access galleries with the emplacement connection drifts and the junctions of the emplacement galleries with the respective connection drift. These underground strategic points enable the inspector to safeguard the underground nuclear material flow at various stages of intensity. Largely the same restrictions as for Model 1 apply to this model. A termination of safeguards with backfilling of the gallery would have to be possible, or the geological repository itself would have to be regarded as a sufficiently safe barrier. The access of inspectors to strategic underground points would indeed present a serious obstruction to a diversion in the repository, but it cannot be ruled out with sufficient certainty.

Model 3 comprises Model 2 and moreover as an additional measure the access of inspectors to all underground facilities and installations. Measures for containment and/or surveillance are thus suggested in all the aboveground and underground facilities and installations of the final repository, including the waste storage area and the infrastructure (workshop, etc.).

Safeguards applied would comprise activities such as

Pre-operational phase

- design verification before startup

Operational phase

- design reverification after driving a new tunnel
- identity/integrity verification of the FDP at entry (reception control)
- camera monitoring at the unloading facility above ground to prevent undeclared unloading processes (replacement by a dummy)
- C/S measures (camera, detectors) at the strategic points at the two shafts to prevent undeclared material flow (backflow)
- camera monitoring at the underground unloading points (from the hoisting cage on rails along the access gallery; from the transport truck on an emplacement machine without rails) to prevent replacement by a dummy
- camera monitoring at the entrance to the emplacement tunnel to observe the emplacement process and prevent recovery until the tunnel has been sealed
- recording the duration and speed of run in the case of the hoisting engine, transport truck and emplacement machine as back-up measures for camera surveillance (if available and necessary)
- inspector access to the strategic points underground either on the random basis or at any time.

Post-operational phase

- termination of safeguards after backfilling all shafts, decommission of aboveground facilities as well as demonstration of non-recoverability (if recognized as impossible by means of mining technology)
- routine examination of the site by visual inspection for safeguarding against activities which could indicate a reopening of the repository or other measures for recovering the material.

The safeguards models have to be subjected to a comprehensive diversion analysis, in order to allow for an effectiveness evaluation.

In the phase of above ground transport diversion strategies may comprise the following possibilities :

- diversion of complete final disposal packages (FDP)
- replacement by dummy FDP
- removal of nuclear material from FDP.

The same diversion activities may apply for the phase of under ground transport, as long as no backfilling on site has taken place. In addition, the diversion would not be complete, unless one or more of the following activities are achieved :

1. retransportation of FDPs to the surface
2. repacking the nuclear material in an underground hot cell facility and retransportation through existing facility
3. reprocessing in an underground facility
4. clandestine connection to the surface.

In the phase of storing the FDPs in backfilled areas of the repository, however before closing the mine, diversion strategies would be directed at access to the emplaced FDPs by uncovering FDPs via existing transport paths or by-passing dams via exploratory floor, waste store, etc. The possibilities of recovering the nuclear material would then be identical with the above mentioned activities 1. through 4.

Finally, in the post-operational phase of the repository diversion could only be achieved via purpose-built boreholes or by sinking a shaft or by reopening the repository from a considerable distance to recover a number of FDPs.

Discussion and conclusions

Based upon these three safeguards models for the final disposal of LWR spent fuel, a diversion analysis was compiled as well as an evaluation of effectiveness leading to the following results:

Sufficient safeguards can be ensured both in the phase of aboveground transport as well as in the phase of transporting the FDPs under ground until they are filled-in on site. During the operational phase of the repository safeguards on FDPs already backfilled can consist of permanent design verification (Model 3). However, unsolved problems can be seen in evaluating their effectiveness. The same is true of verifying the integrity of the shut-down geological repository in the post-operational phase. The safeguards effectiveness during the various operational phases of the final repository is shown in Table I.

Four approaches are suggested and discussed for solving the safeguards problem.

An initial approach is perceived in altering the existing IAEA safeguards philisophy. But presently the IAEA considers it necessary to quantify objective variables by compiling numerical detection objectives (significant quantity, detection time, probability of detection, probability of false alarms). The probability of detection is the essential variable in the IAEA safeguards towards which the planning of safeguards, employment of resources and evaluation of effectiveness are oriented. Since there is currently no procedure for quantifying the probability of detection in applying containment and surveillance measures, safeguards models which are largely or, as required in the case of the final repository, almost exclusively based on C/S measures cannot be objectively planned in this model nor is their effectiveness computable. This leads to them being classified as unacceptable by the IAEA.

A second approach is seen in the further development, and possibly redevelopment, of safeguards elements. In the operational phase of the final repository the problem consists in communicating a quantifiable certainty to the safeguards authority by suitable measures that the emplaced material is still present. Strictly speaking, this quantification is only possible for accountancy measures. No methodology has yet been developed for numerically determining the information content of C/S measures; the error associated with C/S verification cannot be precisely specified. This problem can generally be mitigated in other facilities by implementing material verification in principle by accountancy measures and by only employing C/S measures for subsidiary quantities of material and for limited periods as a supportive measure. These restrictions (limitation to subsidiary quantities and defined periods) must be dispensed with in the case of the final repository. Safeguards would thus only be possible with a pure C/S concept and there is no contractual nor methodological basis for this. I.e., even presuming that safeguards elements were to be redeveloped or further developed, thus enabling C/S-supported monitoring of the emplaced material, its inclusion in the safeguards system would only be possible as a supplementary measure. On their own they do not represent a basic solution to the problem under consideration.

Adaptation of the reference concept to the currently valid safeguards practice is discussed as the third approach. The starting points for this discussion are conditioning the material in such a way (dissolution and dilution) that the termination criteria for safeguards are fulfilled, or emplacing the material in such a way (recoverable) that it remains accessible for verification measures. Both methods of treatment are unacceptable as realistic alternatives. By dissolving the fuel and conditioning in the form of PAMELA ingots the capacity e.g. of the Gorleben salt dome would not even be sufficient for a single annual throughput of 700 t of nuclear fuel.

Final Disposal Phase	Safeguards Effectiveness Model 1	Model 2	Model 3
Phase 1: Transport above ground, leaving the conditioning facility - beginning shaft transport	acceptable		
Phase 2: Transport below ground, beginning shaft transport - backfilling on site	acceptable since FDP's still accessible		
Phase 3: Storage during the operational period, backfilling the FDP's - sealing the geological repository (backfilling the shafts)	unsolved problems		
Phase 4: post-operational phase, after backfilling the shafts	unsolved problems		

Table I. Safeguards effectiveness during the various operational phases of the final repository

Approach	Model 1	Model 2	Model 3
Alterations to the existing IAEA safeguards philosophy	IAEA would have to accept purely a C/S safeguards concept		+ intensive permanent facility safeguards
	currently no solution		
Further and possibly redevelopment of safeguards elements	no basic solution to the safeguards problem		
Adaptation of the reference concept to valid safeguards practice	no realistic possibility in sight		
Institutional approaches	additional proliferation barrier but not a solution to the safeguards problems		

Table II. Assessment of the solutional approaches

By emplacement in such a way that the material would remain accessible for further verification, apart from the technical feasibility, the essential objectives of the final repository concept would not be fulfilled, namely isolating the material from the biosphere and from possible further human access. Accessible underground emplacement would probably raise so may problems for reasons of heat removal and rock stability that this could no longer be regarded as a modification to the reference concept but would rather require compiling a new concept.

As a fourth approach, possibilities for solutions in the institutional sector are discussed. The starting point for institutional approaches is the fact that in order to implement a diversion a considerable amount of organizational work must also be undertaken in addition to the necessary technical measures. By forms of multinational co-operation, additional barriers could be erected in the organizational sector which would make a diversion more difficult and would increase the risk of detection. A further aspect is that by extending international involvements, the states would probably be more vulnerable to sanctions.

Consideration of institutional aspects received essential impulses through the INFCE Conference and is reflected in the IPS Working Group. It must, however, be remembered that institutional aspects are regarded by the IAEA as supplementary measures and not as an alternative to stringent technical monitoring. Institutional models with multinational co-determination or co-operation undoubtedly represent an approach to general NP problems of a final repository due to the associated proliferation barrier. However, they are not appropriate for solving the safeguards problem. In this connection the special role of Euratom will be discussed, which has proprietary rights to nuclear material and special rights in the storage of nuclear material on the basis of contractual boundary conditions.

On the basis of the facts and analyses compiled, and especially taking into account todays IAEA safeguards philosophy having most weight on material accountancy, the conclusion becomes apparent that the waste management strategy with a direct final repository is problematic from safeguards aspects since doubt is cast on the technical realization of a safeguards concept.

For certain types of fuel element where reprocessing is not envisaged and not worthwhile, Art. 35 VA can offer a possibility of a solution. In this case of the limited emplacement of spent fuel elements it could be possible to negotiate international safeguards according to this article.

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/3/ A. von Baeckmann, J. Powers, IAEA concerns about Advanced Containment and Surveillance Concepts or other Alternative Safeguards Concepts, INMM Conference, July 1981

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Letters to the Editor

Remarks on

"Comments on Inspection Goal Criteria for Material Accountancy"

To the Editor,

In his article "Comments on Inspection Goal Criteria for Material Accountancy", in the October 1986 ESARDA Bulletin, Mr. Canty argues that inspection goals used by the IAEA "imply unreasonably high verification efforts on the part of the inspectors, and, by implication, an unreasonable degree of intrusiveness into plant operation." In support of this contention, a statistical example is given for which it is shown that the inspector would have to sample 100% of items in the material balance to meet the "accountancy verification goal quantity," and the "current IAEA-internal goal of 95% detection probability for diversion of one goal quantity."

Readers not familiar with the subtleties of "variables-mode" sample-size requirements should be cautioned that the type of sample-size calculation performed in this article is not very meaningful in the region where this 100% result is produced, in the sense that almost any sample size between fairly small values and infinity can be obtained by making small changes in the input parameters. An example is given below. This is because once a relatively small number of samples per stratum have been taken, detection capability is not affected very much at all by the variables-mode sample size. For this reason, IAEA variables-mode sampling algorithms are more complex than the calculation presented in the article, and would not results in the high values quoted. Thus, the link between design objectives, variables-mode sample sizes, and inspection effort is not as simple or as strong as that portrayed in the article.

Canty's article quotes a 95% detection probability as the "IAEA-internal goal quantity." IAEA literature /1,2/ consistently uses the range of values 90-95% as that "normally" adopted or "adopted for planning." Adopting the mathematics given in the paper (and setting $\theta = \gamma = 1$, giving inspector and operator random and systematic errors the same magnitude) yields the following formula for "n", the sample size :

$$n = 2RN/(1+N+R(1-N))$$

where

$$R = ((1.645 + U_{1-\beta})/3.28)^2.$$

Results for n for various detection probabilities and values of N (the total number of items) are :

Desired detection probability (%)	Total number of items (N)		
	N = 25	N = 100	N = 250
95.25	31	415	*
95	25	100	250
94	15	26	31
93	11	16	17
90	6	7	8

* detection impossible with desired probability for any sample size.

Thus even in the example presented in the paper, although 100% sampling would be necessary to achieve a 95% detection probability, a detection probability of 90% (within the IAEA's stated goals) requires only modest sample sizes. Required variables-mode sample sizes will behave in a similarly radical manner as a function of other parameters, such as goal quantity or measurement accuracy. A very small improvement in the operator's or inspector's measurementy accuracies, for example, would dramatically reduce these sample sizes.

Therefore, while the article does show that the IAEA goal quantities and detection probabilities should not be chosen to match those of the operator precisely if the IAEA and operator have identical measurement capabilities, I would disagree with the contention that IAEA goals, as presently articulated, necessarily imply burdensome sampling requirements.

References

1. IAEA Safeguards Glossary, IAEA/SG/INF/1, p. 23
2. H. Grümm, "Designing IAEA Safeguards Approaches", 1980 INMM Proc., p. 18

Sincerely,

J.B. Sanborn

Sir,

My article on safeguards accountancy goal criteria was intended to demonstrate, on the basis of a simple, analytically tractable example, the consequences of the IAEA's insistence on "high assurance" for quantitative material accountancy verification. In his criticism, Mr. Sanborn correctly draws attention to the sensitivity of the variables sample size formula for the MUF-D test. He points out that if, for example, inspector and operator variances are exactly equal, then a required detection probability of 90% (as opposed to 95%) reduces the sample sizes in my example to more acceptable values.

The discussion is admittedly academic and it was not my intention to imply that the inspectorate would or could apply 100% variables sampling in reality. For inspection planning the IAEA might, and as I understand, actually does make the assumption that its verification accuracy will be as good as or better than that expected of the operator. Taken together with a detection probability goal of 90% for diversion of one AVG quantity this will lead to a moderate variables sample plan. The difficulty really arises in the **evaluation** of the accountancy data, i.e. the determination of actually achieved variances in order to compare the MUF-D statistic with its correct decision threshold. An increase in, say, the operator's σ_{MUF} over that assumed for planning purposes will lead to a decrease in detection capability below that which was both required and planned for. Here it should be noted that facility operators often do not even report their material balance uncertainties due to a lack of reliable error models. The Agency is then forced to estimate them on the basis of difference comparisons for a small number of verified batches. Worse still, some MUF strata are not measured or verified at all and are necessarily (but of course invalidly) excluded from MUF-D evaluation altogether.

To reformulate the point I was trying to make: even in a highly idealized situation, nowhere approached in existing large bulk handling plants, the AVG is virtually unattainable with 95% confidence. It is therefore counterproductive to set stringent quantified detection goals as criteria for effective safeguards in real safeguarded facilities. The very fact that they are calculable invites one to show that they are not being met, indeed cannot be met, under routine plant conditions and with existing resources. The effectiveness of technical safeguards measures relates to the way in which they build political confidence in the non-proliferation commitment of member states. This is far too subtle and complex as to be expressible in terms of kilograms and statistical probabilities.

Yours sincerely,
M. Canty

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