IAEA-TECDOC-786

Experience with spent fuel storage at research and test reactors

Proceedings of an Advisory Group meeting held in Vienna, 5–8 July 1993

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

January 1995

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The originating Section of this document in the IAEA was:

Nuclear Materials and Fuel Cycle Technology Section
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria

EXPERIENCE WITH SPENT FUEL STORAGE AT RESEARCH AND TEST REACTORS
IAEA, VIENNA, 1995
IAEA-TECDOC-786
ISSN 1011–4289

Printed by the IAEA in Austria
January 1995

FOREWORD

Irradiated fuel from research and test reactors has been stored at various facilities for several decades. As these facilities age and approach or exceed their original design lifetimes, there is mounting concern about closure of the fuel cycle and about the integrity of ageing fuels from the materials point of view as well as some concern about the loss of self-protection of the fuels as their activity decays. It is clear that an international effort is necessary to give these problems sufficient exposure and to ensure that work begins on appropriate solutions. The future of nuclear research, with its many benefits to mankind, is in jeopardy in some countries, especially countries without nuclear power programmes, because effective solutions for extended interim storage and final disposition of spent research reactor fuels are not yet available.

Some countries with fuel originally enriched in the USA have been faced with these problems only since take-back of foreign research reactor fuel of US origin was suspended in 1989. The other major supplier country, the former Soviet Union, never took back spent research reactor fuels from client States and at present Russia has no plans to take back the fuel in question. Consequently, many research reactor operators need to expand their irradiated fuel storage facilities right now, and to face the fact that they must find a solution for the final disposal of their fuels.

To obtain an overall picture of the size and extent of these problems, an Advisory Group Meeting on Storage Experience with Spent Fuel from Research Reactors was convened in Vienna, 5-8 July 1993, and attended by twelve participants and three observers representing thirteen different countries. These proceedings contain the country reports presented at the meeting.

The IAEA wishes to thank all of the participants in the meeting for their contributions to this document, which summarizes the experience with spent fuel management at research reactors in nine different countries. The IAEA officer responsible for the organization of the meeting and for the compilation of this document was I.G. Ritchie of the Nuclear Materials and Fuel Cycle Technology Section.

EDITORIAL NOTE

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SUMMARY OF THE ADVISORY GROUP MEETING

This Advisory Group meeting (AGM) was held from 5 to 8 July 1993 at the IAEA's headquarters in Vienna and was chaired by Dr. W. Krull of GKSS in Germany. Twelve nominated participants and three observers from thirteen different countries took part in the sessions and discussions of three working groups set up during the AGM. Thirteen 'country' reports were presented, eleven of which are summarized in this document.

OBJECTIVES OF THE MEETING

The objectives of the AGM were as follows:

- To evaluate the overall picture of irradiated fuel storage from research and test reactors worldwide.

- To address the problems faced by research reactor operators who can no longer return spent fuel to its country of origin.

- To assess the widespread concerns about the integrity of ageing fuels in ageing spent fuel storage facilities.

The first two days of the meeting were essentially devoted to information exchange and the presentation of experience with spent fuel storage at research reactors in the various countries represented at the AGM. In particular, each participant focused a part of his or her presentation existing problems and concerns.

It became clear from presentations on experience in different countries that the AGM was very timely. Moreover, the participants believed that the IAEA has a very important role to play in promoting information exchanges on spent fuel storage from research and test reactors and in advising individual Member States with specific problems in this area.

Following the information exchange, the participants split up into three working groups:

- Group 1 reviewed all of the presentations and identified and enumerated common problems and concerns.
- Group 2 reviewed the current activities of the IAEA in this area and listed recommendations for future IAEA activities based upon the deliberations of the AGM.
- Group 3 reviewed the Questionnaire on irradiated fuel circulated by the IAEA to research reactor operators in Member States excluding Canada, France, Russia, United Kingdom and the United States of America and the preliminary responses to the Questionnaire. This group also produced a revised version of the Questionnaire to be sent to those countries excluded from the first circulation.

STATUS

During the presentations on experience with spent fuel management in the countries represented at the AGM, the different steps, problems and concerns in the back-end of the fuel cycle were discussed with emphasis on the status of the back-end policy in the country in question and the current problems of the reactor operators. Many different views were expressed but, on the whole, the situation is reflected in sufficient detail to see *where we are* and *where we should be going*. It was pointed out by several participants that an enormous amount of positive experience of spent fuel storage has been accumulated at research reactors worldwide, but that more quantitative information on such topics as cladding corrosion rates is needed to predict fuel behaviour in extended storage. In addition, other problems, whose solutions will need international co-operation and information exchange, will have to be addressed in the future.

From the start-up of a research reactor there are spent fuel storage positions available for different purposes, e.g. reshuffling, repair, accumulating spent fuel for shipment or, in some cases, for all the spent fuel expected to be discharged during the initial design lifetime of the reactor. Many research reactors, especially the low power reactors, but also some high power reactors, have stored all of their fuel elements from the initial start-up until the present day, a period of 30 years or more in some cases. In the past, research reactors with higher power ratings routinely shipped their spent fuel to reprocessing facilities in the USA (Savannah River, Idaho Falls), France (Marcoule), UK (Dounreay) or Belgium (Eurochemic). Similarly, research reactor operators in the former Soviet Union shipped their spent fuel to reprocessing facilities in Russia (Majak), although other research reactor operators with Russian origin fuel never had that option. Because some of these reprocessing facilities have been operating for only a short time, and others have stopped accepting spent fuel (US reprocessing facilities have stopped accepting US origin HEU since the end of 1988 and US origin LEU since the end of 1992), there is an increasing need to expand existing storage facilities. This need was common theme in all of the presentations. Expansion of spent fuel storage facilities involves many different issues:

- Design and construction of compact racks using neutron absorbers or geometrical separation of fuel elements to satisfy calculated or measured criticality criteria.
- Design and construction of single tier or multi-tier (up to three) racks taking into account the needs of safeguards inspectors, local seismic conditions and ease of general inspection.
- Compliance with licensing procedures (sometimes extremely restrictive).
- Detection, location and inspection of failed fuel elements in storage.
- Literature research on materials problems, especially for new or substitute materials to be used in storage facilities.
- Increasing demands for longer term control of corrosive environments and the integrity of the fuel cladding.
- Consideration of the loss of self-protection of spent fuel, e.g. when the radiation dose in air at 1 m distance falls below the allowable limit (1 Gy/h) the fuel must be considered as fresh.
- Choice of storage in pools, hot cells, horizontal channels, dry wells, vaults, or transfer and transportation containers, etc.
- Dry storage in air (with natural or forced convective cooling) or an inert atmosphere.
- Auxiliary storage at-reactor or away-from-reactor.

In many cases an expansion of storage facilities at the site in question is possible. But in other cases not enough space is available so that there is the need to build a separate, auxiliary storage facility, which may be required to operate as an *interim* store for up to 40 or 50 years. Careful investigations are necessary before embarking on such an expansion programme and are already being made at several facilities. One presentation noted that a license has already been granted for a fifty year period of dry storage. In other cases, studies are underway to find a location for, and to define the criteria to be satisfied in, the auxiliary interim storage. These are points that must be looked at carefully and, in general, the same questions have to be answered for either an auxiliary interim storage facility or for an expansion of the existing storage facilities at the site in question. The main differences are usually the anticipated storage times.

But in any case, if the spent fuel elements are to be shipped away from the research reactor to an interim storage site or a reprocessing facility, licensed transport casks must be available. However, because of the increasingly stringent international and national specifications for nuclear fuel transport containers, in several cases the licenses for existing casks have expired. Since many new calculations and hardware changes must be made to re-license them or procure licenses for new designs, long delays are expected before licensed casks will be available in many countries. Consequently, some reactor operators have absolutely no possibility at present of shipping fuel away from the at-reactor storage facility. These problems must be solved and new casks must be built and licensed.

It has become clear that storage expansion and/or auxiliary interim storage construction is an important step to keep many research reactors in operation. Nevertheless, this is not the final solution. The only final options are reprocessing of the spent fuel or its direct disposal. In the case of the reprocessing of the spent fuel, the reactor operator and/or the reprocessor have to answer two important questions (assuming that the problems associated with the transportation have been resolved).

- What is to be done with the reprocessed fissile material (uranium and in some cases plutonium)? Can it be used for the fabrication of new research reactor fuel elements or sold for other peaceful uses, and will the operator get a credit for it?
- What is to be done with the waste from the reprocessing? Will the waste be kept by the reprocessor or will it be returned to the operator? What are the specifications for the transportation cask for the waste from reprocessing? Is the operator in fact able to take back the waste or are there national restrictions, limitations and laws which make it impossible?

In the case of direct disposal, there is a need to have both a repository and suitable containers for the final disposal. Countries are aware of this problem and intensive work is ongoing on the subject, but considerable delays are expected before direct disposal of spent nuclear fuel is available anywhere.

At every step in the back-end of the fuel cycle major or minor actions are required from the responsible regulatory authorities. In some cases they act directly and are well informed of the present crisis faced by many reactor operators who can no longer ship spent fuel back to the country of origin. Nevertheless, the responses of some regulatory authorities have threatened the continued operation of research reactors in the country in question, e.g.:

- operators have no permission to insert fresh fuel into the core;
- no new fuel can be purchased;
- operators must demonstrate that for a six year rolling period there are enough storage positions and/or a reprocessing contract and/or a contract for interim storage to absorb the fuel discharged from the reactor.

These issues are of grave concern for the reactor operators and are forcing them to look for solutions quickly. It is believed that the only acceptable solution for many at present is to return the spent fuel to the country where it was originally enriched. If supplier countries took back the spent fuel and retained any waste from reprocessing, if only for a reasonable time period of say five years, it would give a period of grace that would allow those countries that wanted to continue to operate their research reactors time to produce a *home grown* solution. Collecting together spent research reactor fuels in the few supplier countries would also help to reduce the proliferation risk.

Wet storage of spent fuel in at-reactor pools is the most commonly used storage option in Member States. However, for long term storage of aluminium clad MTR fuels for periods exceeding 30 to 40 years, dry storage is considered to be preferable because of the known instability of aluminium in water. Dry storage facilities have been built and used successfully in some cases, although long term experience with this method of storage is rather limited.

For wet storage, a great deal of experience indicates that corrosion of aluminium fuel cladding can be minimized to negligible levels by appropriate control of water chemistry and temperature, especially after passivation of the aluminium surface in contact with the water. For MTR fuel with aluminium cladding, the generally accepted water chemistry parameters are a pH in the range of 5.5 to 6.5, specific conductivity of less than 1.5 μS/cm and low concentrations of chloride, sulphate and copper ions. On-line purification of pool water, with re-circulation through filters and ion-exchangers is the norm. However, submerged, cartridge type, ion-exchange units of a non-regenerative type show considerable promise by avoiding the production of liquid waste and facilitating handling and disposal operations. They also help to reduce the radiation exposure of personnel and do not produce a loss of pool water inventory in the case of pipe failure.

The integrity of storage pools has been very good in general. Stainless steel pool liners are recommended, but one facility has had excellent results with an aluminium liner, perhaps because of the presence of pure lead sheet shielding in the same pool. Other materials such as epoxy coatings on concrete have been used, but these have a tendency to blister and peel eventually, necessitating refurbishment of the reactor pool walls. Storage pools located above the ground level are considered to be better than those below ground, especially in areas prone to flooding. Difficulties in maintenance of underwater fuel transfer and handling equipment was mentioned by some participants. Such difficulties seem to be part of the general problem of ageing equipment in ageing facilities and are likely to increase in both severity and frequency as time goes on. It was suggested that manufacturers should take into account maintainability in the initial design phase of such equipment. Failure of fuel elements, resting in a vertical position on the bottom of a pool, by buckling under their own weight was mentioned by one participant. Storage by suspension from the top solved this particular problem, although the creep of cladding in all types of storage facilities remains a concern.

Experience with dry storage of spent fuel from research reactors includes dry-wells, vaults, hot-cells, horizontal concrete channels, vertical concrete canisters and various casks. Forced or natural air circulation has been used in some cases whereas inert gas atmospheres have been used in others. Monitoring of humidity, temperature, pressure and activity has been employed for surveillance of dry stores, but some doubt was voiced that clad failures in long term dry storage could be detected. Some participants reported the adaptation of at-reactor facilities such as hot-cells and underground rooms to expand fuel storage capacity.

Shielded casks are used for the transfer and shipment of spent fuel. One participant reported the appearance of surface activity on such a cask due to *sweating* of activity caused by excessive weathering of the cask during shipment. Unfortunately, many of the older casks available to research reactor operators in the past do not conform to present day safety requirements and are no longer licensed for shipment or storage of spent fuel.

In summary, lack of adequate storage capacity for spent fuel is threatening to curtail and even terminate the operation of several research reactors. Most participants agreed that as of now, the only acceptable solution is to send the spent fuel back to the country of origin where it will reprocessed or permanently stored.

CONCLUSIONS

1. In response to the general trend to store research reactor fuels for longer and longer times in wet *interim* storage, there is a need to identify from the recorded experience to date, the critical parameters controlling corrosion and other forms of material ageing leading to the degradation of mechanical and physical properties. There is a large amount of information at the various research reactor sites that has never been analysed, correlated and summarized. This needs to be gathered together and conclusions drawn from it that should culminate in a *code of good practice* for the safe and reliable storage of research reactor fuels in water.

2. It is clear that those involved in the storage and management of research reactor fuels have a useful source information available from the parallel but more advanced development of the storage of fuels from nuclear power plants. Nevertheless, it is concluded that the specific problems of research reactor fuels should be identified and evaluated, especially those related to materials, operational procedures in storage facilities and criticality evaluations associated with rack design and expansion, in an attempt to define and prioritize areas where information exchange is needed or research efforts need to be initiated or strengthened.

3. Special attention should be paid to the conditioning and packaging of failed fuel. There is widespread concern that at some time in the future there will be a spate of failures of aluminium clad fuels all occurring at roughly the same time and corresponding to about the same long time in wet storage. In such a case, there would be a need for international collaboration that would somehow ensure the availability of an underwater canning capability for failed MTR fuel. In addition, the participants saw that there was an urgent need to develop *an exhaustive set of standard safety criteria* applicable to the development, construction and operation of interim storage facilities for research reactor fuels.

SPENT FUEL STORAGE AT THE ASTRA REACTOR, SEIBERSDORF

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Abstract

The storage of spent fuel at the ASTRA research reactor, Seibersdorf, Austria, is described as well as some of the problems that have been encountered. Of special interest is the description of the adaptation of a hot cell for the interim storage of spent fuel.

1. General

Since 1960 the Austrian Research Centre at Seibersdorf operates the 10 MW pool type research reactor ASTRA.

The reactor has been one of the first facilities in Europe participating in the US Program for conversion to fuel of low enrichment. In fact, the ASTRA core has been chosen as a reference core in this program (1980-1989).

Since that time the facility is in possession of a variety of used MTR type fuel elements, some of them special test elements (Table 1).

2. The storage problem

When US reprocessing facilities stopped accepting spent fuel, the storage capacity for irradiated fuel elements had to be increased by adding a "storage box" (in addition to the existing storage racks) in the pool.

However, the available space for storage in the pool was drastically reduced when it was decided to install equipment for silicon doping in the reactor. This necessitated the removal of all south side beam tubes and - even more important - it used up much of the precious storage space.

The situation is illustrated in Figs. 1 and 2. It shows the pool area (1), the shielding concrete (2) and the core (3), beam tubes (7), thermal column (11) and silicon irradiation (12). It also shows a "wet" hot-cell (8) which can be connected to the pool via

TABLE 1. ASTRA FUEL ELEMENTS FOR INTERIM STORAGE

	U-Al_x metal	U_3O_8-Al oxide	U_xSi_y-Al silicide
HEU	38	—	—
MEU	5	—	—
LEU	—	3	26

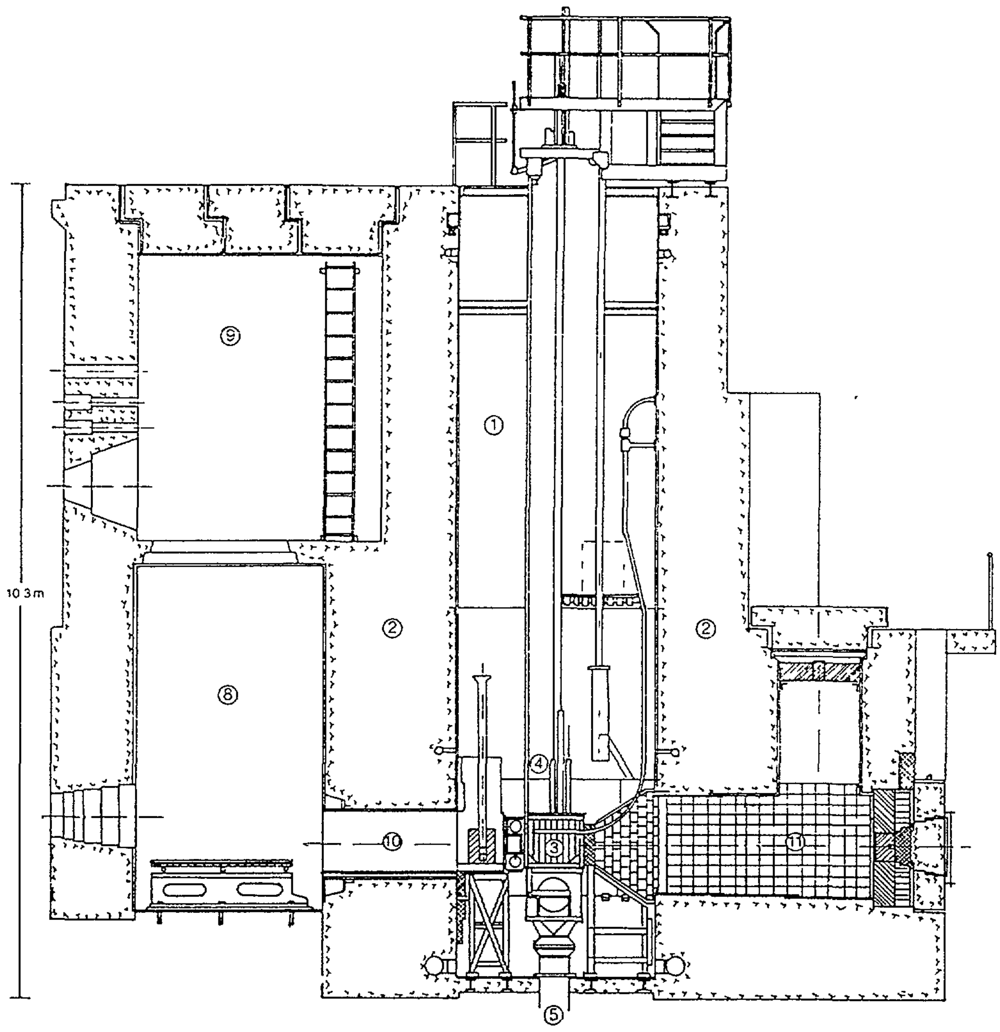

FIG. 1. Vertical section of the reactor block.

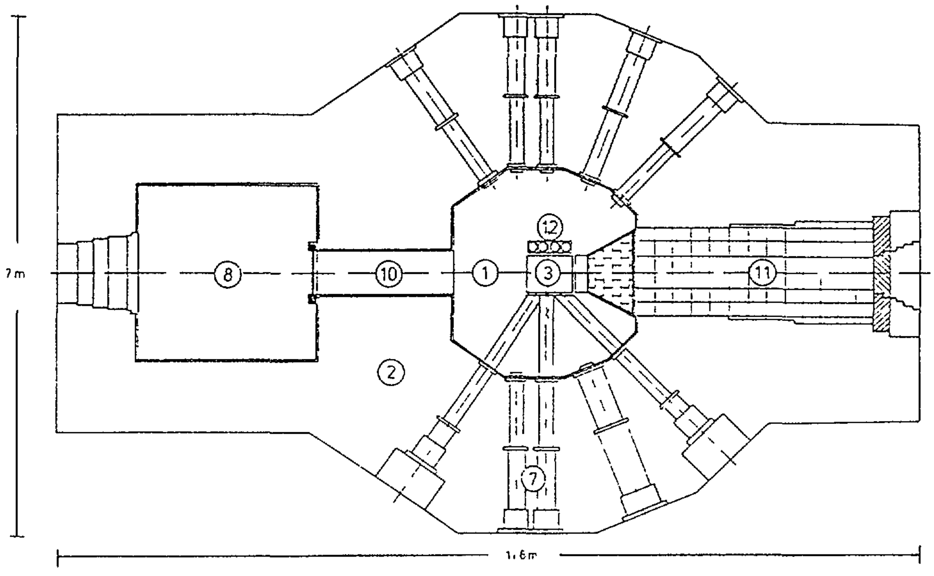

FIG. 2. Horizontal section of the reactor block.

an access channel (10). Fig. 1, the vertical section, shows that this "lower hot-cell" (8) is also connected to a dry "upper hot-cell" (9). From there it is possible to manipulate fuel elements in (8) by using standard fuel handling tools.

These hot-cells (originally intended for handling irradiation experiments) had already been used previously as an interim storage for fuel during maintenance work.

3. Adapting the Hot-Cell for fuel storage

The given situation suggested on adaptation of the Lower Hot-Cell for (interim) fuel storage.

For this purpose 4 aluminium storage boxes, each having 7 x 7 storage positions have been constructed and installed (Fig. 3).

Subcriticality is ensured by geometry: The boxes have every second position mechanically blocked, so it is only possible to load a "chess-board pattern".

This arrangement has been proved experimentally to be subcritical by simulating the pattern on the core grid-plate.

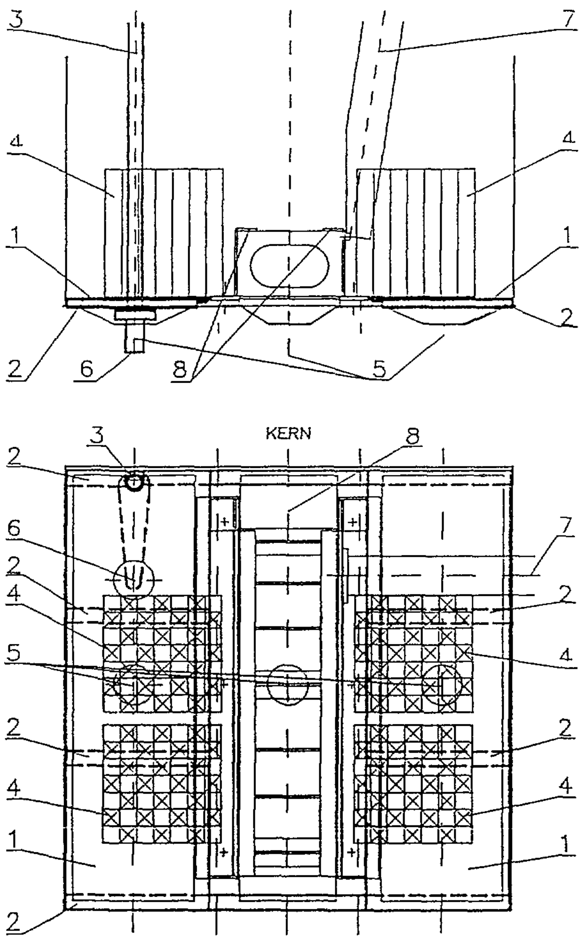

FIG. 3

Safe geometry is considered preferable to absorbers because of possible corrosion problems.

As a next step the Hot-Cell - which had an independent water system - was connected to the beam tube system: Pool water, which "scavenges" the gap between beam tubes and concrete shield, is now fed to the bottom of the Hot-Cell, then passes upwards through the storage boxes and returns via an overflow pipe to the primary cipcuit. The overflow pipe automatically keeps a constant **water level**, however, an additional electric level alarm has been installed.

Water quality is of course identical to the reactor primary coolant.

Fig. 4 shows an axonometric view of the arrangement.

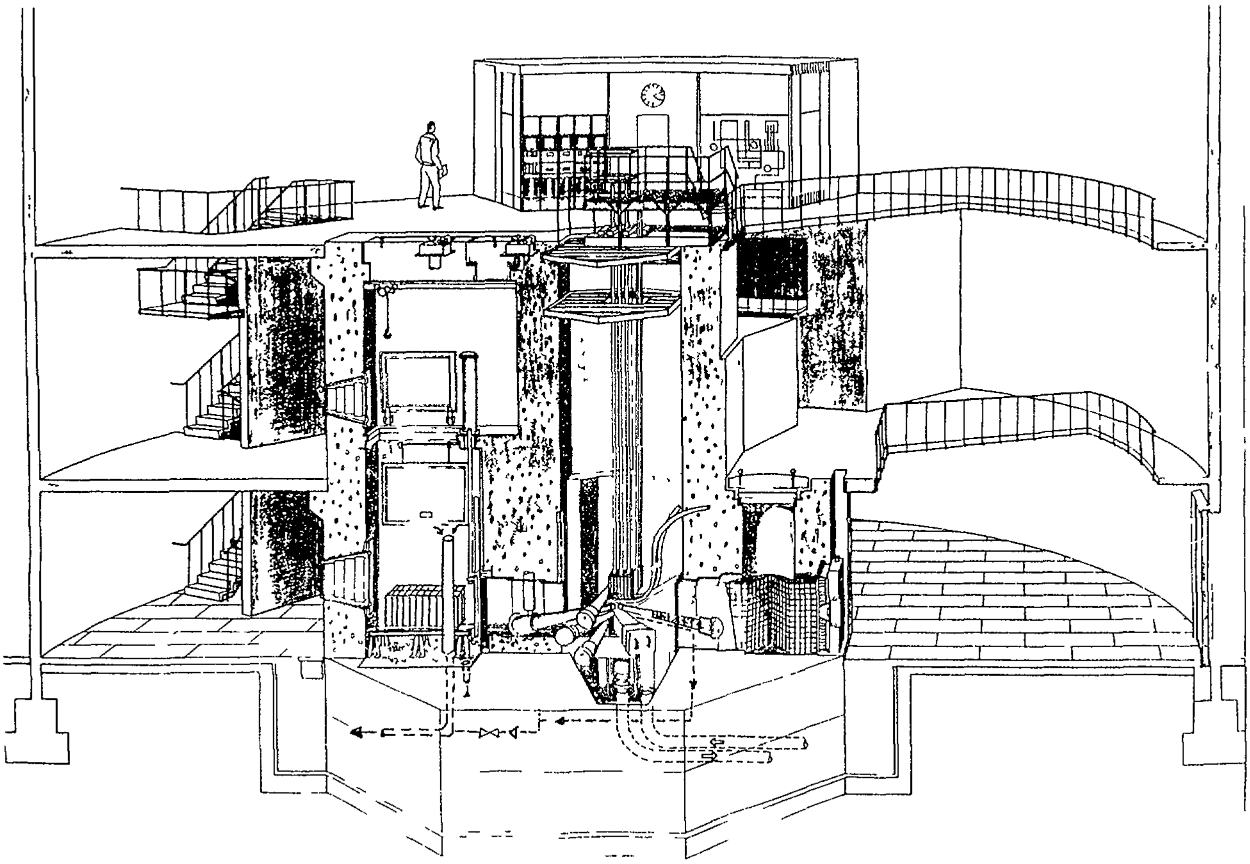

FIG. 4.

The **capacity** of the boxes in large enough to accept all fuel elements presently in use at the ASTRA Reactor. This is necessary for the case of unforeseen work in the pool, when all fuel would have to be transferred to the storage area.

Obtaining the permission from safeguards and licensing authorities was no problem, since the Lower Hot-Cell is inside the secured reactor area and it has been previously used and licensed as a "temporary fuel storage".

Nevertheless this solution is a compromise and can not be a long term solution the "back end" of the fuel cycle.

STORAGE EXPERIENCE WITH FUEL FROM RESEARCH REACTORS IN BELGIUM

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Abstract

Experiences and problems with the storage of irradiated fuel at research reactors in Belgium are described. In particular, the interim storage of spent fuel elements at the BR2 reactor in Mol, which has posed special problems, is described in some detail in the Annex to this paper.

1. INTRODUCTION

SCK/CEN owns 5 research reactors, some of them still in activity, others now decommissioned.

BR1 is a 3.5 MWth graphite moderated and air cooled reactor. It went critical in 1956 for the first time. It is now only used at low power for research, activation analysis, dosimetry, etc. on a discontinuous basis.

BR2 is a powerful (120 MWth) Material Testing Reactor. It went into operation early in 1963 and has been intensely used for research projects on the behaviour of structural materials and nuclear fuels. It will be operated on a continuous basis up to mid-1995 or mid-1996. A decision for continued operation must be taken by end of 1993 based on technical and economical studies.

BR3 is a small (40.9 MWth) PWR reactor. It was operated from 1962 until 1987 and used primarily for testing of advanced fuel. The plant was finally shut down on 30 June 1987 and is now being decommissioned.

BR02 is a nuclear mock-up of the BR2 reactor. It was mainly used to study the nuclear characteristics of the various BR2 core configurations and to determine the precise irradiation parameters for complex experiments. BR02 is now being decommissioned.

VENUS is a critical assembly which came into operation in 1963. It is used to perform reactor physics experiments in support of various LWR research progammes.

2. EXPERIENCE AND PROBLEMS WITH THE STORAGE OF IRRADIATED FUEL

2.1. BR1 reactor

BR1 is fuelled with natural metallic uranium with Al cladding. A dry storage is also available. It is made of concrete with horizontal channels. The dry storage can contain about 20% of the reactor loading. There is at present no plan to dispose of the irradiated fuel (1037 fuel elements are already in storage).

2.2. BR2 reactor

The BR2 reactor is fuelled with 93% HEU. Fuel elements are made of a core of Al-U cermet sandwiched between aluminium plates. Storage is available in the containment building (about 200 places) and in the storage channel outside (800 places available in January 1993).

By January 1993, the storage capacity was nearly exhausted. The storage capacity is now being expanded to allow continued operation to mid-1996 at the latest and to allow an inspection and maintenance of all of the compartments of the storage channel. Storage expansion is accomodated by reracking, i.e. introduction of new high density racks and modifications to old storage racks.

A description of the present underwater storage facilities and the plans for storage expansion are described in the Annex of this paper.

One hundred and fourty-four fuel elements will be transferred to AEA Dounreay for reprocessing late in 1993. This transfer is needed to allow the refurbishment of the storage channel.

Alternate solutions to underwater storage should be available for operation after mid-1996. An option to refurbish the reactor is now under study.

Up to now, different possible solutions have been examined:

- Reprocessing at the Dounreay plant.
- Dry interim storage in thick containers, to be stored in a building on the Belgoprocess site (Mol).
- Dry (interim) storage in thin containers to be stored in a shielded building - foreseen for storage of vitrified waste from Belgian power plants - on the Belgoprocess site.

Technical and financial aspects of these solutions have been examined and compared with available results from Germany.

The possibility of returning the fuel back to the USA is now again open and is being followed by a Research Reactor Group under leadership of the Edlow International Co.

2.3. BR02

BRO2 fuel is non-active. Most of the fuel elements can also be used in the BR2 reactor. Consequently, there are no specific problems of fuel storage.

2.4. BR3

Two hundred fuel assemblies containing ± 2000 fuel pins are stored underwater at the BR3 site since the final shutdown of the plant.

The fuel pins are clad with Zircaloy and contain U and Pu oxydes with initial enrichments up to 10%.

Also for BR3, different alternate solutions were examined.

- Reprocessing by COGEMA or AEA.
- Dry storage in thick containers.
- Dry storage in thin containers.

As for BR2 fuel, the dry storage option was considered on the Belgoprocess site. A prototype dry storage cask (Mini-Castor) has been built and the different options are being technically and economically compared.

2.5. Venus

The fuel is made of BR3 type pins with various enrichments and compositions. No particular storage problems exist due to the fact that the fuel is not active (no power operation).

3. CONCLUSIONS

Storage problems exist for two research reactors in Belgium (CEN/SCK). The storage problems at the BR2 are very acute and constitute a threat for further continued operation. An alternate solution to the on-site underwater storage *has* to be found within the next years. The problems experienced for the BR3 reactor are quite different: the plant is being decommissioned and the fuel can be stored safely underwater for a long period (Zircaloy cladding). Long term problems (buildings, infrastructure, etc.) can however be encountered. The fuel is non-'standard' PWR fuel and cannot be treated in existing commercial LWR reprocessing plants.

Annex:
INTERIM STORAGE OF SPENT FUEL ELEMENTS AT THE BR2 REACTOR

1. INTRODUCTION

The BR2 reactor is owned and operated by the Belgian Nuclear Research Center (SCK/CEN) in Mol (Belgium). The reactor went critical for the first time on the 29th June 1961. It was put into service with an experimental loading in January 1963. On the 31st December 1978, the reactor was shut down to replace the beryllium matrix. Routine operation was resumed in July 1980.

The operation of BR2 is foreseen up to mid-1995 or mid-1996. A decision to operate further is now under evaluation. A decision is to be taken by the end of 1993 on the basis of a technical and economical study for the BR2 refurbishment. A refurbishment - together with a second replacement of the Beryllium matrix - is indeed needed after ~35 years operations to replace equipment and to comply with modern safety standards.

Historically, BR2 has always closed its fuel cycle by reprocessing the HEU fuel by EUROCHEMIC, COGEMA (Marcoule) or the U.S. D.O.E. (Savannah River or Idaho Falls). The last reprocessing campaign took place in 1981. By January 1989, preparations were going on for 4 new shipments (in total 144 Fuel Elements with the Pegase container) to Savannah River, when the decision was taken by D.O.E. to halt the return to the U.S. of U.S. origin irradiated fuel.

Since then, the continued operation of the BR2 reactor required early in 1991 a limited expansion (2 × 56 fuel elements) of the storage capacity.

By the end of 1991, however, it became clear that a large expansion of the storage capacity had to be foreseen to permit a continuation of BR2 operation until 31.06.1996 at the latest. Also, by this time a renewal by the DOE of its off-Site fuels policy appeared very improbable for the short term (2 to 3 years) with the possibility that it might be terminated permanently in the future. Consequently, alternative solutions such as reprocessing in Europe (by AEA Technology UK and Cogema France), by storage in thin or thick containers had to be considered as the hope for a US solution disappeared and with it the ability to close the fuel cycle at a low price.

In this context, the option was taken by November 1991 to expand the interim water storage at the BR2 site to allow continued reactor operation until its foreseen shutdown (for refurbishment or decommissioning): time was indeed needed to evaluate alternate solutions offered by foreign countries, or to develop a typical solution for Belgium. The project was started with the goal to have a new licensed and installed storage capacity by the end of 1992.

2. UNDERWATER STORAGE FACILITIES AT BR2

2.1. Storage pools

In the containment building, storage is accomodated in one of the two small (3.6 m × 2.4 m and 8 m deep) pools adjacent to the reactor pool. Fuel elements in this storage constitute the "live" stock for the management of the reactor core loadings. When they reach their final burnup, they cool down for 100 days and are then transported outside the containment building to the storage channel. The storage capacity in the containment building amounts to 190 standard fuel elements (Figs 1 and 2).

The storage channel, outside the containment building is situated in the process building (Figs 3 and 4). It is constructed along a north-south axis. It is 3.1 m wide and 7.7 m deep in all compartments but one, where it is 17.5 m deep. The storage channel is divided in to seven compartments (numbered 091 to 097) which can be separately isolated.

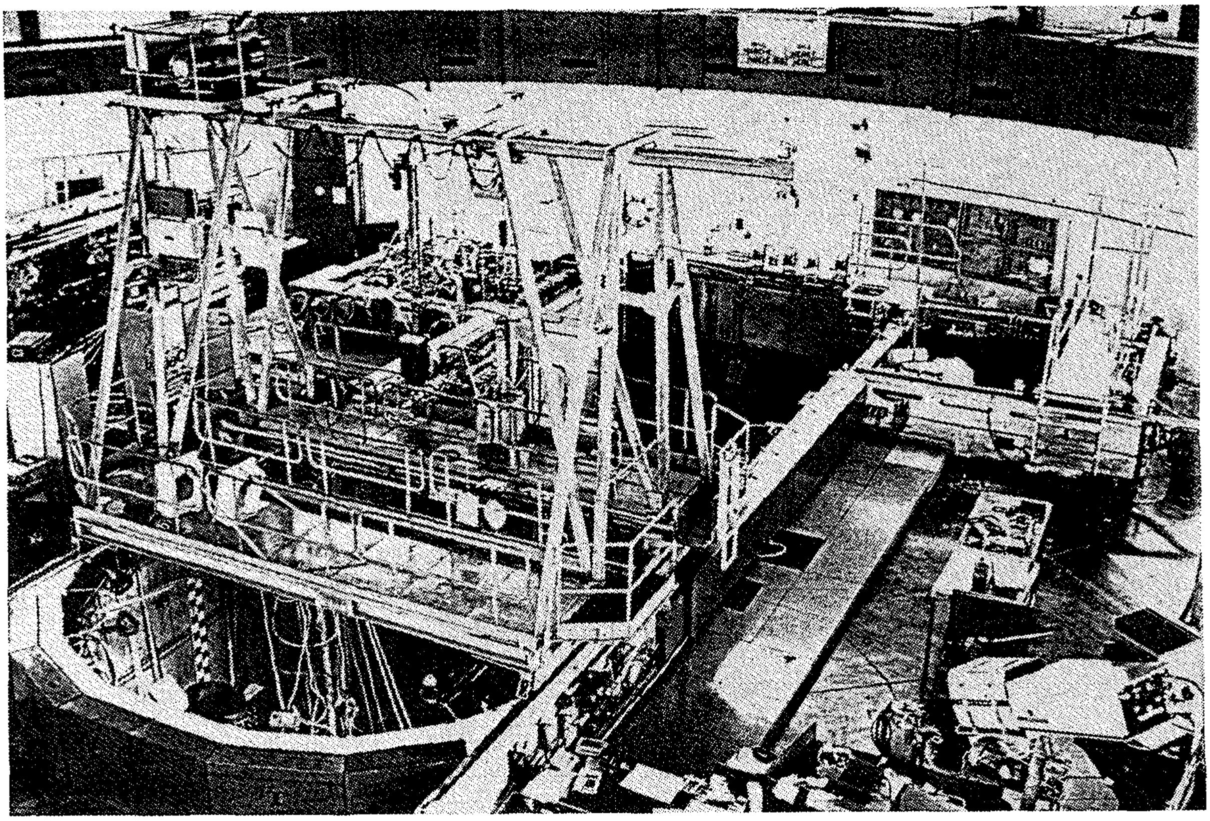

FIG. 1. Reactor pool – Main pool.

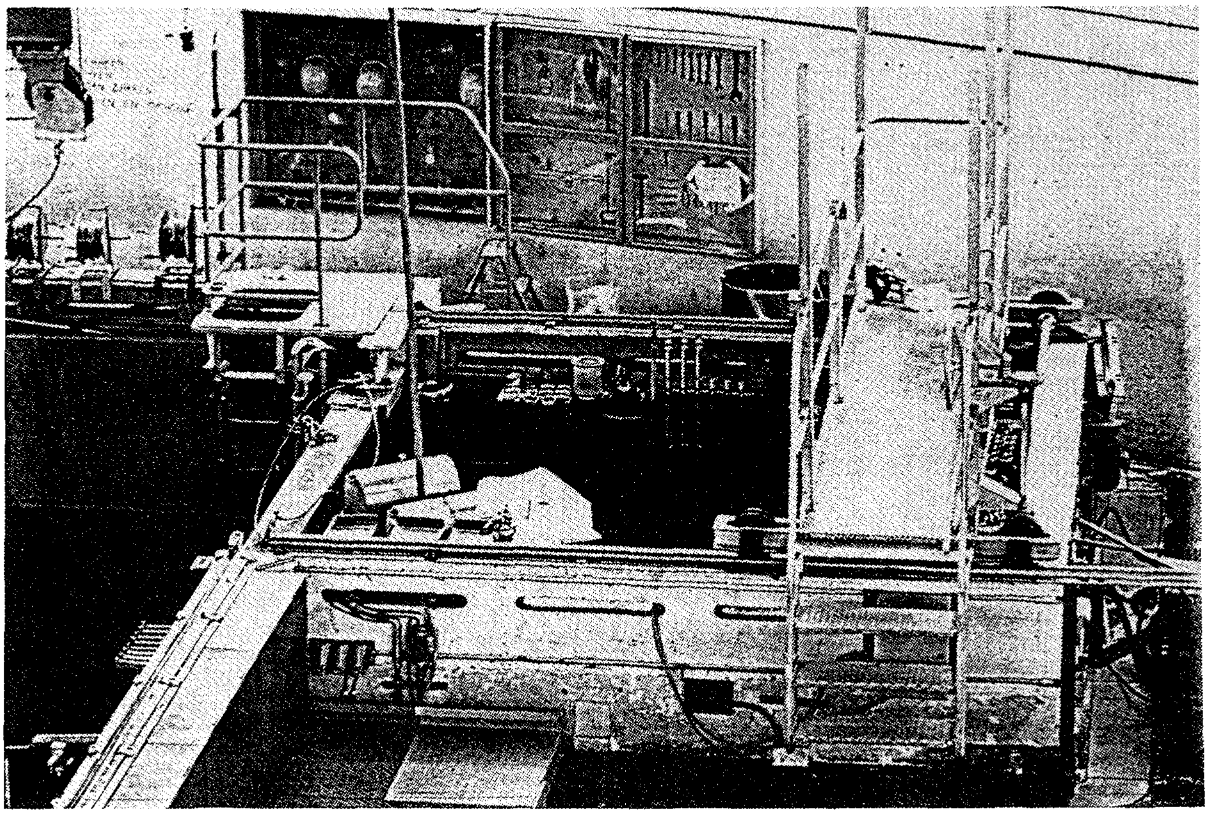

FIG. 2. CMF – Auxiliary pool.

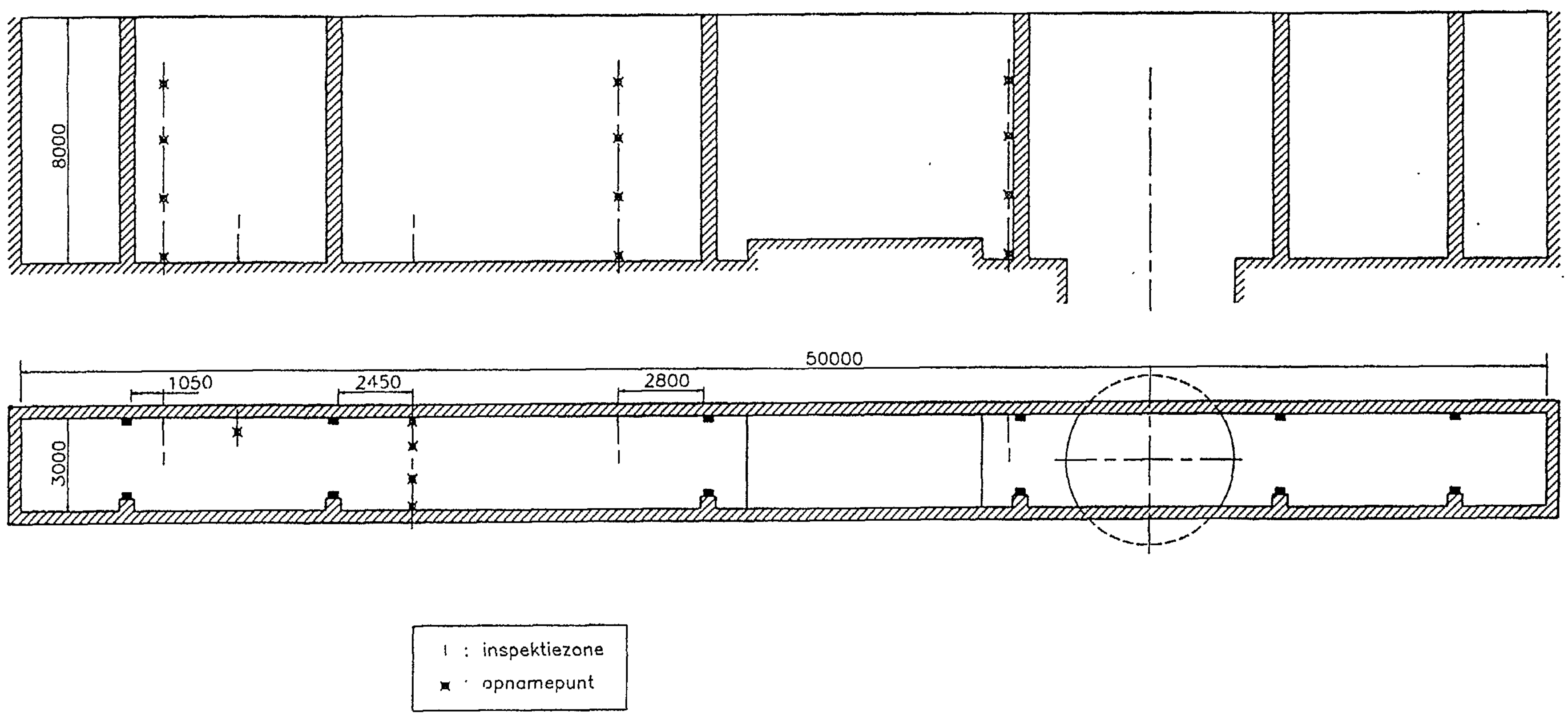

FIG. 3. Storage channel.

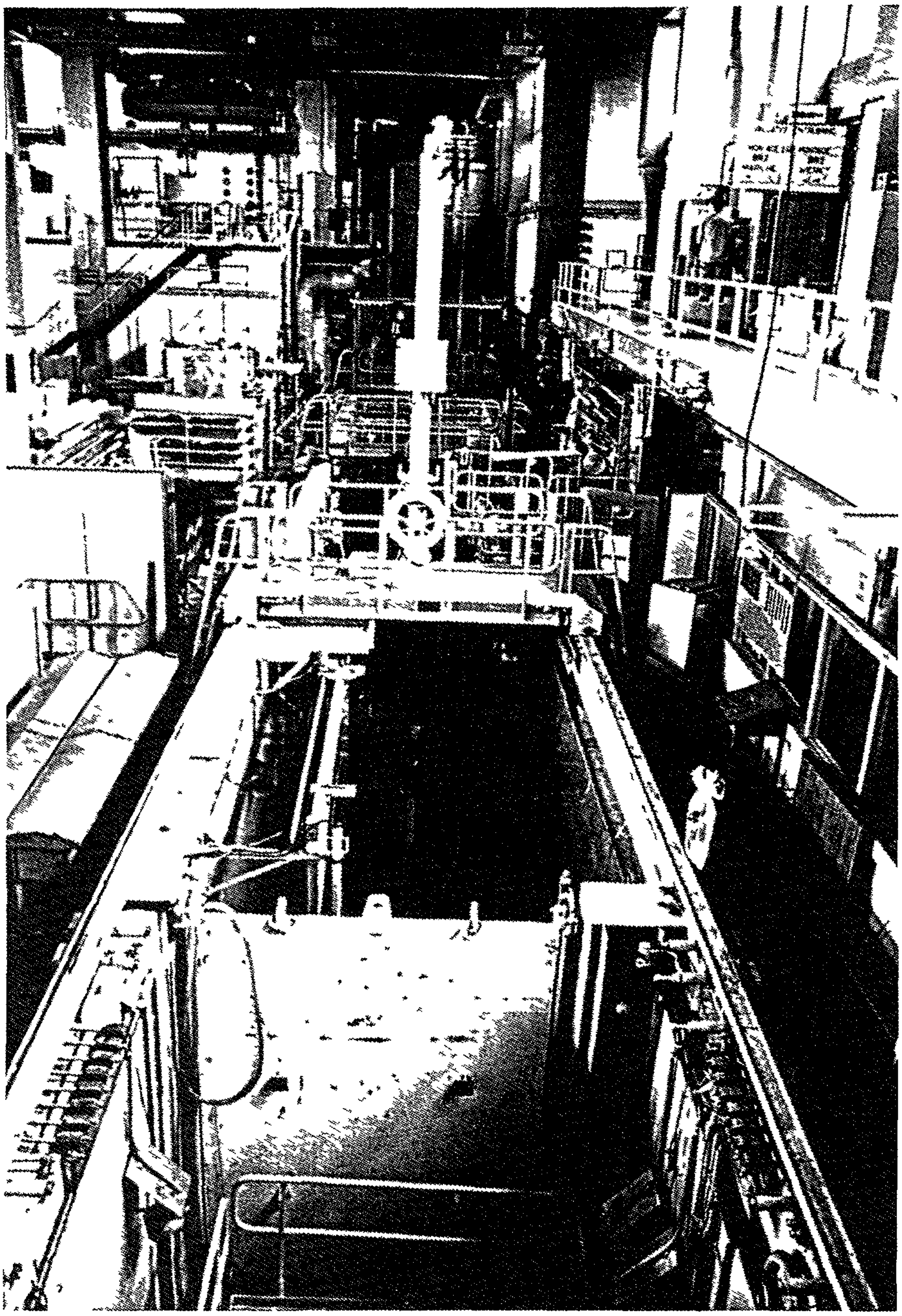

FIG 4 Storage channel

The walls of the storage channel are made of 120 or 170 cm thick barytic concrete and covered by a painted cladding made of an ordinary steel, 8 mm thick.

The storage channel is used for many purposes.

- Compartment 091. contains the BR02 reactor, a zero power mockup facility of the BR2 reactor
- Compartment 092: storage for BR2 fuel and Cobalt calibration equipment.
- Compartment 093: storage of BR2 fuel, gamma irradiation facility (RITA) and hydraulic test facility for control rods.
- Compartment 094. storage of experimental fuel under various forms and storage of experiments after irradiation.

- Compartment 095: transfer between the containment building, storage channel and hot cell.
- Compartment 096: loading/unloading space for containers and storage of cut BR2 fuel.
- Compartment 097: hydraulic rabbit.

2.2. BR2 standard driver fuel elements

The BR2 fuel element has a 30 in. (762 mm) fuel length which is a common characteristic of all fuel elements and is derived from the height of the beryllium matrix. The diameters of the available bores in the Beryllium matrix determine the natural outer diameters of all fuel elements, either 84 or 200 mm (Figs 5 and 6).

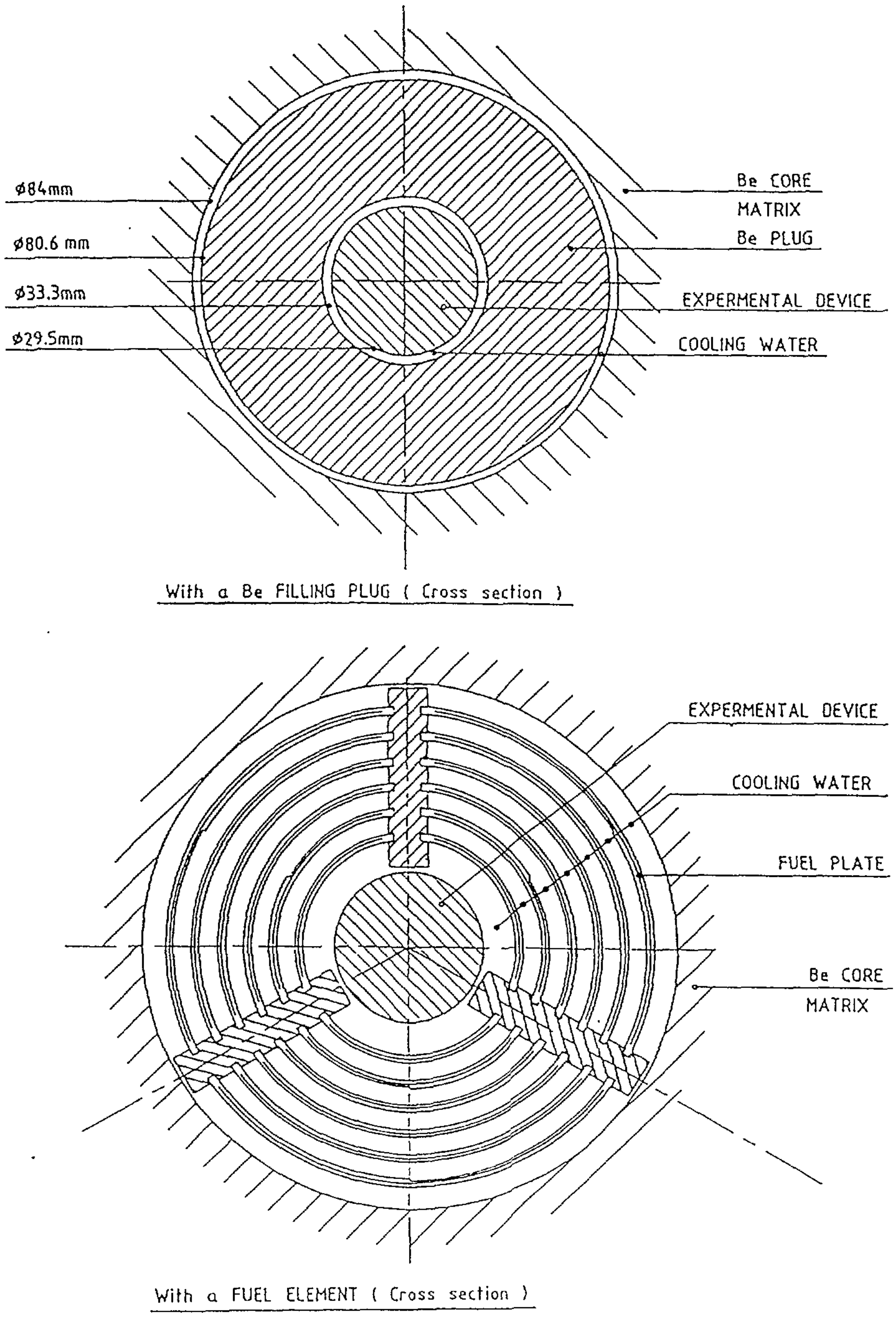

FIG. 5. Beryllium channel, 84 mm.

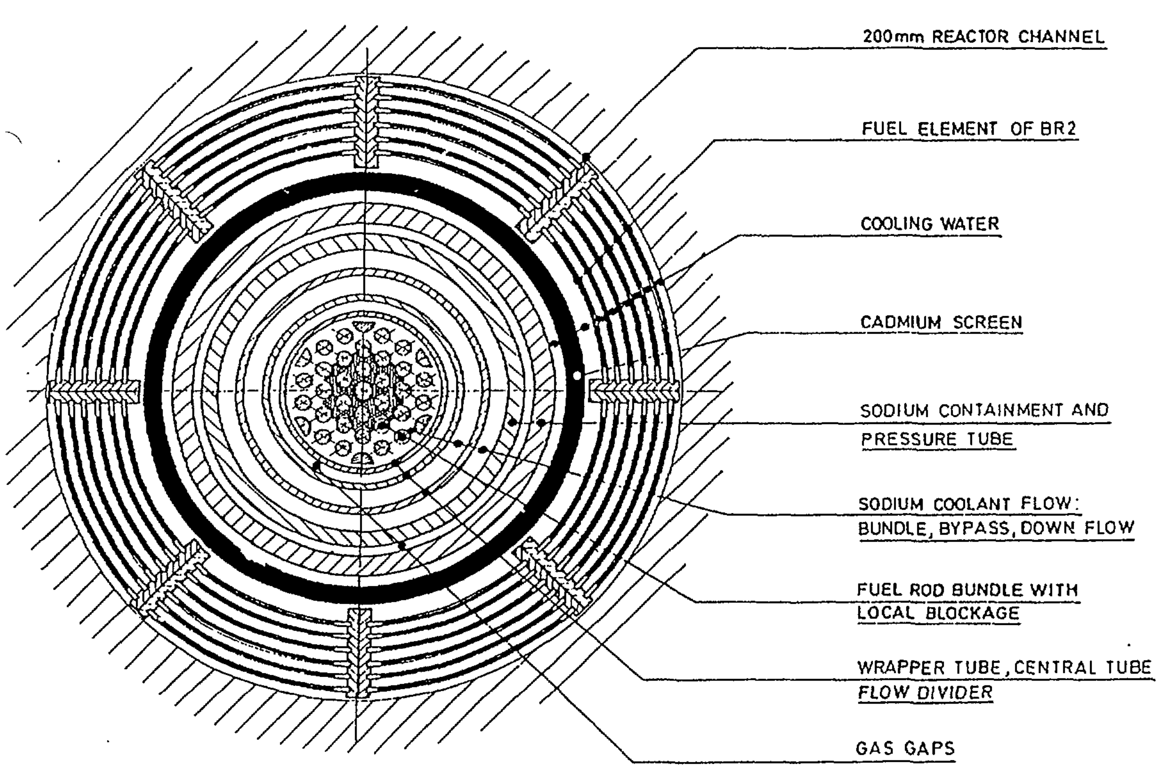

FIG. 6. Beryllium channel, 200 mm.

The standard fuel element is characterized by the following main geometrical parameters:

- Maximum number of concentric fuel tubes:- 6.
- Wall thickness of each fuel tube: 1.27 mm comprising a 0.508 mm core.
- Cladding layers of 0.35 mm each.
- A 3 mm water gap thickness between tube walls.

A cross-section of a standard fuel element shows each cylindrical tube to be an assembly of 3 equal 120 segments made of rolled plates. These fuel plate segments are mechanically fixed into three grooved solid radial webs (Figs 7 and 8).

The inner fuel tube of each fuel element encloses a free, open ended volume available for irradiation purposes and one or more fuel tubes can be omitted, beginning with the inner fuel tube, to modify the inner volume.

All fuel elements are provided with end fittings at both ends. The end fittings carry the rollers by which the fuel elements are guided during the loading and unloading operations, and centered in the hole of the matrix during reactor operation. Bayonet coupling devices are fixed on the end fittings and allow a handling tool or a supporting tube to be coupled.

The fuel plates are the basic constituants of all the fuel tubes. Fuel plates are fabricated by the picture frame technique.

From 1971, standard BR2 fuel elements have been fabricated with cermet fuel cores. A standard element with 6 tubes contains 400 g of U-235. The uranium content per unit area of a fuel plate amounts to 60 mg U-235/cm². Burnable poisons, such as Sm_2O_3 and B_4C, are added to the fuel cores.

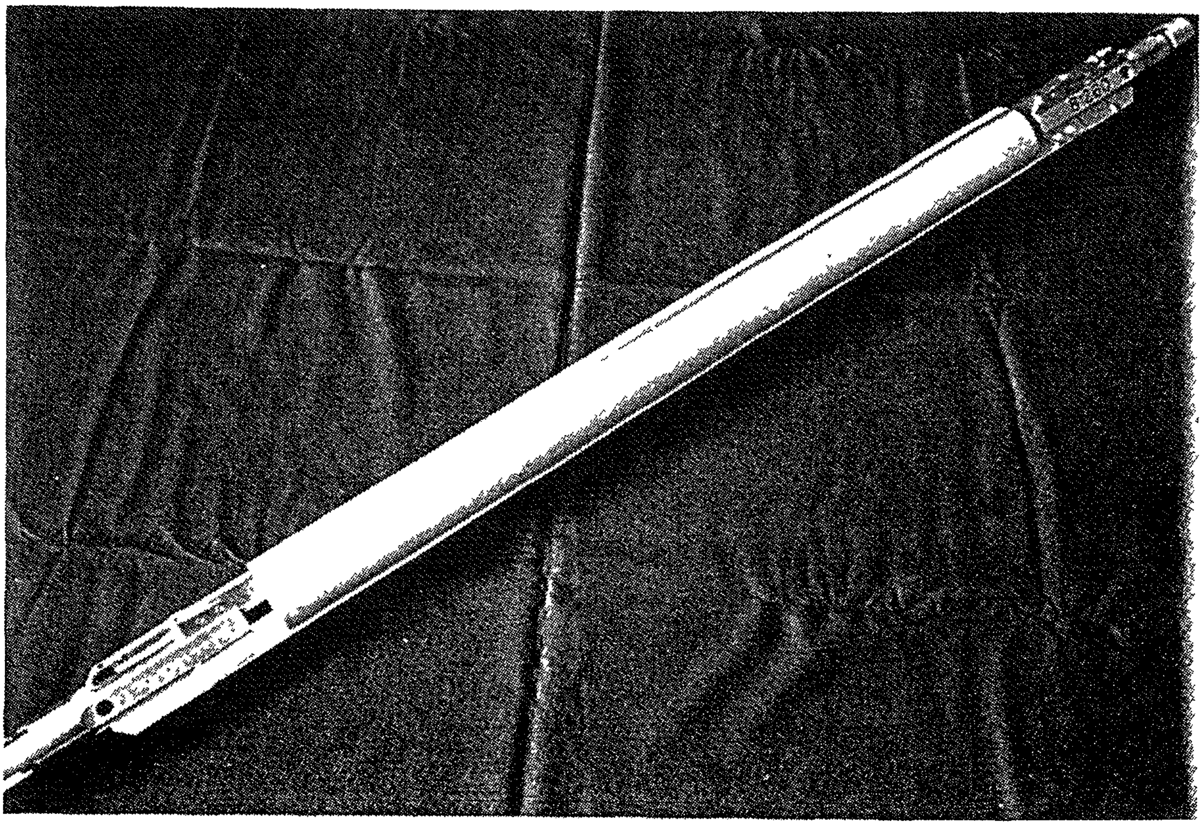

FIG 7 Standard fuel element

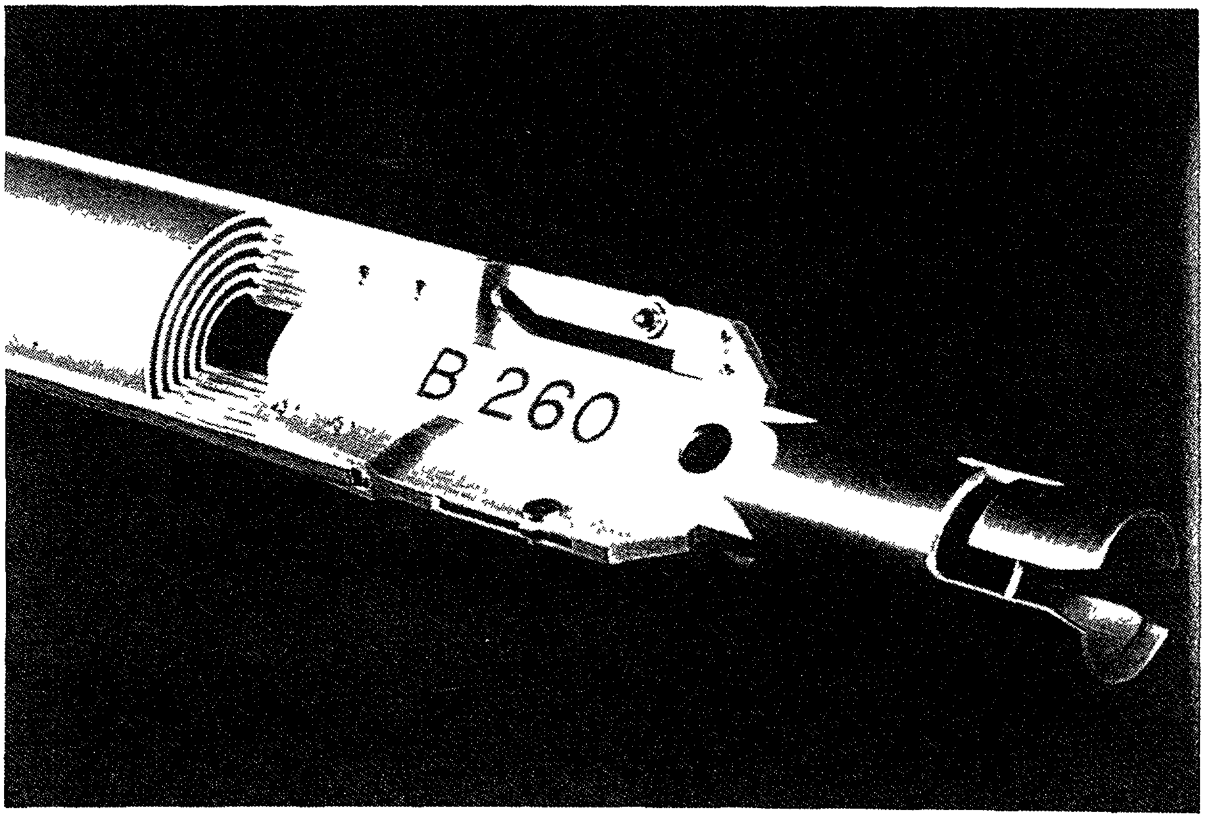

FIG. 8. Standard fuel element.

2.3. Storage racks

2.3.1. Critically considerations - general

A maximum limit of 0.90 has been fixed for K_{eff} for all storage locations in and around BR2, with the exception of the reactors BR2 and BR02.

Criticality calculations of assemblies of standard BR2 fuel elements are conservative if they are made for *fresh alloy elements containing 244 g U-235* - these alloy fuel elements were used before 1971.

2.3.2. Storage under water of BR2 fuel elements

2.3.2.1. Containment building

In the containment building, two storage racks are available. The grid distance between the axes is 108 mm. Every other tube is cadmium lined for the first rack. All the tubes are cadmium lined for the second (Fig. 9).

The K_{eff} is <0.74 for an infinite array of cells provided that at least every other tube is cadmium-lined.

2.3.2.2. Storage channel

Standard fuel elements can be stored in four different types of racks:

- For the first type the grid distance between the cell axes is 152,7 mm. There are no tubes, only an upper and a lower grid plate. In the lower grid plate, the fuel elements are locked by means of their end-fittings. The rack can store 56 standard fuel elements.

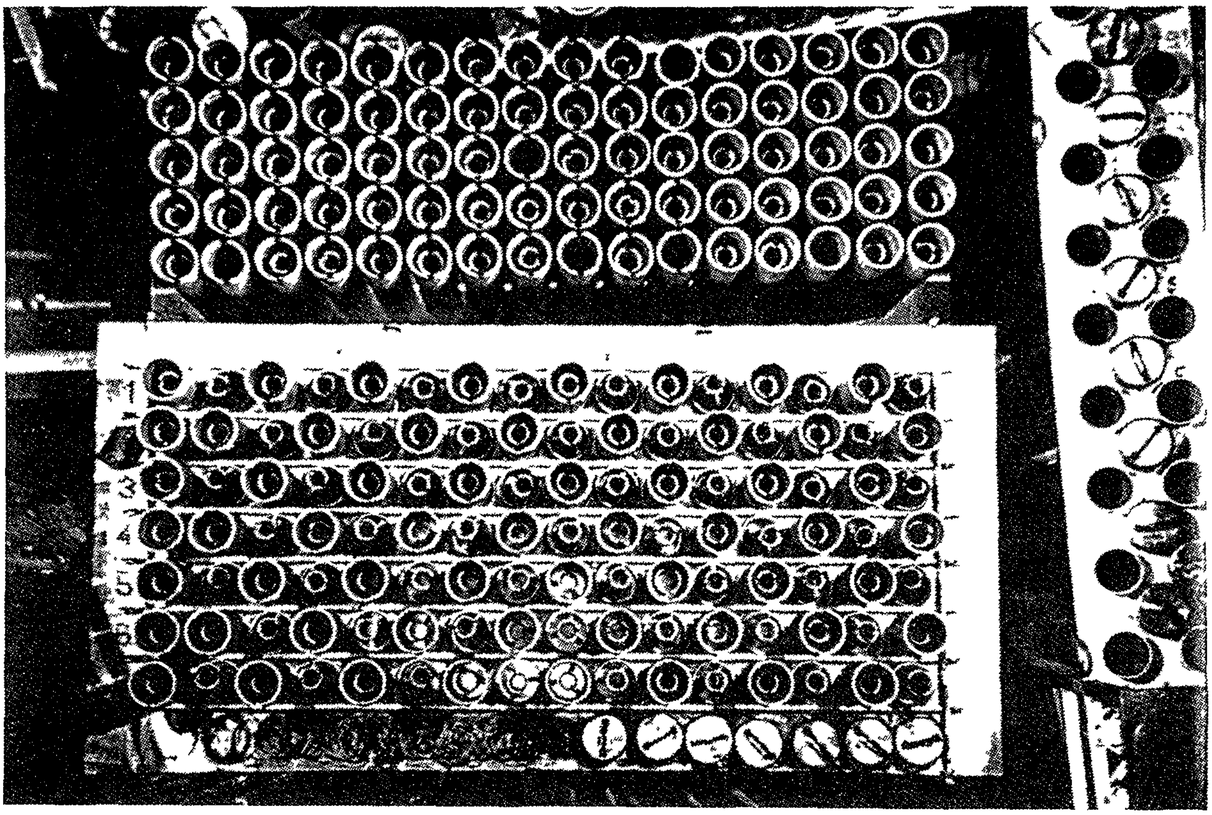

FIG. 9. Storage racks in the containment building

- For the second type (Fig. 10), the grid distance between axes is also 152,7 mm. Initially it was 108 mm, but later on half of them were mechanically blocked. Fuel elements can approach each other in such a way, that the distance between surfaces is 107.5 mm as in the previous case. The rack can store 56 fuel elements.
- For the third type, again the grid distance between axes is 152 mm. This rack can contain 150 standard fuel elements which have been cut for shipping to a reprocessing plant. Approach from alongside is possible up to 55 mm between surfaces.
- For the fourth type, a square lattice of 150 mm was chosen. One position out of two is equipped with a cadmium tube. The rack can contain 105 standard fuel elements.

An infinite array of fresh alloy elements in water with thick axial reflectors is subcritical when the grid distance is greater than 11,2 cm. The Keff value is about 0.63 for a grid distance of 152.7 mm.

2.4. Actual storage capacity for standard BR2 fuel elements

In the containment building, the two racks can contain 192 standard fuel elements. In the storage channel, the existing standard racks can contain up to 800 standard fuel elements. In January 1993, the storage capacity of this storage channel was nearly exhausted.

3. PLANS FOR STORAGE EXPANSION

3.1. Needs for storage extension

Late in 1991, a project was started to define, study and realize a storage expansion in the storage channel taking into account:

- the ultimate date for reactor shutdown (30 June 1996);
- the possible reactor power level in routine operation from 1992 on;
- a maximum of 210 days of operation per year.

A maximum of 400 new storage places were needed to allow the continuation of operation of the reactor under the assumption that no fuel could be disposed off in the meantime.

On the other hand, there is a need for a complete overhaul of the storage channel. Indeed, only compartments 091 and 092 were repainted twenty years ago. In many places, the paint has peeled off and signs of corrosion of the cladding are visible. Although there was no direct danger for the channel leaktightness (indeed a first partial inspection demonstrated that the visible corrosion was only superficial), the decision was taken to foresee the possibility of a storage channel overhaul, complete inspection and maintenance.

This means that:

- for each compartment, the stored fuel should be removed and stored at another place;
- each compartment should afterwards be isolated, emptied, inspected and maintained.

Moreover, all these activities should not interfere with the normal operation of the reactor.

The overhaul operations on the storage channel should begin with the compartment 093 with the largest storage capacity (540 fuel elements).

Based on these two needs - continuation of reactor operation and overhaul of the storage channel - the permission was asked from the Licensing Authorities to expand the storage capacity with 752 new storage places. By end of June 1992, a formal request with a safety analysis was prepared.

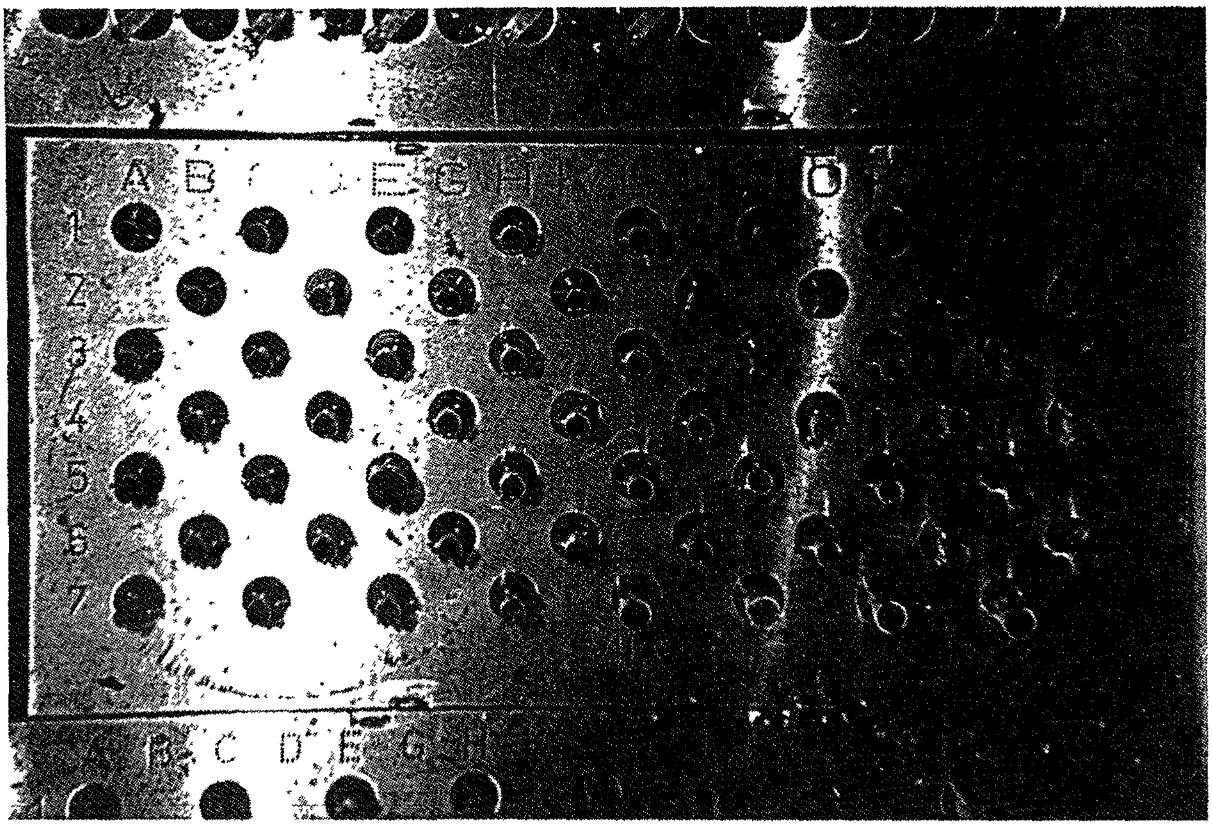

FIG. 10. Storage rack, first type, storage channel.

The project of storage expansion was examined and approved in February 1993 by the Safety Authorities under two conditions:

- each compartment should be inspected and maintained;
- an alternate solution to underwater storage at the BR2 site should be available for continuated operation of the reactor operation after 1996 i.e., after the shutdown for refurbishment.

3.2. Storage expansion study

A storage expansion with 752 new places involves a complete reshuffling of the storage channel with new high density storage racks. This means the study of new high density racks and the revision of a set of analyses for possible internal and external accidents.

The summary of the safety analysis case is presented in Table I and briefly commented upon below:

3.2.1. Design of the high density racks

The design is comparable to the design of the existing racks, i.e., dimensions, materials, general concept, etc., but the storage capacity is upgraded to a maximum of 112 places in place of 56.

This upgrading is realized by placing absorber plates between rows of storage places.

Each absorber plate is made of a cadmium sheet - 1.15 mm thick - sandwiched between two stainless steel plates. By construction, the cadmium sheet cannot move within the sandwich and *leaktightness* of the sandwich is assured by welding of the stainless steel plates. The dimensions of

TABLE I. STORAGE EXTENSION STUDY - SUMMARY

1. INTRODUCTION
2. REALIZATION OF THE STORAGE EXTENSION
2.1. Current storage capacity
2.2. Analysis of the needs for a storage extension
2.3. Design of a new high density storage rack
2.3.1. Basis for design
2.3.2. Specifications
2.3.3. Construction
3. SAFETY STUDIES
3.1. Review of existing studies
3.1.1. Internal accidents
3.1.2. External accidents
3.2. New studies in support of the storage expansion
3.2.1. Technical status of the storage channel
3.2.1.1. Description
3.2.1.2. Present situation
3.2.1.3. Conclusions and actions
3.2.2. Fuel element integrity and fission product release
3.2.3. Criticality studies
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3.2.5. External accidents
3.2.5.1. Review of conclusions of former studies
3.2.5.2. Reevaluation of basic assumptions and conclusions
3.2.5.3. Conclusions
4. CONCLUSION

the absorber plates - 852 × 802 mm - are chosen to cover more than the fissile height of the fuel elements. Each absorber plate is firmly attached to the top and bottom plates of the rack (Fig. 11).

Structural materials are made of stainless steel, except for the top and bottom plates made of aluminium.

For other reasons - see Section 3.2.2 - the storage capacity of the high density rack is reduced to *96* places by blockage of the central row.

3.2.2. Safety studies

3.2.2.1. Critically studies

Calculations have been performed with the Monte-Carlo code KENO-IV which is able to handle 3 dimensional problems.

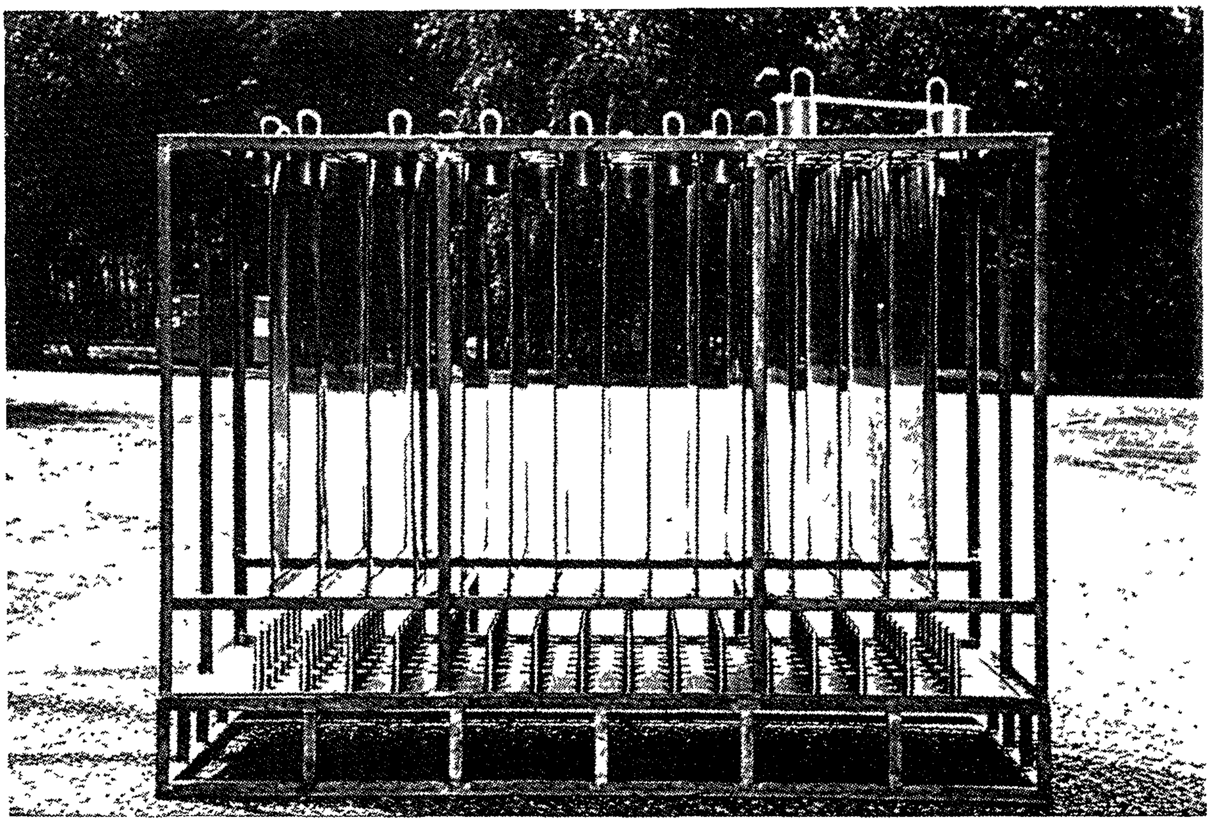

FIG. 11. High density rack.

For comparison, the same calculations were applied to the low density version of the rack (*56* storage places). The results are summarized in the following table:

CASE	K_{eff}
Low density (56 storage locations)	0.64
High density all positions occupied (112)	0.80
High density central row unfilled (96)	0.72

The high density racks thus satisfy in normal conditions the acceptance criteria K_{eff} 0.90.

Sensitivity studies have been performed by the same code to take account of possible deformations - compaction - of the racks under accidental conditions (see Section 3.2.2.2).

These studies showed that the original low density racks do not present reactivity problems under the studied accidental situations; it is also the case for the high density racks where the central row is not filled up.

3.2.2.2. External accidents

In 1988, the consequences of an airplane crash on the storage channel have been evaluated as well as the associated probabilities.

In the considered accident, a number of fuel elements corresponding to 2 racks are damaged and contribute to the atmospheric release.

As the 1988 study was considered as very conservative, the basic assumptions were critically reviewed and some reevaluated. It was so shown that even a higher density of fuel elements in storage could not result in a higher release to the environment.

Moreover the criticality studies performed (see Section 3.2.2.1) indicated that a criticity could be certainly excluded by limiting the storage to *96* places in each storage rack. So direct and indirect possible consequences from an airplane crash (fall of heavy objects in the storage pool) could be avoided.

In conclusion it was possible to show that the use high density racks did not jeopardize the current 'accepted' limits for the consequences of an airplane crash.

4. PLANS FOR IMPLEMENTATION OF THE STORAGE POOL EXPANSION

Two high density racks have been constructed. The first one is now already available in the storage channel as rack 15 in compartment 092.

As the license for the storage pool expansion (Section 3.1) will specify that each compartment of the storage pool must be inspected and maintained, a plan for a complete reshuffling of the storage pool was prepared.

The plan foresees that the compartment 093 should be first inspected and maintained (Fig. 12).

The reshuffling implies the use of two new high density racks (numbers 15 and 16), the transformation of 4 low density racks into high density racks (numbers 4HD, 5HD, 7HD, 3HD) and the removal of the BR02 reactor. The compartment 093 should be available by end of February 1994.

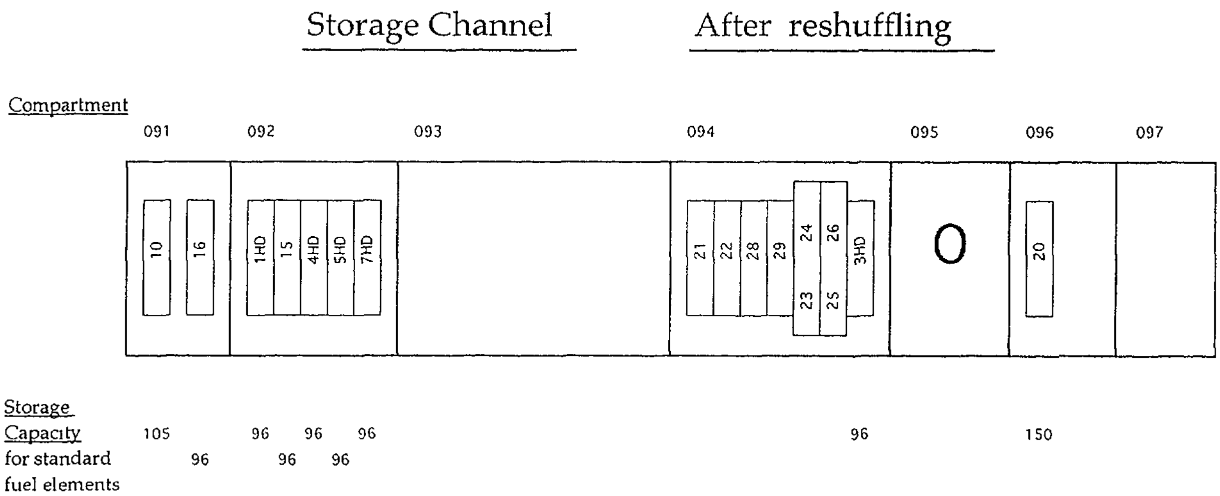

FIG. 12. Storage channel after reshuffling (March 1994).

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STORAGE EXPERIENCE WITH FUEL FROM RESEARCH REACTORS IN DENMARK

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Abstract

The DR 3 reactor at Risø National Laboratory is a 10 MW PLUTO type MTR. The reactor has been operating for more than three decades and the back-end of the fuel cycle is based upon reprocessing. Storage before shipment has up to now taken place in a pool with controlled water chemistry. The lapse of reprocessing unexpectedly declared by US DOE in 1988 has caused considerable problems. Firstly, old casks has had to be put into use as dry storage and, secondly, a dry storage facility had to be constructed. The latter is expected to be operational by the end of 1993.

1. BACKGROUND

Risø National Laboratory has for more than three decades been operating a 10 MW MTR of PLUTO type. The fuel elements are at present of the tube type design.

Highly enriched fuel has been used up to December 1990 when operation with a fully converted LEU core started.

Over the years 950 fuel elements have been reprocessed in the UK, about 70 elements in France and 684 elements in the USa. All fissile material has been of US origin except for those used in the early sixties, when fissile material was of UK origin.

The LEU fuel element geometry is identical to that of the HEU elements. The meat is U_3Si_2 in an Al matrix. The fissile content is about 10% higher than that in the HEU elements.

2. STORAGE

The normal spent fuel cycle comprises two dry storage facilities for storage before cutting. Cutting is done in a pool which is also used for storage before shipment. Subcriticality for the storage racks is controlled by geometry (distance between positions) and for handling by administrative rules. The storage capacity in the pool is 96 fuel elements, corresponding to a little less than three years of operation.

The water chemistry is controlled by means of a system with a filter and an ion exchanger. Conductivity and pH is surveyed on a monthly basis, specifications are: pH to be within 5-7 and conductivity to be less than 10 μS. The normal residence time for the cut elements in the pool has been a few years. A few elements have, due to special circumstances, (early LEU fuel elements), stayed in the pool for about 10 years.

Experience with water storage has been good, with no fuel elements having failed.

The cessation of shipments to the USA in late 1988 forced us to reconsider the situation. As a short time remedy three old casks were put in operation, increasing the storage capacity by 75 positions. As a longer term solution it was decided to construct a new intermediate dry storage facility, as described below.

The present storage situation is that the spent fuel inventory comprises 210 fuel elements of which 88 are HEU fuel and 122 are LEU fuel. 72 fuel elements are stored in the pool and 75 elements are stored in three casks.

3. INTERMEDIATE DRY STORAGE FACILITY

As mentioned above, the new storage facility was constructed to cope with delays in shipping and reprocessing.

It was decided that the capacity should correspond to at least 10 years of operation, and that dry storage should be adopted as the safety of longer term wet storage was not sufficiently documented. At-reactor location was preferred, as this would minimise operational costs. Further requirements were: vertical storage, no restrictions regarding burn-up, long decay time before transfer and ventilation.

The above principles resulted in a facility comprising four blocks placed vertically under the floor of the active handling bay of the reactor.

The top face of the blocks is a mild steel plate, level with the floor. Each block has 12 storage holes accommodating 9 sets of fuel tubes in each, i.e. the total capacity of the facility is equivalent to 432 fuel elements.

The blocks were prefabricated and placed in holes drilled into the ground under the floor. Prefabrication assured easy control of the concrete quality. The space around the blocks in the holes was filled with concrete grout and the top 1.8 m around the blocks was formed as a massive block cast in heavy concrete.

Each block is an octagonal cylinder 0.88 m across (between flat faces) and 7.375 m high.

The 12 storage tubes are arranged in a triangular mesh with a 226 mm centre line distance. The tubes are made up of mild steel and are mutually supported by spacers assuring correct geometry. The upper part of the holes are stepped and a mild steel plug 755 mm long constitutes shielding above the fuel. Each hole is closed by a disc with an O-ring which seals against the hole in the top plate. The bottom of each tube is closed by a mild steel disc with a small stud in the centre.

Inside the holes a stainless steel liner is suspended by means of a ring with a conical seat. Ventilation air is guided down through the liner and up through the gap between the two tubes.

Each storage hole houses nine sets of fuel tubes. To minimise handling three sets of tubes are stacked one upon the other on a common post and three post are stacked one upon the other.

The posts are made of stainless steel and as they, together with the liner made of the same material, are the only components in contact with the fuel tubes, no corrosion problems are to be expected.

A special cask will be used for transfer on the fuel tubes from the pool to the storage facility.

The facility is expected to be operational by the end of 1993.

SPENT FUEL STORAGE EXPERIENCE AT THE ET-RR-1 REACTOR IN EGYPT

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Abstract

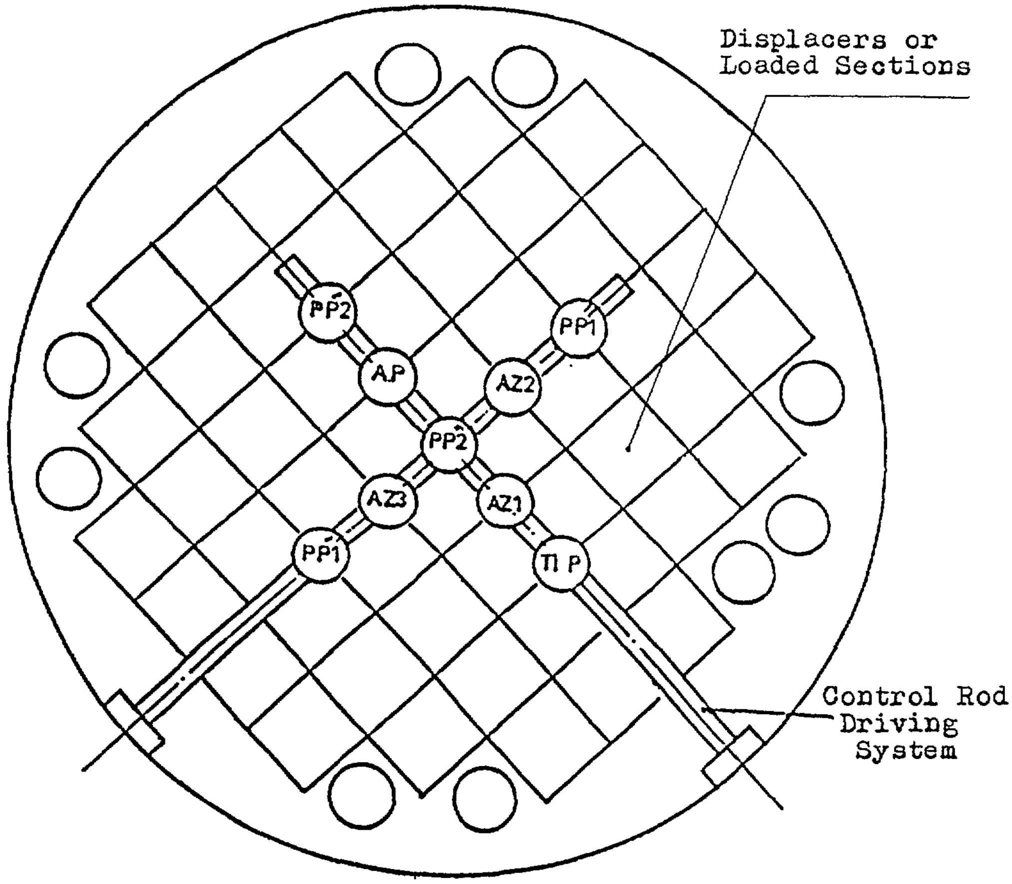
The pool type spent fuel storage is considered to be the most appropriate and flexible system. Recently, the pools which were designed to be short term storage for spent fuel became quasi-permanent storage because of the recent prohibition on reprocessing of spent fuel. One of the most economical ways to accommodate more spent fuel is to improve the efficiency of these stores making sure that the system remains subcritical as well as verifying other safety aspects. The effective multiplication factor of any array of fuel assemblies should be less than 0.95. The subcriticality in spent fuel storage facility is assured by adjusting geometric separation of the assemblies and also neutron poisons can be added. The results showed that it is quite possible to increase the capacity of Inshas reactor spent fuel store from 60 to 90 fuel elements maintaining well accepted safety margin by adding a second rack. The calculations indicated also that it is possible to store at least 50 fuel assemblies in an internal storage surrounding the ETRR-1 core. This internal storage will increase the excess reactivity of the reactor by 0.00385. This increment in the reactivity can be compensated easily by either reducing the fuel inventory in the core or by using the rod absorbers of the reactor control system.

Moreover, there are attempts to construct a compact spent fuel store (wet or dry) of capacity twice that of the present storage. This capability can be increased using fixed neutron poisons.

1-INTRODUCTION:-

The first Egyptian research reactor ET-RR-1 is a 2MW thermal reactor, tank type, with distilled light water as a moderator coolant, and reflector. The average neutron flux is 10^{13} n/cm^2. sec. It was designed for isotope production as well as in and out of core experiments. It went critical in the fall of 1961.

The reactor core consists of an aluminum cylinder for the fuel surrounded by a shield water tank. It was designed to accept a maximum of 51 fuel basket (Fig 1) each containing 16 fuel element.



- First Shim Manual Rods PP_1 and PP_1 .
- Second Shim Manual Rods PP_2 and PP_2 .
- Safety Rods AZ_1 , AZ_2 and AZ_3 .
- Automatic Regulating Rod AP .
- Precision Regulating Fine Rod П P .

FIG. 1. ET-RR-1 reactor core.

These fuel elements type Ek-10 are made of uranium dioxide dispersed in magnesium matrix, enriched by 10% U 235 in the form of rods cladded by an Al jacket (Fig 2). They are assembled in square arrays forming fuel baskets. Four configurations of baskets- depending on their position in the core are present, one, two, and three cut corner to allow for channels of control rods. (Fig 3).

2-Spent fuel storage

2.1 Design features:-

Adjacent to the reactor, under the main hall floor lies the spent fuel storage (Fig 4). The storage is assigned for keeping damaged and used fuel baskets. It consists of two tanks mounted one within the other, a receiver, a block of cells, pipe, frame, and a

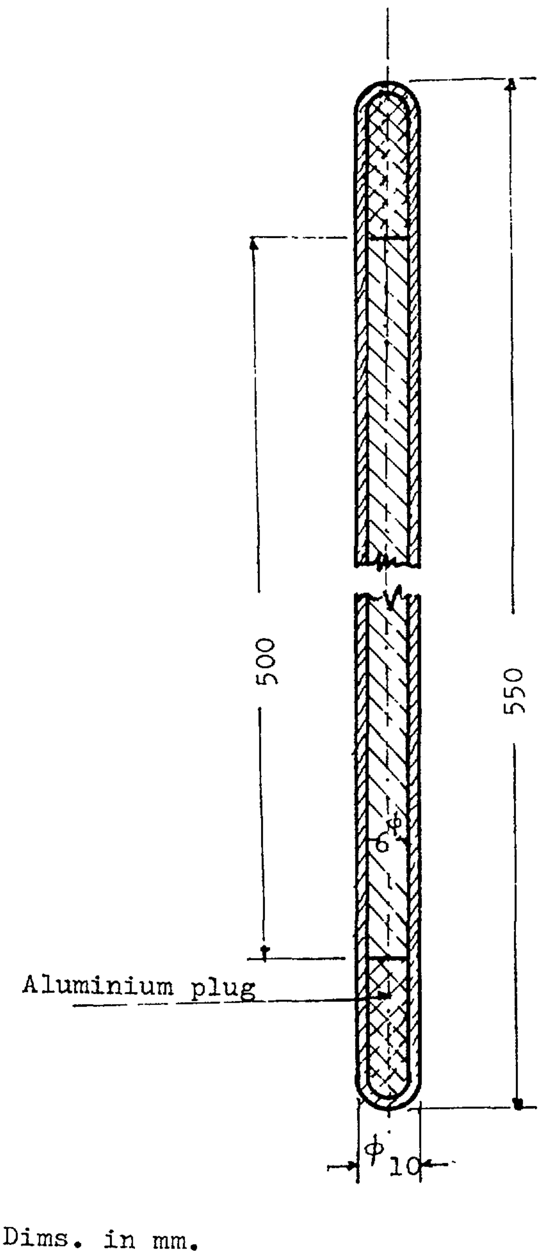


FIG. 2. Fuel element of ET-RR-1 reactor.

series of assemblies and parts. The inner tank has a rectangular shape and it is made of Al alloy, filled with distilled water 3m height for cooling and shielding against radiation. The outer tank is made of stainless steel.

A sloping spout is leading from the reactor to the storage, fuel baskets are sunk through this spout into the receiver and set on the bottom of the inner tank where the block of cells is mounted. There are 60 equally spaced cells in the block. One fuel basket may be set in each cell. In order to avoid displacement, the receiver is fixed to the tank wall. In order to avoid deformation

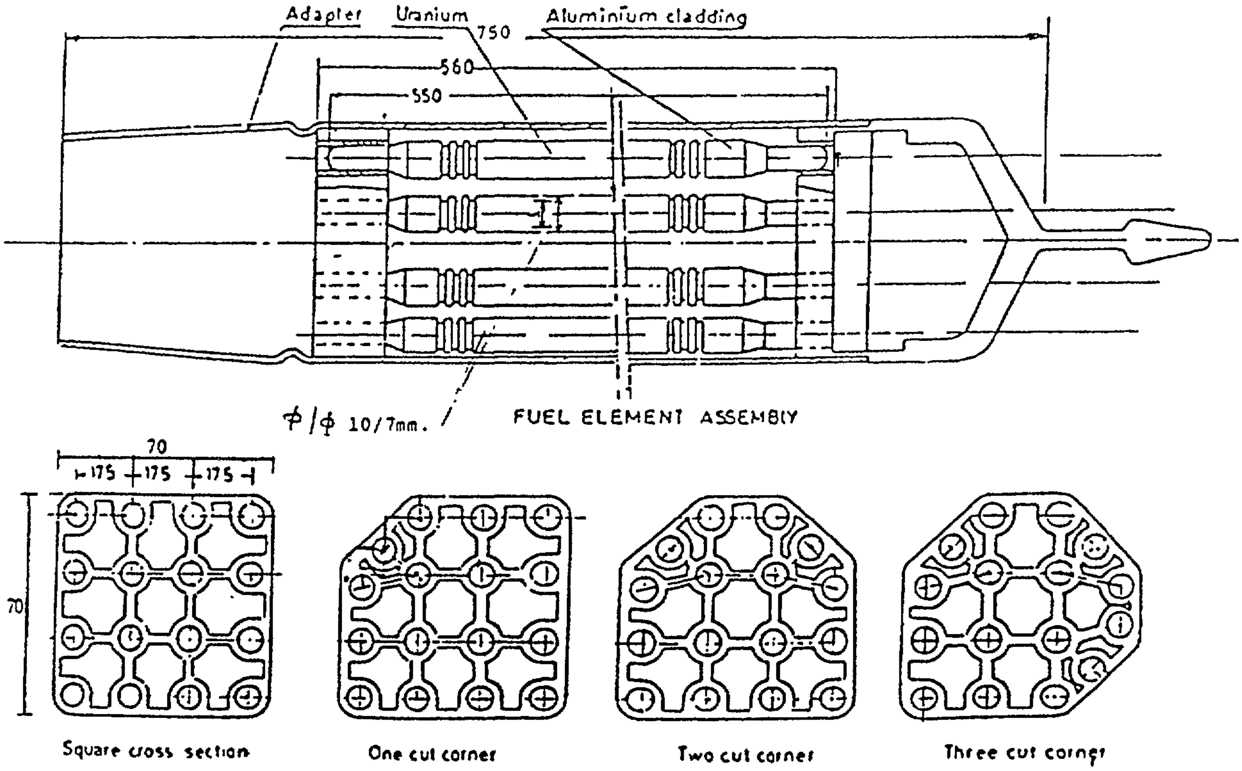


FIG. 3. Horizontal cross-section of the fuel element bundles.

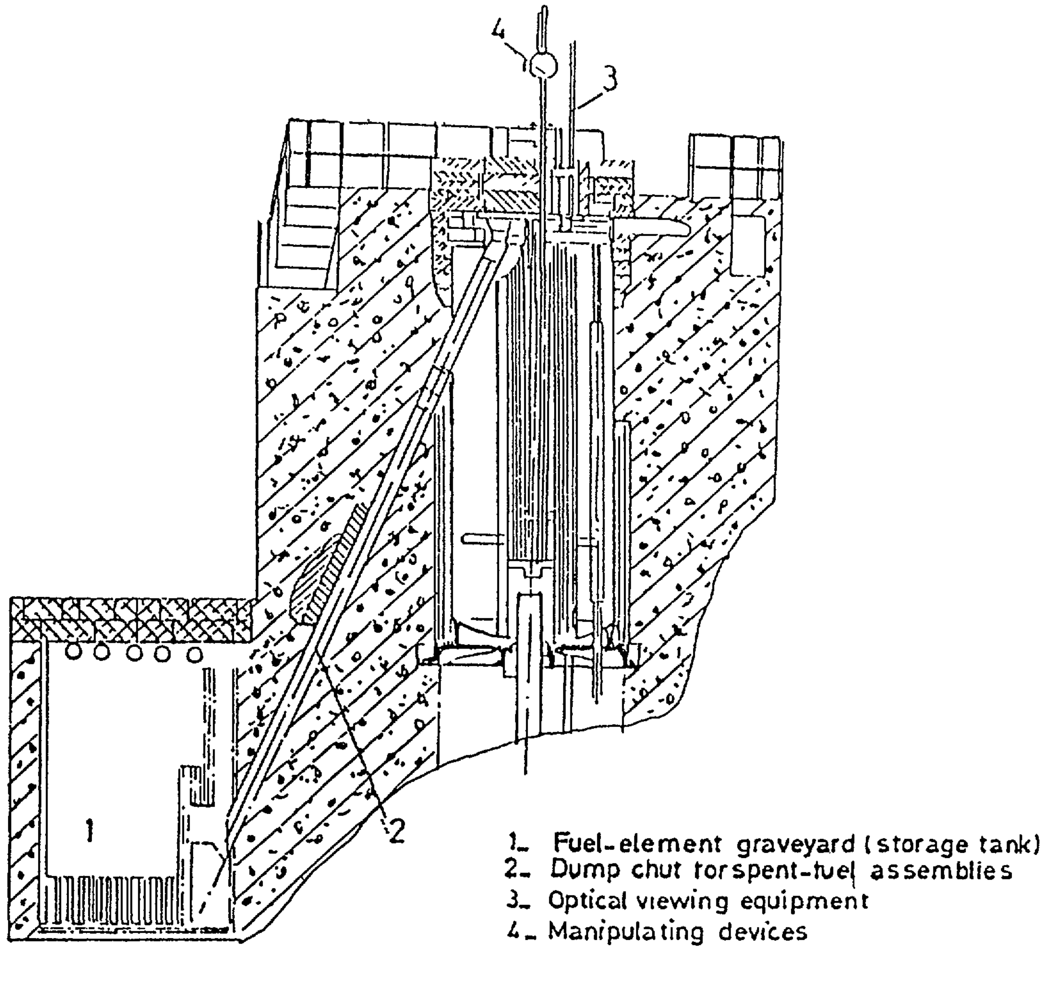


FIG. 4. Cross-section through the reactor at the depth of the fuel element dump chute.

of the fuel basket casing during their discharge to the storage through the transportation spout, the front wall of the receiver and its bottom are made of rubber (Fig 5). To maintain a constant water level in the storage tank, an overflow pipe with a hydraulic seal is fastened to the inner wall of the internal tank through which the excess of water running into the inner tank goes into drainage. On the bottom of the inner tank, there is a pipe for connection with the line leading to the level indicator. In the upper part of the tank there are openings through which the storage is ventilated, and a warning signal appears in the control room if a reduction in rarefaction or water level occurs [1]. The storage tank is connected to the special drainage system by a manual valve controlled pipe. The storage is top-covered with a 30 cm. thick cast iron shield consisting of three plates laid on lead pads in a frame. There are apertures in the plates (which are normally closed) through which fuel baskets are extracted from receiver to cell blocks. The storage is filled with water by gravity feeding from the distillate tank through a pipe penetrating the cast iron shield.

2.2 Shield against radiation

The upper shield from activity consists of a water layer 3 m height in addition to a 30 cm thick layer of cast iron. The thickness of the side concrete shield with (specific gravity 3.2 gm/cm^3) is 140 cm.

2.3 Nuclear design of the storage:-

From the nuclear point of view the storage is designed to ensure subcriticality where k is calculated to be equal to 0.945 (Fig 6).

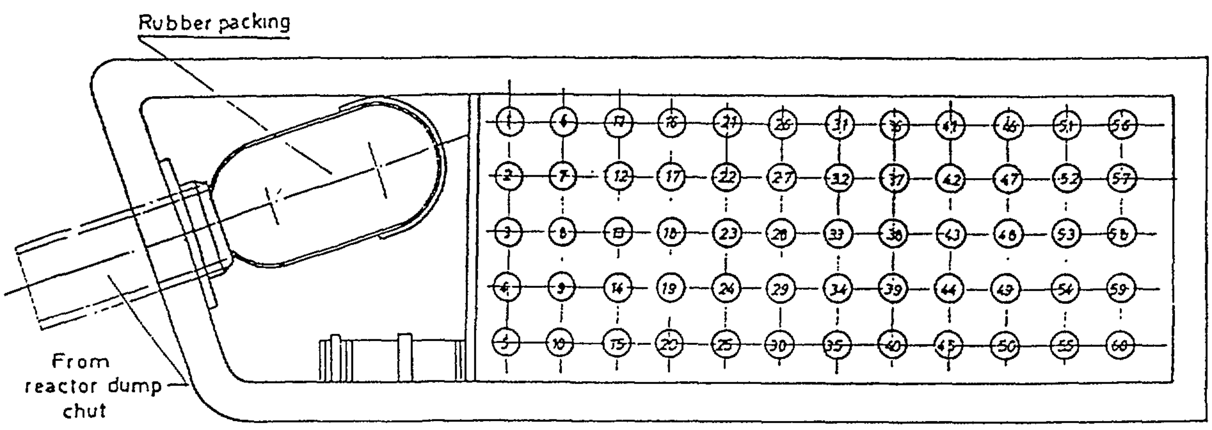
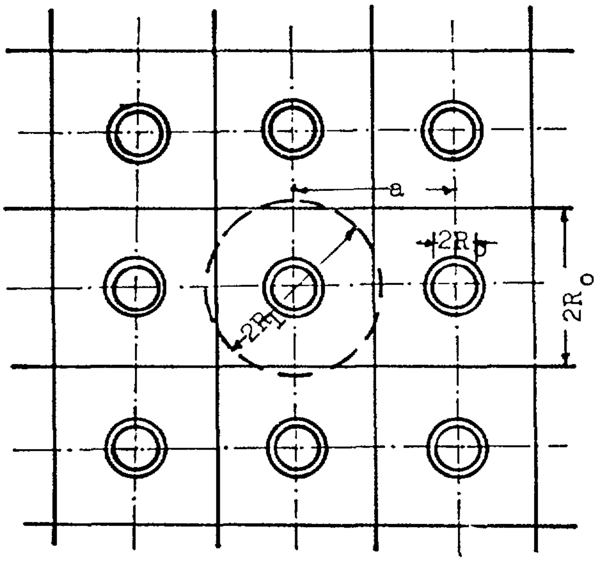


FIG. 5. Spent fuel storage.



Fuel elements height H = 500 mm

FIG. 6. Physical model for nuclear design of spent fuel storage.

3. Need for expanding the spent fuel storage

As was previously mentioned, the spent fuel storage contains 60 places for burned fuel. Up till now 37 places are filled and 23 are still empty. There are 44 fuel baskets in the reactor core. Therefore, in an emergency situation where it is necessary to evacuate the core, the empty spaces in the storage can not accommodate all fuel baskets in the core.

An IAEA project was agreed upon to make a safety analysis report and in service inspection of ET-RR-1 reactor. The SAR was prepared , but the ISI could not be done because it was necessary to evacuate the core from water and fuel, since there was no extra place for the fuel to be transferred [2] to, this part of the project was not achieved and an IAEA mission came to Egypt, discussed the situation and recommended the extension of the spent fuel storage before beginning the ISI [3]. For these reasons, it was decided that it is important to increase the capacity of the present storage.

4. Need for Building a new storage

In the summer of 1992 an in service inspection for the spent fuel storage was made with the help of Russian experts from PNPI, Catching, San petersburg. It was possible to inspect the storage walls and the free parts of the bottom only because the tank bottom has no access due to lattice design since the supports are welded to the tank bottom. The visual inspection was done using a TV camera and video recorder to evaluate the conditions of the metal surface i.e detection of metal defects (pits, cracks, cavities... etc) as well as the geometrical size of detected defects. An Ultrasonic method was also used to measure the defect depth and material thickness.

The inspection showed a considerable number of white spots which were considered as corrosion products of Al alloy (Fig 7). The average diameter of these spots is 30 mm. When scrapping one of these spots, a pit appeared of about 10 mm diameter and 2 mm depth. Deposits of about 10 mm thickness were found on the surface of the tank bottom.

These damages were attributed to the poor quality of water among other things. It was noticed that water quality is different from one place to another particularly under the lattice which has only an insignificant water exchange with other places in the tank, There is no water circulation inside the storage and also there is no system of water purification. A final decision on the status of the storage tank was not taken. To come to a conclusion it is necessary to carry out additional studies which are possible only when the storage is empty from fuel.

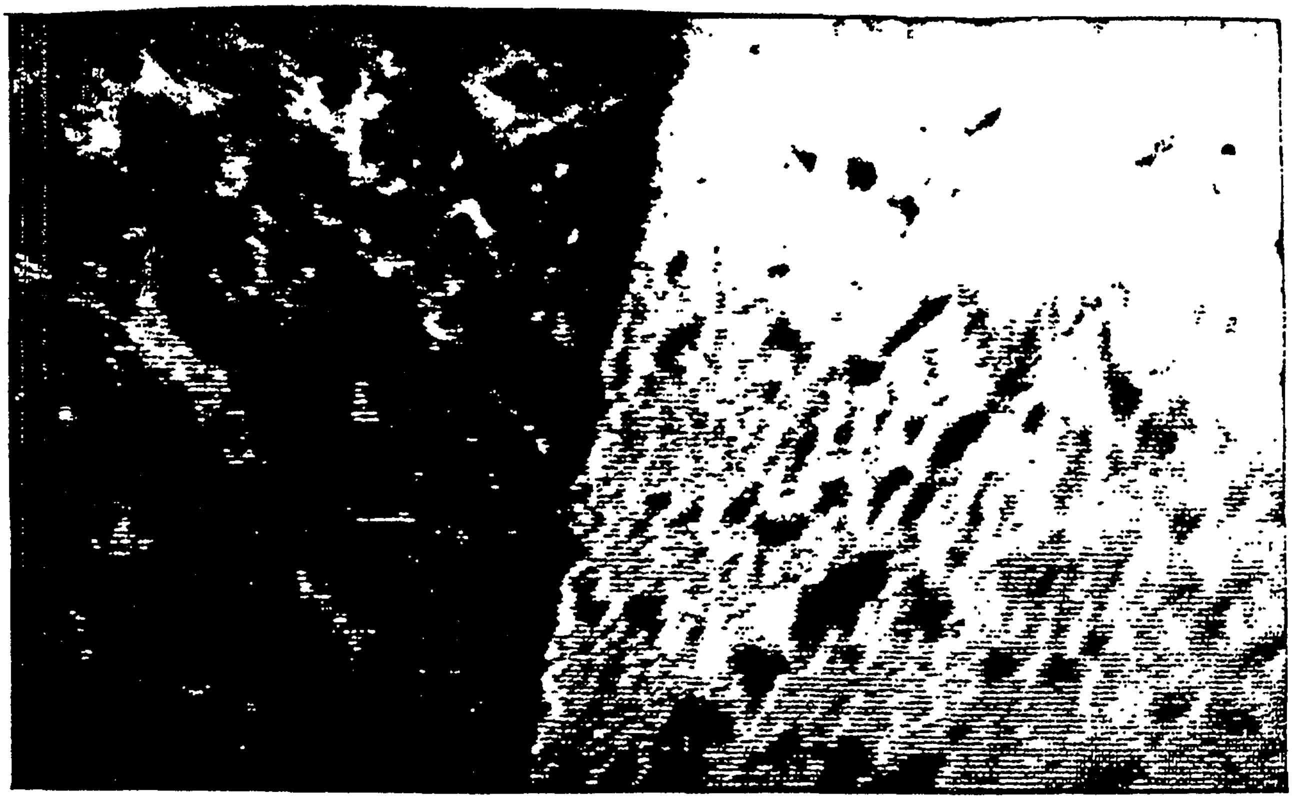


FIG. 7. Deposits on the surface of the fuel storage tank.

A construction of a new spent fuel storage provided with an independent system of water purification which makes possible the water circulation in the storage tank was recommended.

5- Extension of the storage

As a result of the mentioned conclusions, four alternatives were discussed, namely, expansion of the present storage by adding a second rack, building another wet or dry storage or the construction of an internal storage in the shield tank.

5.1- Expansion of the present storage

For the first proposal i.e. putting a second rack in the present storage, physical calculations were done that intended to calculate the number of extra fuel basket that can be added in the storage without increasing the gamma doses outside the storage over the permissible limits [4] are calculated making sure that the multiplication factor does not exceed the limiting value in order not to approach criticality [5].

In order to find out the number of fuel baskets which can be stored in the water pit, it is necessary to calculate the total level of radioactivity which can be stored by the shielding material used [6]. When the storage is opened (case 1) the shielding material is an ordinary distilled water ,3m thick. When the storage is covered (case 2), the shield consists of 3m thick distilled water plus 30 cm cast iron. The dose outside the shield is given by:

$$Gamma\ dose = S\ e^{-\mu R}.B/4\pi R^2 \qquad case1$$

$$= S\ e^{-(\mu_1 R_1+\mu_2 R_2)}.B_1 B_2/4\pi\,(R_1+R_2)^2 \qquad case2$$

where

S= source strength in Curie

U= linear attenuation coefficient in cm

R= thickness of the shielding material in cm

B= build - up factor

For determining S, the fuel history was traced and radioactive decay was calculated. The following developed formula can accurately calculate S [7],

$$S = \sum_{i=1}^{NL} \frac{Pi}{NFI} \sum_{j=1}^{NFI} \left[\ (T_{ij}-T_{oj})^{-0.2} - (T_{ij})^{-0.2}\ \right]$$

where,

NL = total number of loadings,

NFl = number of spent fuel baskets in each loading,

Pi = power in watts generated from spent fuel storage,

Tij = total time (operation + cooling),

Toj = operation time

To be more safe, calculations of the effective multiplication factor were made based on the assumption that the fuel is fresh [8] i.e has its original enrichment. the calculations of radioactivity were made based on the assumption that the fuel is just taken from the core to be loaded in the storage without decay, i.e without cooling time. So, large safety margins were taken into consideration for both reactivity and radioactivity calculations Also emergency evacuation was taken into consideration.

Reactivity calculations were done using a computer code in which the multigroup diffusion equation in 3- dimensions is solved by finite difference technique [9]. Results of calculations showed that 30 fuel baskets can be added . A rack was designed (Fig 8) which consists of a stainless steel tube arranged in the form of a rectangular frame with hooks. This rack can be put in the peripheries of the original grid at a distance of 80 cm above with 30 hooks for carrying fuel baskets. These hooks were fastened at certain distances such that the fuel basket to be added will be in the middle between every two fuel sections originally placed in the storage. This situation will not prevent extraction or insertion of fuel baskets in the lower grid originally present . The rectangular frame is fastened on four legs that go down in the storage corners. The rectangular frame was designed to resist different stresses exerted so that the hooks and the frame can bear the fuel weights without any deformation. It must be noted that the addition of this frame will not prevent the use of any measuring or purification system inside the storage.

Fuel sections that have a long cooling time and consequently a lower radiation activity will be hanged on the hooks in the new frame. This is due to the fact that the shield against radiations from these elements will be less than the original shield of fuel in the lower grid by the distance between the rack and the original grid. These fuel baskets having more cooling time were found to be the elements taken out of the core at a time more than 11 years.

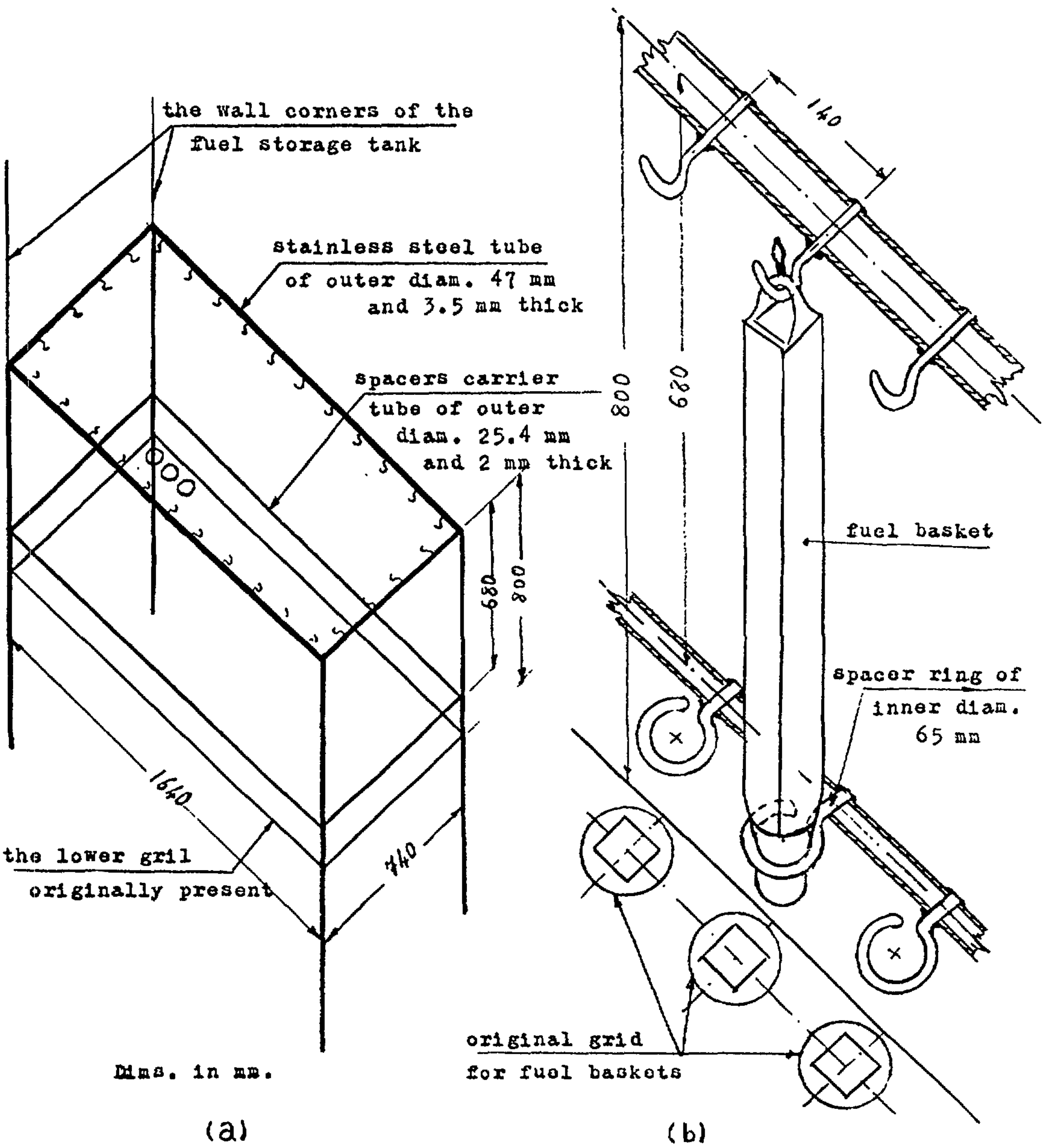


FIG. 8. A rack with 30 hooks is placed above the old fuel storage tank.

5-2 Wet storage

Another proposal considered is to construct a new spent fuel storage in the pump room of the reactor (Fig 9). This proposal consists of two versions [10].

Version A:

In this version the proposed storage will be installed in the site of the present filter [11] . The storage consists of two vessels made of steel, a main vessel and a backing vessel. The storage is equipped with a circuit of water treatment and system of sampling, filling and discharging pipelines and systems of water

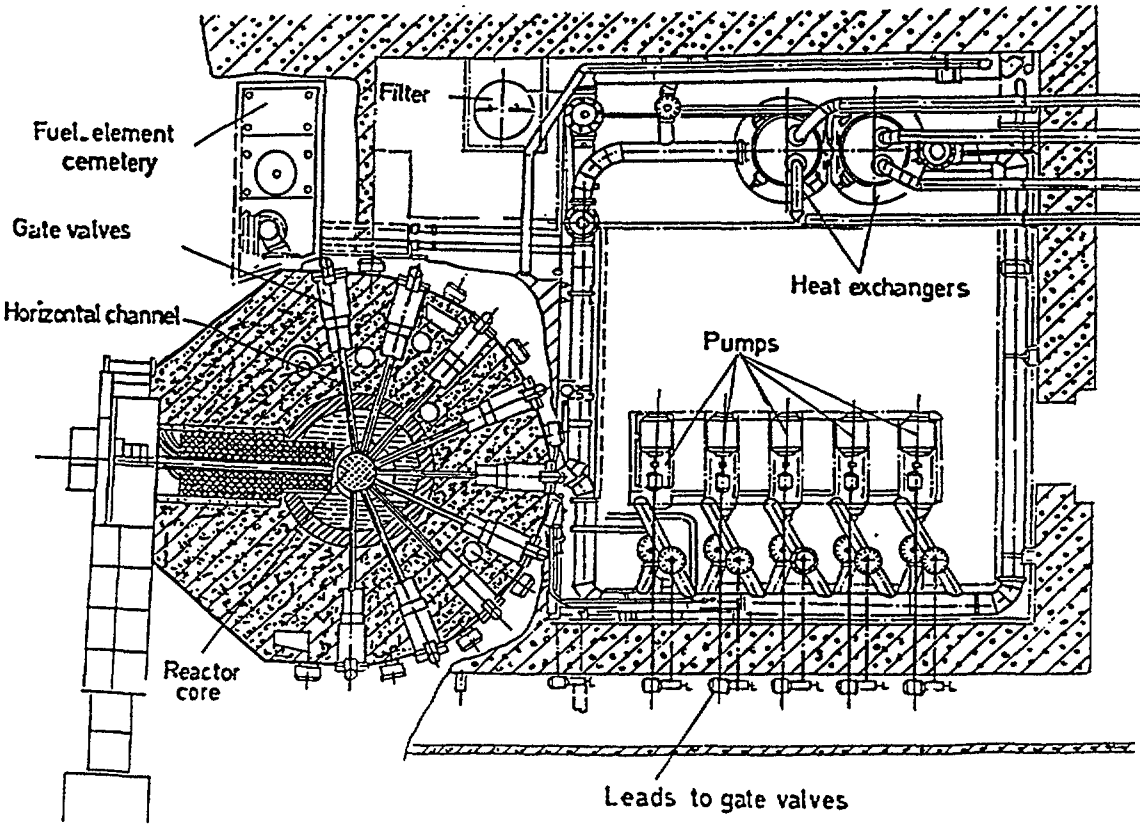


FiG. 9. ET-RR-1 reactor.

leak monitoring. The storage has iron shield from both sides. It is loaded and unloaded with the help of transporting container via the opening in the floor of the main hall (beside the present storage and lies directly above the filter) the capacity of the storage is 330 fuel assembly .

Version B

In this version the storage is placed in the site of the present dearator. The capacity of the suggested storage is 110 fuel assemblies. The storage design is the same for both versions as well as the volume of the dismantled equipment. The suggested version, A has a number of advantages compared to version B. The storage capacity is higher, besides transport system for loading and unloading of fuel is less complicated.

5-3 Dry storage

Another proposal was discussed that is to build a dry storage in a tank already present outside the reactor building that was built for solid waste but it was not used yet. There are 4 tanks connected to each other and have the dimensions shown in (Fig 10).

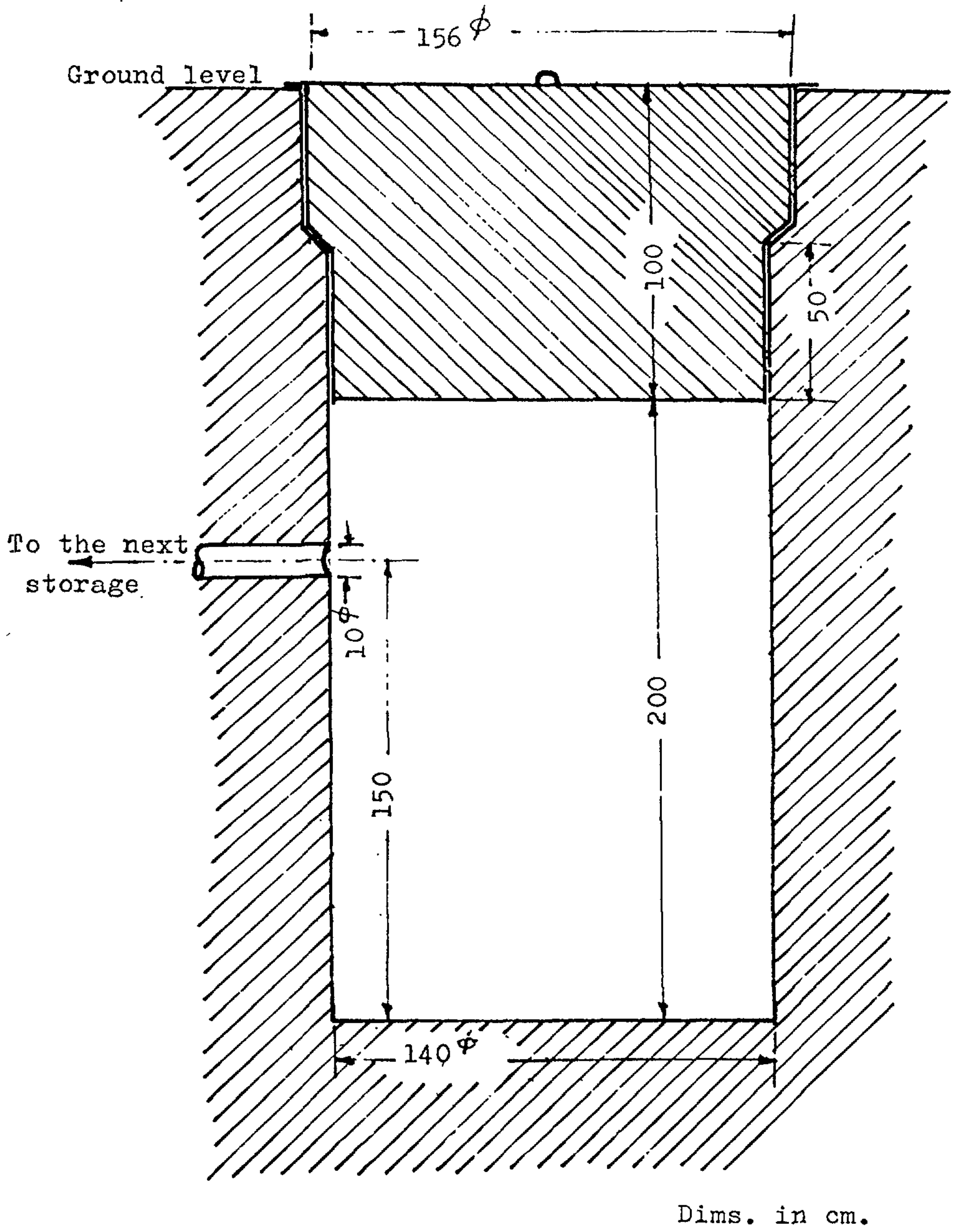


FIG. 10. Proposed tanks for dry storage.

Fuel elements that were taken from the core from a long time and undergone suitable activity and thermal decay can be transferred to this storage. But this proposal is still under study specially the way of transferring fuel from reactor building to this tank and also the way of making this tank under control.

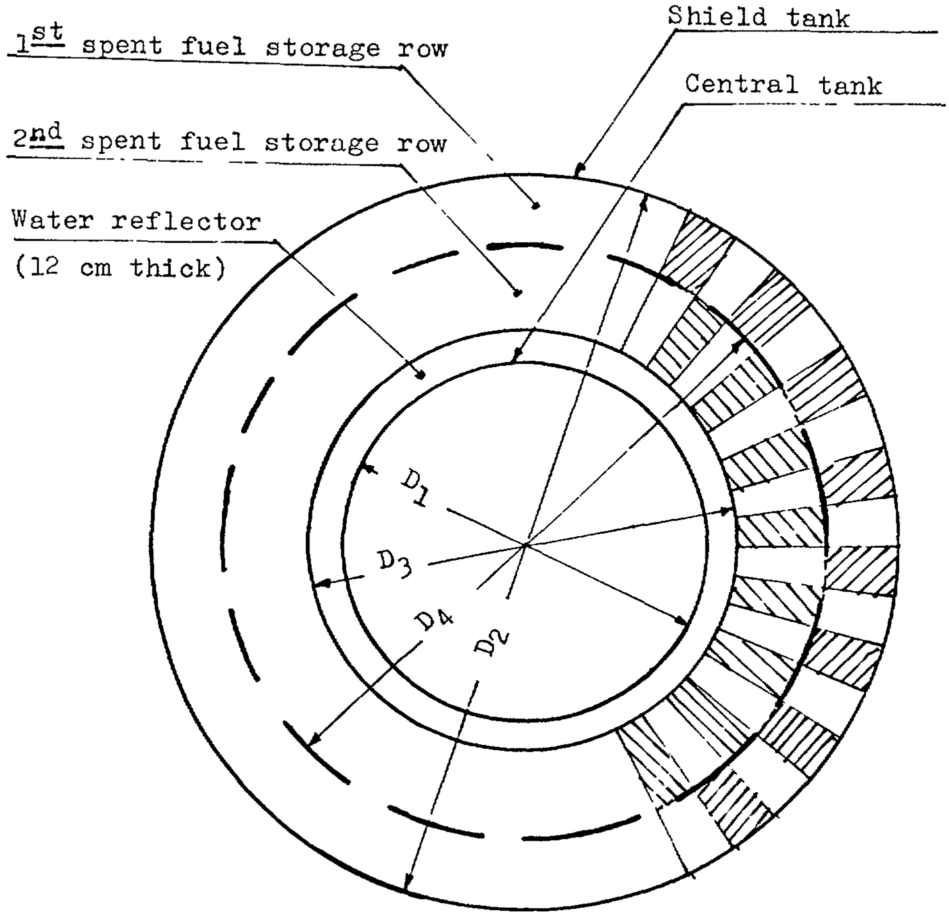
5.4 Internal storage

Internal core storage of island - type is considered as a special feature in core design for some reactors [12]. This island type gives the facility to accommodate a large number of spent fuel assemblies. These subassemblies are stored in steel flasks that are fixed in special racks on the core support structure. This design has the advantages that the nuclear power released is smaller than the decay heat thus *allowing cooling by natural convection.*

The calculations were performed for ET-RR-1 reactor. In these calculations the present water shielding is considered to act as a reflector (with thickness 12 cm) and an internal storage. Both reflector and internal storage play the role of the shielding.

The calculations indicated that it is possible to store at least 50 spent fuel assemblies in two circular rows around the reflector, as shown in (Fig 11) maintaining the operating safety conditions of the reactor.

This internal storage will increase the excess reactivity of the core by 0.00385. This increment in reactivity can be compensated either by reducing the fuel inventory in the core, or using rod absorbers of the control system.



D_1 = 1.130 m
D_2 = 2.300 m
D_3 = 1.370 m
D_4 = 1.830 m

FIG. 11. Setup of the proposed reactor internal storage.

The study is extended to design a compact spent fuel store which has the capacity of twice the present storage. This capability can be increased using distributed or fixed neutron poisons. The new storage can be constructed in a suitable site such as the pump room of the reactor.

ACKNOWLEDGEMENT

The authors wish to record their thanks to Prof. Dr. Sultan for his idea about the extention of the fuel storage tank and his valuable discussions.

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STORAGE EXPERIENCE WITH FUEL FROM RESEARCH REACTORS IN FRANCE

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Abstract

Interim storage of spent fuel at research reactors in France together with French policy based on reprocessing is described. Back-up solutions for spent fuel storage problems involving the PEGASE and CASCAD facilities are outlined.

1. INTRODUCTION

A list of research reactors still in operation in France is given in Table I. Only five of these twenty reactors burn significant quantities of fuel each year. Between 1974 and 1976, CEA reprocessed about 13,5 tons of high-enriched-uranium (HEU) coming from 21 (French and foreign) reactors at the Marcoule site (UP_1 -plant).

From 1976 up to 1988, spent HEU fuels from French reactors were sent to the United States Department of Energy (DOE) Savannah River Site (SRS) or Idaho National Engineering Laboratory (INEL) for reprocessing according to the US DOE's Off-Site Fuels Policy. In 1988, the US-DOE decided to suspend acceptance of spent fuel from abroad.

Since 1988, CEA operators have stored their spent fuels in both:

- the pools of the reactors, and
- the interim storage facilities (PEGASE pool and CASCAD dry storage).

TABLE I. REACTORS STILL IN OPERATION

*	-	OSIRIS	SACLAY
	-	ISIS	SACLAY
*	-	SILOE	GRENOBLE
	-	SILOETTE	GRENOBLE
*	-	ORPHEE	SACLAY
	-	ULYSSE	SACLAY
	-	EOLE	CADARACHE
	-	MINERVE	CADARACHE
	-	MASURCA	CADARACHE
	-	PHEBUS	CADARACHE
	-	CABRI	CADARACHE
	-	SCARABEE	CADARACHE
(*)	-	PHENIX	MARCOULE
	-	CHAUDIERE AVANCEE PROTOTYPE	CADARACHE
*	-	REACTEUR A HAUT FLUX (R H F)	GRENOBLE
	-	HARMONIE	CADARACHE
	-	CRONENBOURG	STRASBOURG
	-	MIRENE	VALDUC
	-	SILENE	VALDUC
	-	AZUR	CADARACHE

* Reactors which burn a large number of fuel elements each year.

A schematic diagram of a standard fuel element used in the OSIRIS research reactor is shown in Fig. 1 and a schematic diagram of the storage containers for spent fuel used in the PEGASE pool is shown in Fig. 2. The long term dry storage facility CASCAD is illustrated in Fig. 3.

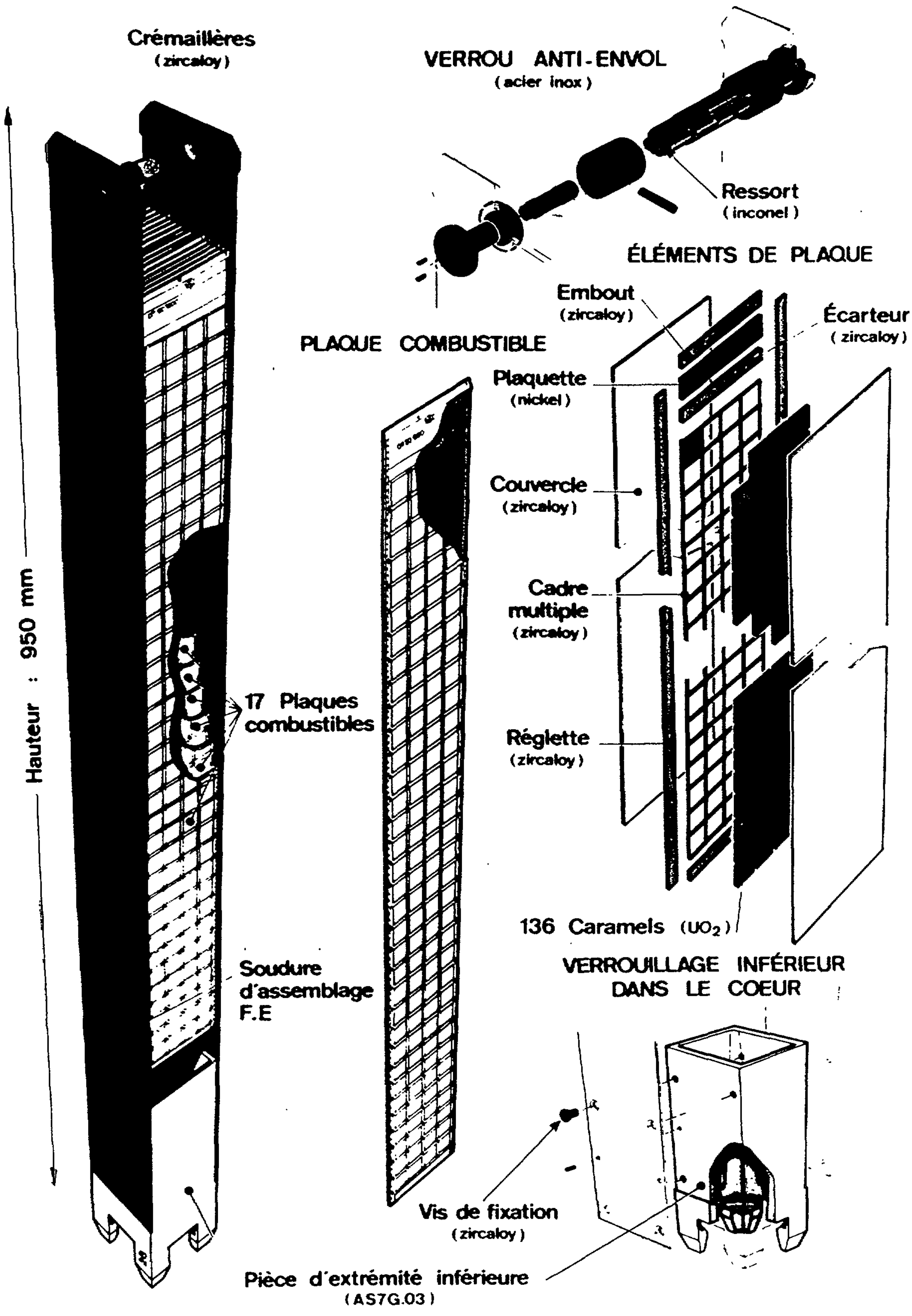

FIG. 1. Standard OSIRIS fuel element.

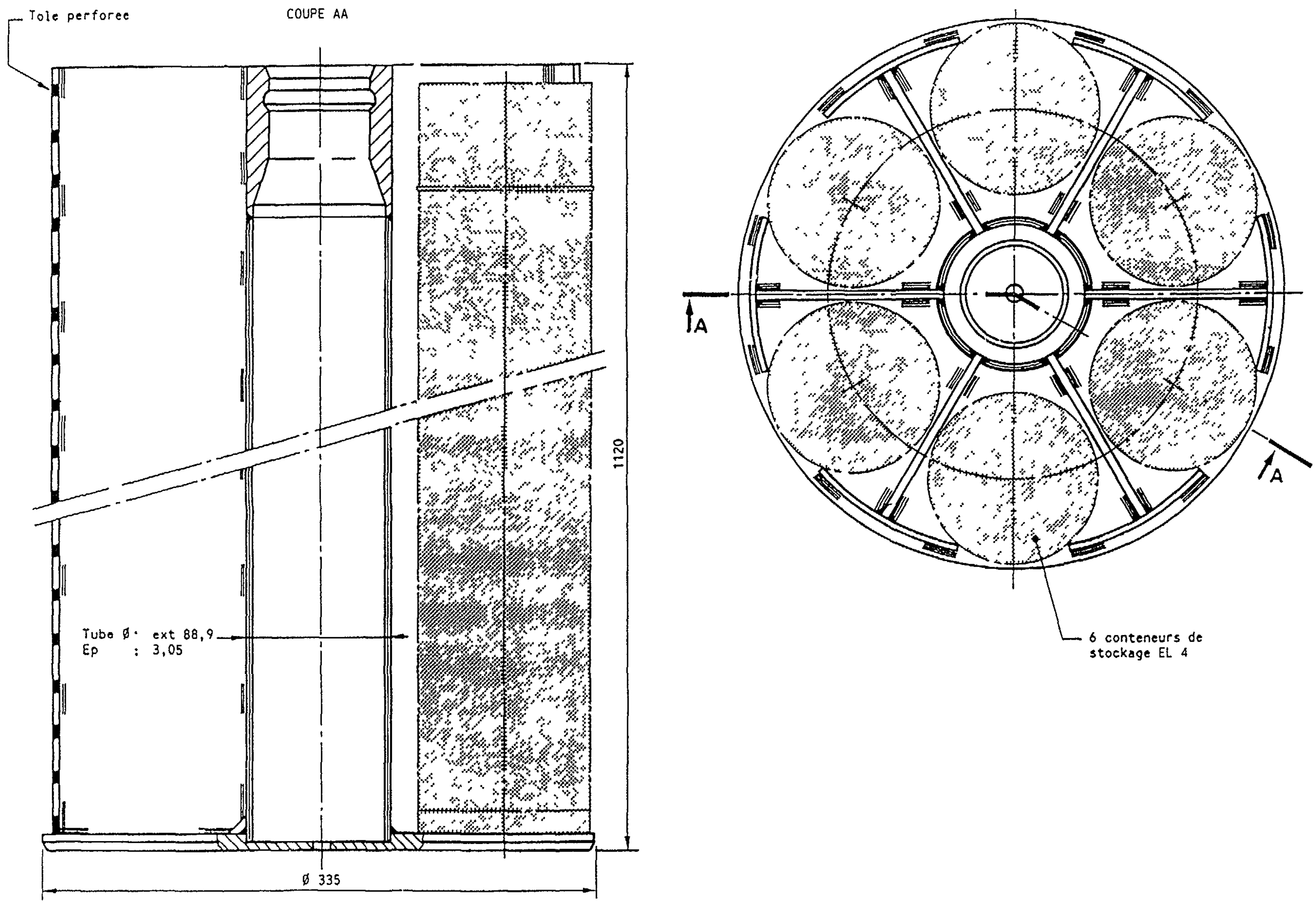

FIG 2 Schematic diagram of storage containers for spent fuel used in the PEGASE pool

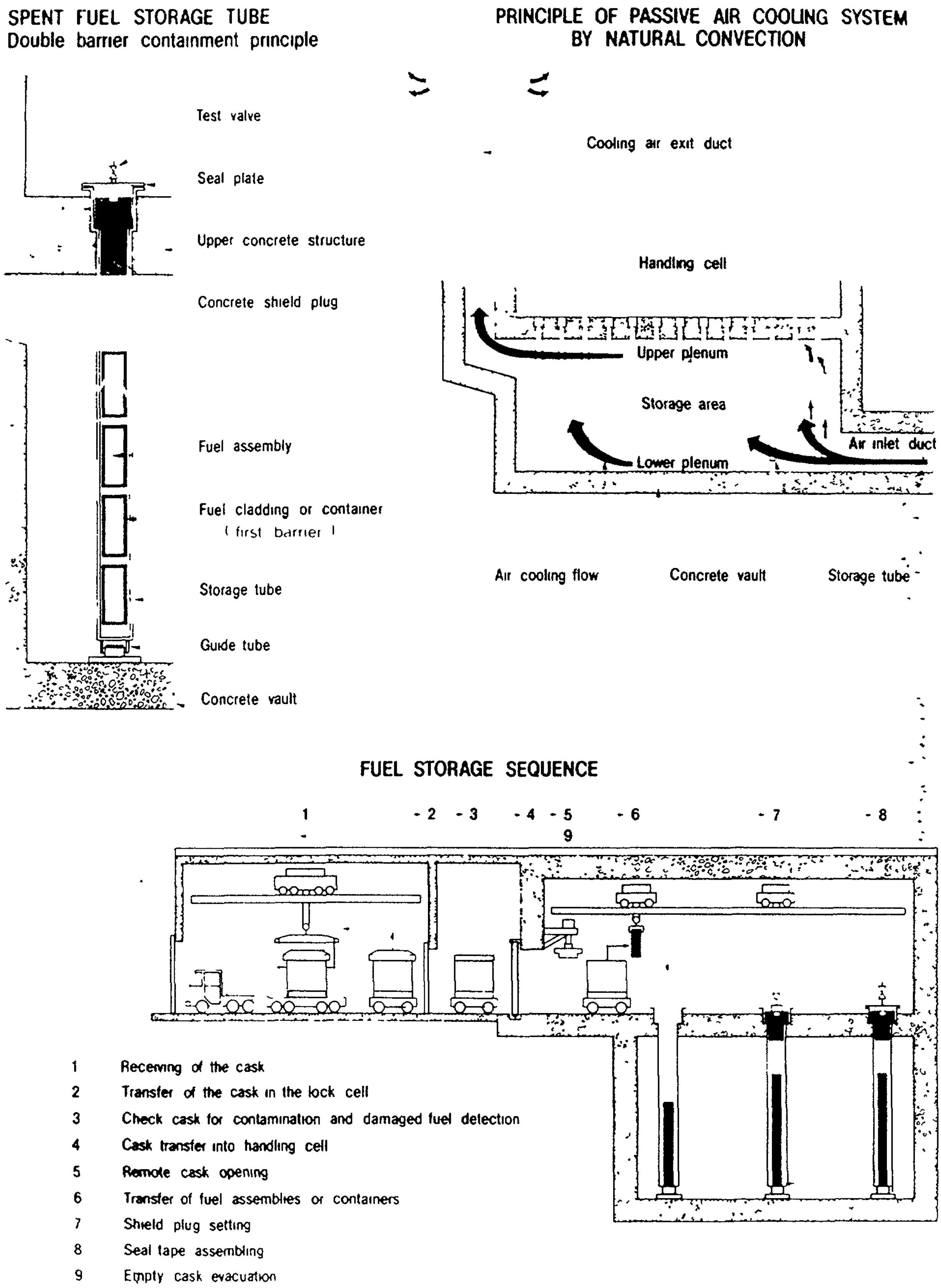

FIG. 3 Long term dry storage facility CASCAD

2. FRENCH POLICY IS BASED ON REPROCESSING

The decision has been taken to reprocess all of the Caramel fuel from the OSIRIS reactor (6500 kg of U_t) in Marcoule using a combination of the facilities of 3 different plants:

- the ISAI Laboratory (CEA) for dismantling,
- the APM (Atelier Pilote de Marcoule) plant (CEA) for dissolution, and
- the UP_1 plant (COGEMA) for extraction.

This campaign has to be finished in 1997, since the UP_1 plant will be shut down at the end of 1997.

At the time of writing, no decision had been taken on the site where HEU reprocessing will take place i.e., at the AEA Dounreay plant or the COGEMA UP_1 plant. Since the continued operation of the Dounreay plant depends upon the receipt of a sufficient number of contracts to justify further reprocessing campaigns, it too may close in the near future. Consequently, COGEMA needs a CEA commitment very soon to plan the campaign before final shut down of the UP_1 at the end of 1997. In both cases decision is expected soon. The decision does not involve technical considerations but is simply a financial decision since no budgetary provisions have been made.

3. TWO BACK-UP SOLUTIONS

It is planned to build a new extraction workshop in the APM plant after 1997 (end of UP_1 operation), the capabilities of the APM would be around 5 tons/year. Or, a renewal of the US DOE's Off-Site Fuels Policy might allow the return of fuels originally enriched in the USA.

To manage the situation until end of the century PEGASE and CASCAD interim storage facilities are large enough to cope with the generated spent fuel. The general policy and storage possibilities are outlined in Table II.

TABLE II. GENERAL POLICY AND STORAGE ALTERNATIVES

	OSIRIS		SILOE	ORPHEE	RHF
Fuel type	UO_2 (Caramel)	U_3Si_2	UAl	UAl	UAl
Enrichment (%)	7,5	19,75	93 ⇒ 90	93	93
Number of F.E. burnt/year	55	70	54	20	6
Number of F.E. accumulated	813	20	339	144	18
Amount of U_t(kg)	6504	50	60,9	87,55	87,5
Location of the F.E.	PEGASE POOL + Reactor site	Reactor site	PEGASE POOL + Reactor site	PEGASE POOL + Reactor site	PEGASE POOL + Reactor site
	⇓	⇓	⇓	⇓	⇓
Scenario to evacuate the irradiated fuels		CASCAD			
	Reprocessing (CEA) in 1995-1996	⇓ direct storage	Reprocessing (AEA, COGEMA) or APM after 1997? or US-DOE	Reprocessing (AEA, COGEMA) or APM after 1997? or US-DOE	Reprocessing (AEA, COGEMA) or APM after 1997? or US-DOE

4. REPROCESSING

As shown in Table III for standard spent fuel elements from ORPHEE, it is useful to recover the uranium still 85% enriched in order to make, by mixing with natural uranium, the 19.75% enriched uranium required to fabricate U_3Si_2 for the OSIRIS reactor.

TABLE III. TYPICAL POST IRRADITION COMPOSITION OF ORPHEE, STANDARD ELEMENT

	Initial composition g/FE	Post-irradiation composition g/FE
U_{total}	907	695
^{235}U	844 (93%)	590 (85%)
^{236}U		47,5
Pu		1,9
^{237}U		1,2

In the uranium coming from reprocessing we find not only ^{235}U but also ^{236}U and other components which may be source of problems. The most important concern is due to the ^{232}U content in the reprocessed uranium which, by radioactive decay, yields ^{212}Bi and ^{208}Tl, both high energy gamma emitters which raise specific problems of radiation protection. The evaluation of these problems has to be made for each reactor, depending on the reactor type and the maximum batch, the ^{232}U content can be 2 ppb or 15 ppb.

If the ^{232}U content is higher than (10-20 ppb) it will probably be impossible to use such uranium in the fabrication plant. From the radiological point of view, CERCA is going to determine the maximum admissible ^{232}U content allowed in the workshops, probably around 12 ppb.

STATUS OF THE BACK END OF THE FUEL CYCLE FOR RESEARCH REACTORS IN GERMANY

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Abstract

The spent fuel situation at research reactors in Germany is outlined together with legal restrictions imposed by the German Atomic Energy Act. Possible solutions including reprocessing in the United Kingdom or France, return of US origin fuel and a home-grown German solution are discussed.

1 Spent fuel situation

The back end of the fuel cycle is of imminent importance for all operating research reactors within the Federal Republic of Germany. This is not only due to the decreasing lack of spare spent fuel element storage positions since the US has offset the acceptance of spent fuel for reprocessing. The stringent needs for solving the problems can be understood when looking at the unique legal restrictions. The special situation has caused an intervention of the licensing authorities for the Triga-Heidelberg reactor to stop the insertion of any fresh fuel element before knowing where to ship the spent fuel elements.

At present there are three Triga, 5 light water moderated and cooled MTR type and one heavy water moderated and cooled research reactors in operation. The research reactor RFR of USSR design and using 10 % and 36 % fuel of USSR origin is without an operating license. But the spent fuel elements from more than 25 a of operation of the RFR is still there from the beginning as the USSR was never accepting spent fuel for reprocessing. All other research reactors besides the Trigas and the RFR have shipped over the years fuel to the US for reprocessing, as these reactors and the Trigas are using fuel of US origin with enrichments between 19,75 % (LEU) and 93 % (HEU).

As the return of fuel to the US is still unsure all MTR type reactors have signed contracts with AEA Dounreay for reprocessing spent fuel elements within the present (last?) campaign. Shipments must arrive at Dounreay not later than the 31.10.1993. After reprocessing the reprocessed uranium has to be taken back and the waste (in cemented form with agreed specifications) is returned to Germany. Some details are given in table 1.

2 Legal restrictions

In an ordinance coming up from the German Atomic Energy Act there is explicitely written that the operator who wants to increase his spent fuel storage capacity by more than 10 % has to pass a public hearing procedure which is normally followed by detailed discussion of the overall safety features of the facility and by court proceedings. This procedure will set up the chance for really increasing the spent fuel storage capacity to a low

Table 1

Reactor	Power	Spent fuel status	remarks
Triga, Mainz	100 kW	no operational problems	
Triga, Hannover	250 kW	no operational problems	1996 shutdown?
Triga, Heidelberg	250 kW	no new fuel elements	operation til 1996?
FMRB - PTB	1 MW	at AEA	< 2000 shutdown?
FRM - Munich	4 MW	at AEA	backfitting
FRG-1 - GKSS	5 MW	ca. 300 spent fuel elements	
FRG-2 - GKSS	15 MW	contract negotiation with AEA	1993 shutdown?
BER-2 - HMI	10 MW	contract with AEA capacity for 4 a limitation in the license	
FRJ-2 - KFA	23 MW	~ 80 % at AEA	under repair

degree. Beside this it is within the decision of the state (not primarily the federal) licensing authority to have the above mentioned procedure even in cases where the desired expansion of the storage capacity is lower than 10 %.

Since many years it was agreed for power reactors that they have to show actually (a rolling procedure) to the licensing authority what are their real (contracted) possibilities for a six years period in advance for the spent fuel storage or removal of the spent fuel elements from the plant. They have to have enough unfilled storage capacity at the plant and/or contracts for interim storage and/or contracts for reprocessing. Options for contracts are not accepted by the licensing authorities.

This year a new ordinance is under preparation by the BMU (Ministry for environmental protection) which demands such a procedure for research reactors, too. This new ordinance (Reststoffverordnung) shall become effective at the end of 1993.

The practical impossibility to increase the storage capacity and the threatening coming up from this new ordinance forcing the German reactor operators to look at all possible solutions to get rid off their spent fuel elements and to ask the BMFT (the Ministry of Science and Technology which is funding most of the operation of the research reactors) to sponsor the development of an own German solution.

3. To get rid off the fuel

a) Cogema

Cogema offered reprocessing services in 91. Many German research reactor operators asked for an offer midth 92. End of 92 all are getting a letter that Cogema don´t intend to start reprocessing activities for foreign customers.

b) AEA

German research reactors where taking the chance to send fuel to Dounreay for reprocessing in 1992 and 1993. But this way causes new problems. The operators must take the waste back after ≥ 5 a and they must know where to store it. The operators must know, too, what to do with the reprocessed uranium.

c) US-DOE

The US option has many steps

- Remember that there was no acceptance of spent fuel from non US facilities since 1989 (except Taiwan)

- Renewing of the US-policy took time. A FONSI was published in 91 and DOE was asking for comments. The only reactor operator sending comments to US-DOE has been GKSS. Many negative comments have been received by DOE so that DOE stops any further activities at that time.

- Beginning of 92 the Edlow company together with lawyers starts a campaign to bring the difficulty with the spent fuel of non US research reactors back on to the table. It was clear very soon that there are many departments and organizations within the US which are pro the renewing of the Off-Site Fuel Policy (like DOS, NRC, ACDA, Congressmen). Many actions were taken. Only DOE was (is?) making in opposition. There are promisses to renew the policy and an EA will be published end of 1993? First shipments may arrive in the US midth of 1994.

- Edlow and the lawyer firm is being payed by ca. 15 companys (3 are from Germany: GKSS, HMI, PTB).

- The renewal of the Off-Site Fuels Policy will probably be limited to fuel from US origin and especially for LEU fuel only for 10 years.

d) German solution

The future for solving these problems seems to be to have a national solution. Therefore within the FRG a storage concept is being developed. See the attached description from the company GNB. The principal work is being funded by the BMFT (Federal Ministry for Science and Technology). Each operator has to buy the necessary number of casks for the interim storage by himself. At present these casks shall be stored at the Ahaus storage plant (HTR, LWR and research reactor spent fueve). The application for the license for the storage was made in March 1993. The application may be granted in 97/98. Beside this a cask license and a transport license is necessary, too. Operators should pay beginning from 94 for the reservation of the storage place. It is clear, that at present no one is able to forsee the development of this licensing procedure so that no credit can be taken from these activities when discussing the demands coming up from the new ordinance. Beside this the contract negotiations between research reactor operators and the company BZA which will store the spent fuel at Ahaus show that there is a severe problem. Within the license of Ahaus there will be the demand that the operators has to take back their spent fuel elements after the operation license of Ahaus expired after 40 a. Under these conditions it can make no sense for the operator to store fuel at the Ahaus plant. To solve this legal problem discussions are going on between BZA, BMFT, BMU and the research reactor operators which have to come to a practicable result in the near future.

Summary

At present we need the US solution for the next time to have the reactors running and to overcome the pressure from the legal demands. If at the end of this decade there will be a national solution (today no one is able to take credit from this as it is only an option and not a timely guaranteed way) all German research reactor operators will store their fuel at Ahaus for up to 40 a. But remember interim storage is no solution for the future. Therefore studies will go on for the final disposal for power reactor fuel and research reactor fuel. A prototype plant for preparing LWR fuel for the final disposal is under licensing at Gorleben. At present there is no solution for the final disposal but there are strong efforts within Germany that direct disposal will be the solution for the future instead of reprocessing.

Appendix 1

Nine research reactors with a power ≥ 100 kW are currently being operated in the Federal Republic of Germany. These comprise three Triga-type reactors (power 100 kW to 250 kW), five swimming pool reactors (power 1 MW to 15 MW) and one DIDO tank reactor (power 23 MW). The German research reactors are used for basic research by neutron scattering in the field of solid state research, neutron metrology, for the production of isotopes and for neutron activation analysis for medicine and biology, for investigating the influence of radiation on materials and for nuclear fuel behaviour. It will be vital to continue current investigations in the future. Further operation of the German research reactors is therefore indispensable. Safe, regular disposal of the irradiated fuel elements arising now and in future operation is of primary importance. Furthermore, there are several plants with considerable quantities of spent fuel, the safe disposal of which is a matter of urgency. These include above all the VKTA facilities in Rossendorf and also the Triga reactors, where disposal will only be necessary upon decommissioning.

The German disposal concept initially envisages the long-term (40 - 50 years) dry interim storage of fuel elements in special containers in a central German interim store with subsequent direct final disposal without reprocessing of the irradiated fuel.

Two transmissions are served onto the upper region of the cask body to facilitate handling of the cask at the reactor stations and in the interim store.

The cask is equipped with a shock absorber system for transportation on public routes protecting it against unacceptably high stresses in the case of a possible accident during transportation.

For storage purposes, the interior of the cask is dried out to a water content of less than 10 g H_2O/m^3 and filled with inert gas (helium). This prevents both corrosion of the stored fuel elements as well as of the metal seals used at the lids.

The following table gives a survey of the essential Castor MTR 2 data.

The dimensions and weight of the CASTOR MTR 2 are designed in such a way that it can be loaded and handled at most research reactors.

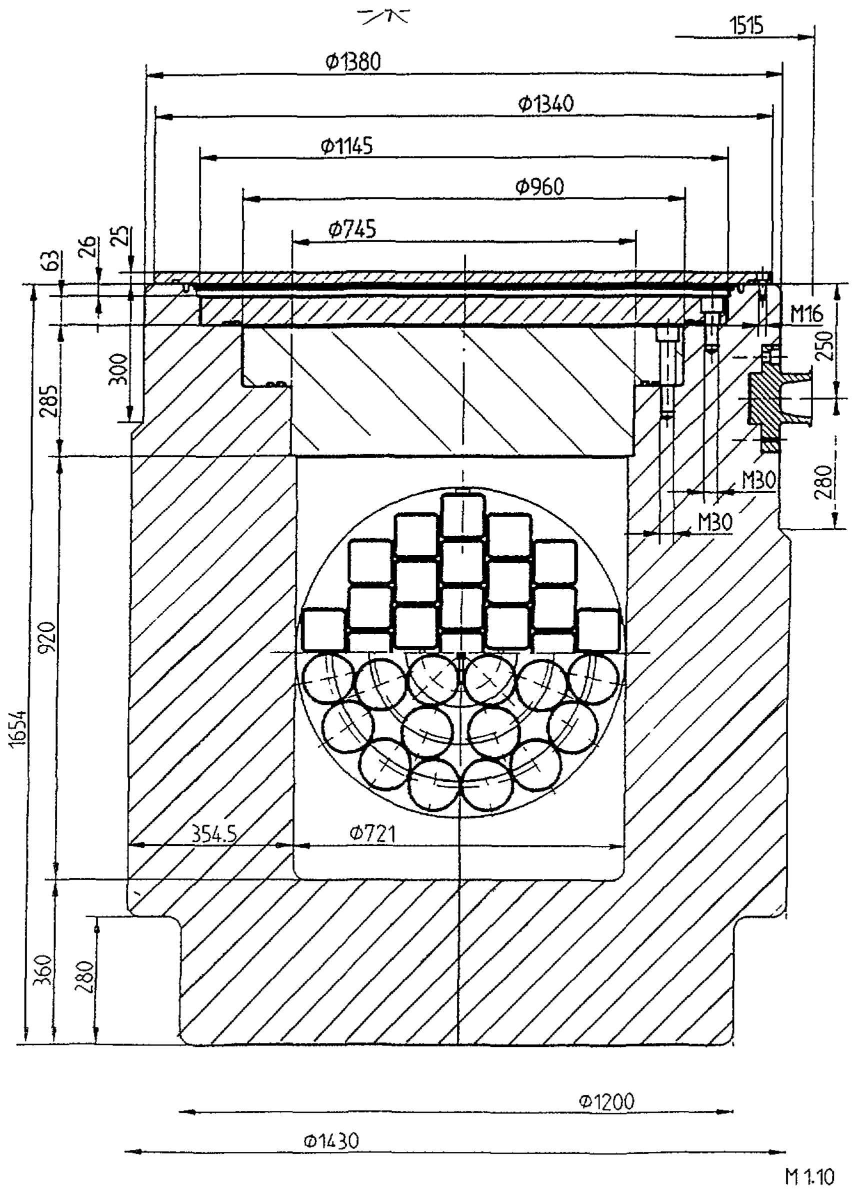

FIG. 1. CASTOR MTR 2.

The Noell Freiburg company is currently developing a mobile unloading facility to be carried out at reactor stations where it is not possible to directly handle the 16 t cask.

External diameter (without shock absorbers)	1430 mm
Overall height (without shock absorbers)	1679 mm
Diameter of inner compartment	721 mm
Height of inner compartment	920 mm
Load	
- box-type MTR fuel elements	33/(28)
- tubular MTR fuel elements	28
- WWR-M2 fuel elements (Soviet design)	49
- EK-10 fuel elements (Soviet design)	42/(28)
- TRIGA fuel elements	78
Cask weight (loaded, without shock absorbers)	16000 kg

The interim storage concept planned on the basis of CASTOR MTR 2 cask only requires handling of the fuel elements during loading at the reactor and only one shipment of the loaded cask from the reactor to the interim store. For these reasons, and due to the large number of fuel elements which can be stored in one cask, this concept ensures a safe and economic disposal of spent fuel elements from research reactors in the long term.

The described CASTOR MTR 2 cask design has been approved by the Federal Ministry for Research and Technology (BMFT) and the operators of the German research reactors. Licensing documents pursuant to transport legislation and the Atomic Energy Act are currently being compiled. The first licensed casks of the CASTOR MTR 2 type probably will be available at the end of 1997.

STORAGE EXPERIENCE IN HUNGARY WITH FUEL FROM RESEARCH REACTORS

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Abstract

In Hungary several critical assemblies, a training reactor and a research reactor have been in operation. The fuel used in the research and training reactors are of Soviet origin. Though spent fuel storage experience is fairly good, medium and long term storage solutions are needed.

REACTORS

In the KFKI (the former Central Research Institute for Physics), six zero power critical assemblies were in operation subsequently between 1959 and 1990.

ZR-1 was a critical experiment using EK-10 fuel elements in the late 50's. The same type of fuel was used in the first period in the research reactor and is being used in the training reactor. The fuel elements used in ZR-1 were later irradiated in the research reactor and they are stored in the Spent Fuel Storage Facility No. 1 of the KFKI.

ZR-2 was a critical facility used for reactor physics experiments in the 60's. The fuel rods were fabricated in Budapest using U_3O_8 fuel. The dismantled fuel is stored in the Central Isotope Storage Facility of the KFKI. Its storage does not require any special treatment or precaution, because its burn-up is negligible, so it can be treated as the fresh fuel.

ZR-3 was a critical experiment using VVR-SM fuel elements in 1967. The same type of fuel has been used afterwards in the research reactor. The fuel elements used in ZR-3 were later irradiated in the research reactor and they are stored in the Spent Fuel Storage Facility No. 1 of the KFKI.

ZR-4 was a small critical facility used for reactor kinetics experiments and also as a thermal column in the 70's. The homogeneous uranium-polyethylene fuel blocks were fabricated in Budapest. The fuel is stored in the Central Isotope Storage Facility of the KFKI. Its storage does not require any special treatment or precaution, because its burn-up is negligible, so it can be treated as the fresh fuel.

ZR-5 was a critical facility using EK-10 fuel elements in the late 60's. It was a mock-up of the training reactor. The fuel elements used in ZR-5 were later utilized in the training reactor and they are still in the reactor pool.

ZR-6 was a critical facility used for reactor physics experiments in the 70's and 80's. The experiments were performed by an international team of VVER user countries. The fuel rods were fabricated in the Soviet Union using UO_2 fuel, similar to the VVER fuel. The fuel rods are stored in the KFKI Atomic Energy Research Institute. Their storage does not require any special treatment or precaution, because their burn-up is negligible, so they can be treated as the fresh fuel.

The training reactor belongs to the Budapest Technical University. It is in use for more than 20 years. The reactor is fuelled by EK-10 fuel elements. Due to the low power (100 kW) all the fuel elements may remain in the reactor pool for a long period. Storage racks are provided for medium term storage in the reactor pool.

The research reactor belonged to the former KFKI, now it belongs to the KFKI Atomic Energy Research Institute. Four phases can be distinguished since the start-up of the reactor in 1959:
- 1959-1967: 2 MW power, EK-10 fuel, 82 elements were burnt
- 1967-1986: 5 MW power, VVR-SM fuel, 780 elements were burnt
- 1986-1992: major reconstruction
- 1993- : 10 MW power, VVR-SM fuel.

FUEL ELEMENTS

Two types of spent fuel elements are stored in Hungary: EK-10 and VVR-SM.

EK-10 is a Soviet designed and fabricated 10% enriched uranium fuel. The initial content is 128 g ^{235}U per element. The cladding material is aluminium, cladding thickness is 1.5 mm. The average burn-up is 23%. The remanent heat emission is about 1 - 1.5 W/element. This type of fuel elements are stored in the Spent Fuel Storage Facility No. 1 of the KFKI from the early 60's (82 elements).

VVR-SM is also a Soviet designed and fabricated fuel element. Uranium enrichment is 36%. The initial content is 38.9 g ^{235}U per element. The cladding material is aluminium, cladding thickness is 0.9 mm. The average burn-up is 50%. The remanent heat emission is about 5 - 20 W/element. This type of fuel elements is stored in the Spent Fuel Storage Facility No. 1 of the KFKI from the late 60's (780 elements), the last ones were sent to the facility in 1986.

STORAGE FACILITIES

In Hungary three storage facilities are being used.

The Spent Fuel Storage Facility No. 1 of the KFKI was built in the early 60's. It is an underground stainless steel pool. The VVRSM type fuel elements are stored in the pool in aluminium tubes, the EK-10 fuel elements are placed directly into the pool. The pool is about 100 m from the reactor building. Fuel elements are transported in special small containers into the facility. Water pH and conductivity are measured once in a month. In case the conductivity measurement shows any anomaly a filtering is possible using a mobil ion change filter. Water level is monitored in the control room.

The Spent Fuel Storage Facility No. 2 of the KFKI was built during the reconstruction period and was put into operation in 1993. The new stainless steel pool is situated in the reactor hall and is directly connected to the reactor tank. The lattice pitch in the storage facility is the same as is in the core. The storage facility contains B_4C rods to ensure the subcriticality of the high density storage arrangement. No spent fuel is loaded into the facility up to now.

The storage rack of the Budapest Technical University training reactor consists of storage channels located in the reactor block. Since no regular refuelling has to be reckoned with, fuel element manipulations are required only in special cases. A special container serves for removal of irradiated fuel elements from the core. A separate hermetically sealable container serves to store damaged elements within the storage channel.

STORAGE EXPERIENCE

The storage experience with EK-10 and VVR-SM fuel elements is fairly good. No fuel failure has been detected in the storage facilities.

PROBLEMS, POSSIBLE SOLUTIONS

As far as Spent Fuel Storage Facility No. 1 of the KFKI is concerned, a great amount of empty places exist in the facility. Nevertheless, there are no intentions to load there further fuel elements. The facility should be emptied. The main reason is that the facility is not designed for a long term use and even having a good experience with the stored material a solution should be found.

At the moment two solutions are under preparation. The first one is based on transporting back the fuel elements to Russia. It is technically feasible and even designed, though the licence of the corresponding Russian containers is expired. The transport should be organized together with the transports of the spent fuel elements from the Paks NPP. As a consequence of the collapse of the Soviet Union and the new Russian regulations the future transports from the Paks NPP are uncertain. The second solution would be to store the fuel elements in the interim dry storage facility to be built (with a high probability) at the Paks NPP. Thus in both cases the solution is tightly connected with the solution for the Paks NPP spent fuel problems.

Obviously, the solution applied for the fuel stored in the Spent Fuel Storage Facility No. 1 of the KFKI will be also applicable both for the spent fuel to be stored in the Spent Fuel Storage Facility No. 2 of the KFKI and the spent fuel stored in the Storage Rack of the Budapest Technical University.

INTERNATIONAL CO-OPERATION

Since it can be expected that spent fuel from the training and research reactors will be stored in Hungary still for a considerable time (perhaps not only in wet, but also in dry conditions), it is very valuable to learn the experience gained at other sites. Unfortunately, the special alloy used in the Soviet aluminium cladding technology does not permit to draw direct conclusions from the experience with fuel of US origin. The IAEA efforts to share the experience gained at various facilities are very much welcome.

ACKNOWLEDGEMENTS

The material was prepared with the assistance of T. Hargitai (KFKI), G. Pőr (BTU) and I. Vidovszky (KFKI). The author wishes to thank for their kind assistance.

EXPERIENCE WITH UNDERWATER STORAGE OF SPENT FUEL FROM RESEARCH REACTORS IN INDIA
(Abstract)

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Over 30 years of experience in the storage of irradiated fuel from research reactors exists at the Bhabha Atomic Research Centre. Experience with wet storage of metallic fuel is dealt with in this presentation.

Vertical underwater storage of metallic uranium fuel elements, resting on the bottom, results in bowing of the fuel during long periods of storage and breakage of the elements has also been experienced in some cases. If the elements are stored by suspension from the top, this problem is not encountered. In the case of breakage of the fuel elements, release of uranium powder into the bays occurs. For removal of the uranium powder, pool clearing equipment using cyclone separators and filters has been successfully employed.

Aluminium Cans with plugs have been used to package fuel elements with clad defects for underwater storage. Due to the leak-tight design of the Cans, pressure build-up has been observed in the Cans, possibly due to uranium-water reaction. Pressure relief provisions have been made to overcome this problem.

For purification of the fuel storage bay water, ion-exchange units that can be regenerated were in use earlier. This was changed to non-regeneratable, cartridge-type resins, to avoid liquid waste regenerate and to allow easier disposal of the used resins. Significant reductions in radiation exposure of plant personnel has also been achieved using this methodology.

Experience with stainless steel pool liners has been very good. Good control of the water chemistry in the bays has been found to be essential in minimising aluminium clad corrosion. Location of the storage bays above ground level is considered superior to locating them below ground level. Ageing of underwater fuel transfer and handling equipment has given rise to maintenance problems and it is suggested that in the future, these aspects should be taken into account at the design stage of the equipment in question.

PRESENT STATUS OF SPENT FUELS IN JAPANESE RESEARCH REACTORS

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Abstract

Spent fuel storage at research reactors in Japan is described in outline. Reactors KUR, JRR-2 and JRR-4 have enough storage capacity at present but an increase of the storage capacity at JRR-3M and JMTR by re-racking is under consideration. For the long term some consideration is being given to dry storage in casks to prevent fuel cladding corrosion.

1. Introduction

In Japan, there are now eleven research and test reactors in operation as shown in Table 1. The spent fuel issues might cause in some of reactors with relatively high power such as JRR-3M, JMTR and so on near future. The present status and future items relevant to the spent fuels from research and test reactors in Japan are described as follows.

Table 1. Japanese Research Reactors in Operation

Name	Owner	Site	Type and enrichment			Max. Power	Start-up date
JRR-2	JAERI	Tokai	D_2O(CP-5)	U-Al	93%	10 MW	1960. 10
				UAl_x	45%	10 MW	1987. 11
UTR KINKI	Kinki Univ.	Higashi-osaka	H_2O(UTR)	U-Al	90%	1 W	1961. 11
TRIGA-II RIKKYO	Rikkyo Univ.	Yokosuka	H_2O(TRIGA)	U-ZrH	20%	100 kW	1961. 12
TTR-1	Toshiba	Kawasaki	H_2O(pool)	U-Al	20%	100 kW	1962. 3
JRR-3	JAERI	Tokai	D_2O(tank)	NU		10 MW	1963. 9
				UO_2	1.5%		
			H_2O(pool)	UAl_x-Al	20%	20 MW	1990. 3
MITRR MUSASHI	Musashi Inst. Tech.	Kawasaki	H_2O(TRIGA)	U-ZrH	20%	100 kW	1962. 3
KUR	KURRI	Kumatori	H_2O(tank)	U-Al	93%	5 MW	1964. 6
JRR-4	JAERI	Tokai	H_2O(pool)	U-Al	93%	3.5 MW	1965. 1
JMTR	JAERI	Oarai	H_2O(MTR)	U-Al	93%	50 MW	1968. 3
				UAl_x	45%	50 MW	1986. 8
YAYOI	Univ. of Tokyo	Tokai	fast (horizontally movable)	U	93%	2 kW	1971. 4
NSRR	JAERI	Tokai	H_2O(TRIGA)	U-ZrH	20%	300 kW	1975. 6

2. Spent Fuels in Kyoto University Research Reactor Institute(KURRI)

KURRI owns the KUR with 5MW which has been operated since 1964 using HEU fuels. The present status of spent fuels from the KUR are as follows.

Fuel : U-Al Alloy (93%HEU), MTR Type

Storage Facility (Environment): Fuelrack in Pool (Water)

Capacity : No.1 Pool ········· 160 Elements

No.2 Pool ········· 300 Elements

Amount in Hand : 200 Elements (As of Sept., 1993)

The Longest Period of Storage : 21 Years.

Annual Output of SF : 12 Elements

In KUR,there are two storage pools for a spent fuel.

① No. 1 pool connects with KUR through canal. Fuels are stored here for about one year after the removal from the core.

② No. 2 pool locates in separated building from KUR and the cask is necessary for the transportation of spent fuels. Maximum storage capacity may be increased up to 600 elements.

Water chemistry of the pool is controlled and monitored.
Aimed electric conductivity is less than 5μs/cm and daily inspected value shows 0.2 ~0.4μs/cm. The pH has been kept in the level of 6 to 7. The storage facility was constructed based on the aseismic design of B class* with story shearing force coefficient of C_I=0.3 for pool and horizontal seismic coefficient of C_I=0.36 for fuel rack. Water level of the pool is continuously monitored by the sensor installed. Integrity of spent fuels during storage is checked by the sipping test on occasion. At the present, the pool leaves relatively big margin for the storage capacity. The current management for these spent fuels is suitable to maintain their integrity.

3. Spent Fuels in Japan Atomic Energy Research Institute (JAERI)

3.1 Research Reactors (JRR-2, JRR-3M and JRR-4)

Table 2 shows present status of spent fuels in JRR-2, JRR-3M and JRR-4. Water chemistry of these pools is controlled and monitored. Aimed electric conductivity and pH are less than 10μs/cm and 5.0~7.5 respectively. Weekly inspected values shows 0.9 to 1.2 μs/cm and pH 5.8. Water level of these pools is continuously monitored by the sensors installed.

Spent fuels during storage in the pool are weekly checked by radioactivity measurement and nuclide analysis of sampled water. Visual inspection of spent fuels and radioactivity measurement of water purification system are also done in order to check the integrity of spent fuels.

Metallic natural uranium (MNU) fuels are stored in the canister of the Dry Storage Facility (DSF) under inert gas environment.

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* see Appendix.

Table 2 Present status of spent fuels in research reactors of JAERI.

Reactors	Fuels	Storage Facility (Environment)		Capacity (Element)	Amount in Hand (Element) (As of Sept, 1993)	Longest Storage Period (Year) (As of Sept, 1993)	Annual Output (Element)
JRR-2 (10MW)	U-Aℓ Plate Type (93%HEU) UAℓx-Aℓ Plate Type (45%MEU)	Plate	SF Pool (wet)	120	HEU 7 MEU 46	20	20
JRR-3M (20MW)	Metallic Natural Uranium (MNU)	Plate	SF Pool (wet)	130	LEU 106	2	・Zero for MNU and UO_2 ・Plate (LEU) 46
	Uranium Oxide (UO_2)	UO_2	SF No. 1 Pool (wet)	450	407	22	
	UAℓx-Aℓ Plate Type (19.75%LEU)	Plate		200	27	――――	
		Plate	SF No. 2 Pool (wet)	80	From JRR-2 MEU 28 HEU 32	20	
		MNU	DSF (Dry)	600	600	31	
JRR-4 (3.5MW)	U-Aℓ Plate Type (93%HEU)	Plate	SF Pool (wet)	97	21	5	5

The storage facility was constructed based on the aseismic design. Fuelrack, pool and the DSF are designed as B class. The horizontal seismic coefficient C_I is 0.36 for fuelrack and the story shearing force coefficient C_I is 0.3 for pool and the DSF. The current management for these spent fuels is quite enough to maintain their integrity.

As for the JRR-2 and JRR-3M, the followings can be pointed out.

① Sixty elements were transferred from JRR-2 to the No.2 spent fuel storage pool in JRR-3 on November, 1991 because of the overflow of the capacity.

② Spent fuel storage pool in JRR −2 leaves a margin due to the termination of reactor operation at the end of 1996.

③ Spent fuels in the JRR-3 Storage Facility will reach to the licensed storage capacity limit of 410 elements for the plate type fuel at the end of FY 1997 and the increase of fuelrack is now under consideration.

④ Reprocessing is now under investigation as for UO_2 and MNU.

In the JRR-4 the pool has a quite enough space for the storage of spent fuels.

3.2 Japan Material Testing Reactor (JMTR)

Present status of spent fuels in JMTR is summarized in Table 3. Water chemistry of the pool is controlled and monitored aiming at the aimed electric conductivity and pH of less than 2 μs/cm and 5.5～7.0respectively. Observed data are as follows.

① Electric conductivity is continuously monitored and shows 0.05～1.5 μs/cm.

② The pH is checked at ten days interval using sampled water and shows 5.5 to 7.0

Water level of the pool is continuously monitored by the sensor installed. Integrity of spent fuels during storage in the pool are weekly monitored by radioactivity measurement and nuclide analysis of sampled water. Storage facility was constructed based on the aseismic design of B class with C_I= 0.36 for fuelrack and C_I=0.3 for pool. The current management for these spent fuels is quite good enough to maintain their integrity.

Since storage capacity of the pool in the JMTR might be exceeded by the production of spent fuels within a few years, the increase of fuelrack is now under way.

4. Spent Fuels in Other Reactors

Present status of spent fuels in the MITRR (100kw) are as follows.

Amount of Spent Fuels in Hand : 65 Elements (TRIGA)

Interim dry storage is now under investigation

At present,the NSRR,TRIGA-II Rikkyo,TTR-1,Yayoi and UTR Kinki have no spent fuels.

Table 3 Present status of spent fuels in JMTR.

Reactor	Fuels	Type	Storage Facility (Environment)	Capacity (Element)	Amount in Hand (Element) (As of Sept, 1993)	Longest Storage Period (Year) (As of Sept, 1993)	Annual Output (Element)
JMTR (50MW)	U-Aℓ (93% HEU)	Plate	SF Pool (Wet)	1115	80	8	~80
	UAℓx-Aℓ (45%MEU)				603	7	
	U_3Si_2-Aℓ (<20%LEU)				0		

5. Conclusions

From the results described above, it can be pointed out that KUR, JRR-2 and JRR-4 could have a relatively enough space for the storage of spent fuels.
As for spent fuels storage in JRR-3M and JMTR, the increase of fuelrack is now under consideration or under way because of the excess of their capacity near future.
This implementation leaves a margin more than several years. The current management for spent fuels is suitable to maintain their integrity. Interim dry storage in a cask might be future issue from the view point of prevention of corrosion for longer storage.

Appendix

Seismic classification in the Guideline for Aseismic Design of Nuclear Reactor Facilities in Japan (after M. Watabe)[1]

Seismic Classification	Definition	Example
Class As	Facilities extremely essential to plant safety among Class A items	Reactor containment, Reactor coolant pressure boundaries, Core shut-down system, etc.
Class A	Facilities important to plant safety or related to radioactive material	Reactor auxiliary building, Emergency core cooling system, Emergency off-gas system, etc.
Class B	Same as Class A but whose rupture might lead to less serious consequences	Turbine Bldg. (BWR), Rad-waste treatment system, etc.
Class C	Facilities not classified as A or B , and the same degree of safety as ordinary industrial facilities	Turbine Bldg. (PWR), Turbine Generator, etc.

1) M. Watabe, " Recent Development in Seismic Design Criteria and Practice in Japan ".

THE PROBLEMS OF TREATMENT OF IRRADIATED FUEL AT RUSSIAN RESEARCH REACTORS

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Abstract

This report describes the problems of safety during the storage and transportation of spent fuel from Russian research reactors. Many research reactors continue to operate in Russia at THE present time. They use many different types of fuel elements and assemblies. The questions of safety during storage and transportation of spent fuel from research reactors is considered in Russian documents on the safety of research reactors. The main features of these documents are described in this report. Three stages of the storage and transportation of spent fuel are discussed: the temporary storage in the pool or in the vessel; the storage in the repository on the territory of the institutes; and the transportation of the fuel to the reprocessing plant. The future plans provide for the solution of the problems of transportation and reprocessing of all types of fuel assemblies which are used in Russian research reactors and experimental facilities. Also, there is an intention to continue the work on reducing the enrichment of the fuel for research reactors and to change the fuel composition from uranium dioxide in an aluminium matrix to uranium silicide in aluminium matrix.

1. INTRODUCTION

From 1946 to the present a large number of research reactors have been constructed in Russia. Twenty one of them continue to operate at the present time. Moreover there are a certain number of prototype reactors, reactors for special purposes and zero power reactors (or critical assemblies).

The total power of all Russian research reactors is about 400 MW. The value of cumulative reactor years for all of them is about 600 (see Table I).

During their operational lives the Russian research reactors accumulated and continue to accumulate a large number of irradiated fuel elements and assemblies of different types, including experimental fuel elements and assemblies.

Naturally, these circumstances required the development of a system of treatment of irradiated fuel that could guarantee the nuclear and radiation safety of the storage and transportation of this fuel. Such system has been created and is continually updated.

The important peculiarity of Russian research reactors is the large variety of types of fuel assemblies that are used in different reactors. At the present time, more than ten types of fuel assemblies are in use. The fuel elements in these assemblies are of tube and rod types, the enrichment of the uranium is 10, 36 and 90%, the height of the active part is from 35cm to about 2m (see some examples of Russian fuel assemblies in Figs 1-4). Different types of fuel materials are in current use, such as uranium-aluminum alloy, a dispersion of uranium oxide in aluminum matrix, etc.

Moreover many types of experimental fuel elements are tested in experimental loops and rigs. It is clear that this situation means that the safety analyses of storage and transportation of spent fuel elements requires a great deal of effort.

2. GENERAL

From Table I we see that the main type of Russian research reactor is a water-water reactor (pool and tank type). For this reason, this report discusses mainly the questions of the safety of the storage of the spent fuel at research reactors of this type, but it should be noted that the general questions of safety are the same for fuel of reactors of any type.

TABLE I. RUSSIAN RESEARCH REACTORS IN OPERATION, UNDER CONSTRUCTION OR PLANNED FOR CONSTRUCTION

	Facility name	Date of criticality	Reactor type	Power, MW	Status	Age, years	Fuel type
1	F-1	1946	graphite	0.024	Oper.	47	
2.	IR-8	1957	pool	8.0	Oper.	36	UO + Al
3.	MR	1963	channel	40.0	Oper.	30	UO + Al
4.	IIN-3M	1972	homo.(l) pulse	$1*10^4$ (pulse)	Oper.	21	Solution
5.	ARGUS	1981	homo.(l)	0.05	Oper.	12	Solution
6.	IR-50	1961	pool	0.05	Oper.	32	UO + Mg
7.	IRT - MIFI	1967	pool	2.5	Oper.	26	UO + Al
8.	AM	1954	graphite	10.0	Oper.	39	UO + Mg
9.	BR-10	1958	fast	8.0	Oper.	35	PuO
10.	WWR-TS	1964	pool	10.0	Oper.	29	UO + Al
11.	SM-3	1961	tank	100.0	Oper.	32	UO + Cu
12.	MIR-M1	1966	channel	100.0	Oper.	27	UO + Al
13.	BOR-60	1969	fast	60.0	Oper.	24	UO +PuO
14.	RBT-6	1975	pool	6.0	Oper.	18	UO + Cu
15.	RBT-10/1	1983	pool	10.0	Oper.	10	UO + Cu
16.	RBT-10/2	1984	pool	10.0	Oper.	9	UO + Cu
17.	IVV-2M	1966	pool	15.0	Oper.	27	UO + Al
18.	WWR-M	1959	pool	18.0	Oper.	34	UO + Al
19.	IRT-T	1967	pool	6.0	Oper.	26	UO + Al
20.	RG-1M	1970	pool	0.1	Oper.	23	UO + Mg
21.	IBR-2	1977	pulse	4.0 (aver.)	Oper.	16	Pu
22.	SVV-1		pool	0.5	Constr.		UO + Mg
23.	PIX		tank	100.0	Constr.		UO + Cu
24.	SPHINX		channel	200.0	In design		UO + Al
25.	MPR		pool	100.0	In design		UO + Al

At first, it is important to note that the requirements of safety during the storage and transportation of spent fuel assemblies are described in many special Russian documents on the safety of research reactors.

In Russia there is a system of different levels of documents on safety of research reactors. The top level document is: "General Requirements for Providing Safety at Research Reactors". This document includes the main safety principles for the siting, design, commissioning, operation, modification and decommissioning of research reactors. It corresponds to such Agency documents as "Safety Standards on Design and Operation" (35-S1 & 35-S2).

The documents of the next level that describe the definite requirements for the safety systems of reactor are very numerous. They consider all of the general questions of safety during the storage and transportation of spent fuel elements.

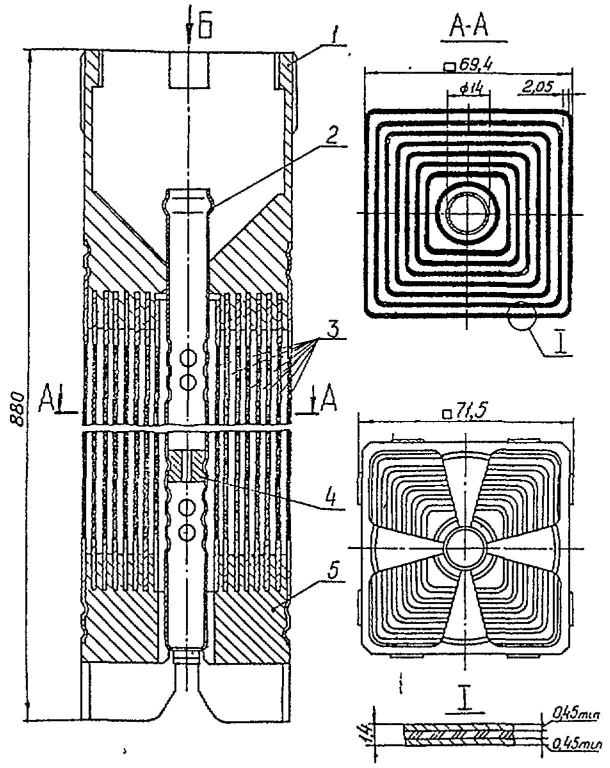

FIG. 1. Fuel assembly of the IRT-3M type; 1 – cap; 2 – displacer; 3 – fuel elements; 4 – throttle; 5 – ending part.

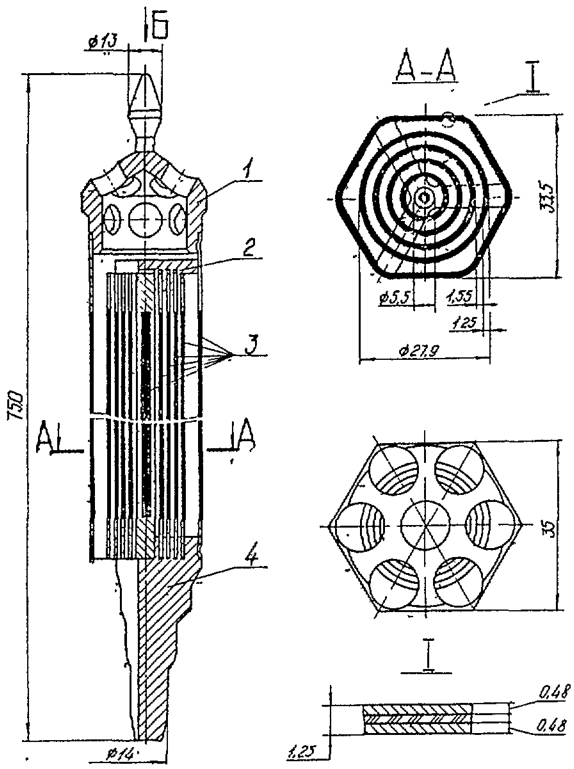

FIG. 2. Fuel assembly of the WWR-M reactor; 1 – cap; 2 – upper grid; 3 – fuel elements; 4 – ending part.

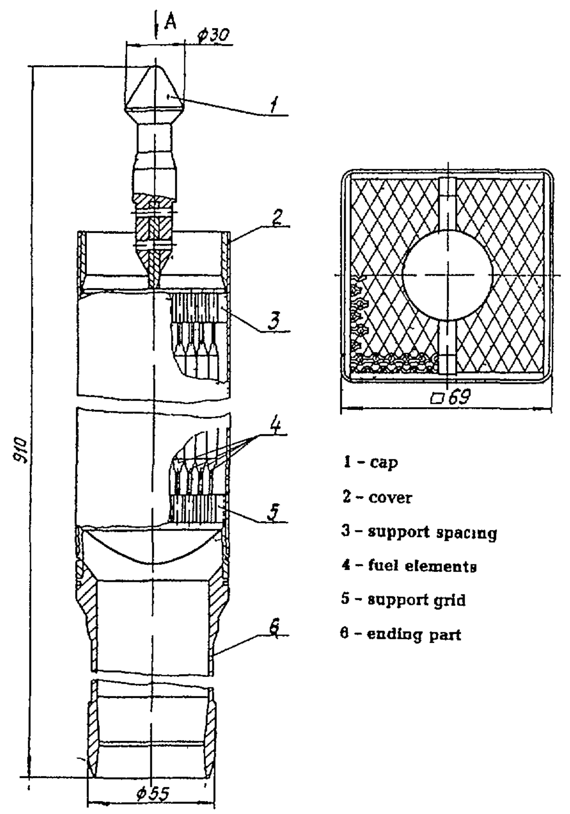

FIG. 3. Fuel assembly of the CM-2 reactor.

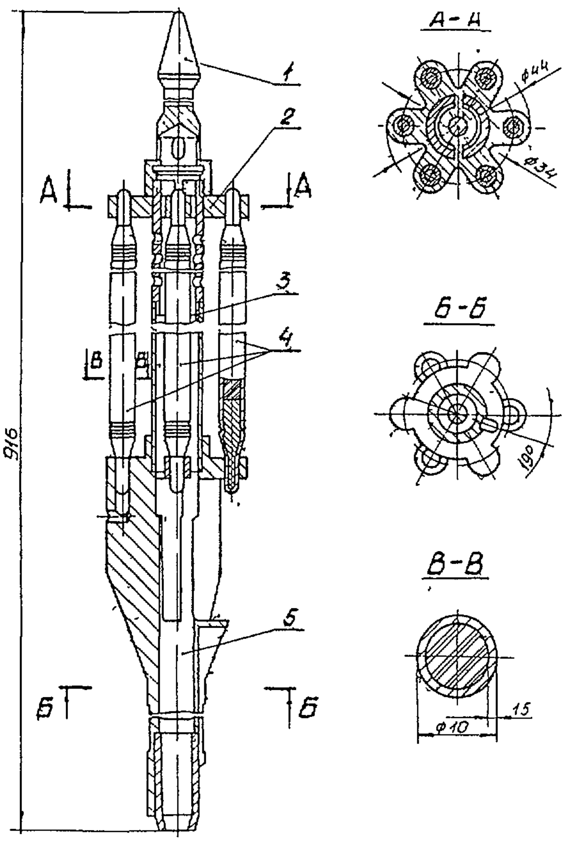

FIG. 4. Fuel assembly of the RG-1M reactor; 1 – cap, 2 – upper grid, 3 – tube, 4 – fuel elements; 5 – ending part

The following main safety requirements shall be provided at all stages of the movement of spent fuel from the core to the reprocessing plant:

- provision of sufficient subcriticality in the storage of the spent fuel elements and assemblies for nuclear safety;
- adequate heat removal from the spent fuel elements for the prevention of damage of the cladding of the fuel elements;
- provision of an adequate coolant chemistry for the minimisation of cladding corrosion;
- physical protection of the nuclear fuel;
- exact registration of the quantity of nuclear materials at all stages of the storage and transportation.

3. SAFETY DURING TEMPORARY STORAGE OF SPENT FUEL

After unloading the fuel assemblies from the active core they shall be transported to the temporary storage.

The questions of nuclear safety of the temporary storage of fuel assemblies are considered in the special document: "Nuclear Safety Regulations for Research Reactors" [1], which mainly addresses the questions of nuclear safety during commissioning and operation of research reactors.

Rates and order of transportation and storage of the spent fuel elements shall be specified in the reactor design. A special document that discusses the problems of safety shall, be included in the design of the reactor. This document is entitled: "The Technical Basis for the Safety of the Reactor" and its contents are analogous to those of the "Safety Analysis Report". The Technical Basis for the Safety of the Reactor such as its design shall be approved by the regulatory body.

The assemblies of spent fuel elements must be stored in special storage racks. These storage racks are located in the pool or in the vessel of reactor. Transfer of the elements from the active core to the storage racks is accomplished by moving the fuel assemblies through the water that provides adequate shielding. When the storage racks become full the spent fuel assemblies must be loaded into the temporary "wet" storage. Usually the capacity of the temporary storage is relatively big. For example at the reactor IR-8 the capacity of the temporary storage is 120 fuel assemblies. This capacity corresponds to 8-10 years of operation of the reactor.

The storage of spent fuel assemblies shall be such as to prevent inadvertent criticality and to provide an adequate cooling of fuel elements. The most important safety parameter during the storage of the spent fuel assemblies, i.e. the arrangement of the fuel assemblies, is calculated by the operational organization. According to the regulations [1] the subcriticality of spent fuel storage shall not be less than 0.05 in all accident situations. These calculations shall be verified by the special division of the Institute of Physics and Power Engineering at Obninsk and after that by the regulatory body.

To avoid corrosion of the cladding of the fuel elements the water chemistry in the pool shall be the same as in the primary circuit of the reactor. For example, for fuel elements with aluminium cladding the value of pH shall be from 5.5 to 6.5 and the specific conductivity shall be less than 1.5 μS/cm. Clearly, these parameters are also defined in special standards.

Conservative assumptions are used to provide an adequate safety margin. In particular, the calculations of criticality are carried out assuming that all fuel elements in storage are fresh, but in the calculations of the radiological consequences it is assumed that all fuel elements have the maximum possible burnup.

The information about radiological situation in the storage and the level of the coolant in it shall be communicated to the control room. In the case of a deviation from the acceptable limits a warning signal shall appear on the control panel.

A special problem is the storage of defective fuel elements and assemblies, since there may be a possibility that fission products are released into the pool. These fuel assemblies must be install in special boxes to isolation them from the pool. However, it is emphasized that occurances of failured fuel elements are very rare and at the majority of Russian research reactors such failures have not happened more than one or twice during the whole history of the reactor.

Information about all transport operations with spent fuel assemblies shall be recorded in a special log. These include the times of loading and unloading of the fuel assemblies from the core, thé burnup of the fuel and the position of the fuel assembly in the storage. The order of the treatment of the fuel is described in the instructions for the safe transportation and storage of fresh and spent fuel, a document that is available at every research reactor.

4. SAFETY OF REPOSITORIES FOR THE SPENT FUEL

Once the decay heating has reduced to a level when the storage of the spent fuel assemblies is possible without cooling by pool water they can be transported to the repositories. Such a repository is avaiable at all institutes that have several research reactors.

The questions of nuclear safety during the transportation of the spent fuel assemblies to the repositories and storage of spent fuel assemblies in them is defined in the special document entitled: "Nuclear Safety Regulations for Transportation and Storage of Dangerous Fissionable Materials"[2]. According to these regulations, all repositories shall be divided into 3 classes depending on the possibility of the repository overflowing in the case of the flooding, failures of equipment in the neighbouring rooms and personnel errors. In repositories of the first class overflowing is impossible even in the case of the flooding. It is obvious that these repositories are dry.

According to the regulations [2], the subcriticality of spent fuel in repositories shall not be less than 0.05 for a full repository and in all accident situations. The calculations of criticality must be carried out by the operational organization, then they shall be verified by special division of the Institute of Physics and Power Engineering at Obninsk and after that by the regulatory body. This procedure is the same as for temporary storage of spent fuel assemblies.

The problem of corrosion of the cladding of fuel elements can arise during the storage of spent fuel assemblies in "wet" repositories but it is important to emphasize that the process of corrosion proceeds very slowly. However, the long-term storage of spent fuel assemblies in dry repositories is preferable.

5. SAFETY DURING THE TRANSPORTATION OF SPENT FUEL TO THE REPROCESSING PLANT

The decay time of spent fuel assemblies varies from one fuel assembly to another and may reach four years for fuel assemblies with very high specific power.

After the decay heating in fuel elements has reduced to a low level they can be transported to the reprocessing plant. The questions of nuclear safety during the transportation of the fuel assemblies are defined in special document: "Principal Safety and Physical Protection Regulations for the Transportation of Nuclear Materials" [3].

The transportation of spent fuel elements is carried out in special dry containers [4]. It is impossible to transport spent fuel assemblies until the operational organization have prepared the necessary documents. These documents shall consider the questions of nuclear and radiation safety, they shall include the calculations of criticality and adequate cooling in normal conditions and in all accident situations.

To receive permission for the transportation of spent fuel assemblies, an operating organization must prepare the necessary documents that must then be approved by the regulatory body, representatives of the security service and local authorities of those territories over which the transportation is intended.

The procedure to receive permission is not very easy. Moreover the permission can be received only if the reprocessing plant has adequate technology for reprocessing of the spent fuel in question. Taking into account the large number of fuel compositions that are used in Russian research reactors and their experimental facilities this problem is very complicated and for this reason many types of fuel elements stay in the repositories for many years.

Special standards describe the requirements for the vehicle and the transport container. New transport containers must be tested and approved by the regulatory body.

During last few years a new cask (cask-19) for the transport of research and experimental reactor spent fuel assemblies has been developed [4]. It is a thick-walled vessel tightly sealed with a heavy cap. The cask allows the shipment of spent fuel assemblies of different research reactors with various fuel types of different cross-sectional shapes and with a total decay heat of up to 360 W. The cask design assures nuclear safety, biological shielding, heat removal from fuel assemblies, hermetic sealing and strength in normal and emergency conditions of transportation.

6. FUTURE PLANS

The most important difficulties with the treatment of irradiated fuel from Russian research reactors are the large variety of the fuel elements and assemblies used in the research reactors and their experimental facilities. For this reason in the near future we intend to finish the development of the safety standards and include in the final document standards for all of the types of the fuel elements and assemblies used in the past and at present. On the other hand, we want to use in the future not more than one or two fuel compositions in the fuel elements of research reactors. This will make it easier to solve the problems of reprocessing of spent fuel.

We also propose to convert the composition used in the fuel elements of Russian research reactors for another reason. This is in connection with the problem of the reduction of enrichment of uranium in research reactors, i.e. nonproliferation. At the present time the main type of the fuel composition in our research reactors is uranium oxide in an aluminium matrix. With this composition it is possible to reduce the enrichment of uranium from 90% to 36%, but to reduce the enrichment to less than 20% it is necessary to use a new fuel composition, e.g. uranium silicide in aluminium matrix. The development of this composition in Russia is now in progress and we hope in the near future to begin using fuel elements with such a composition. Clearly, this development will create new problems of safety during the storage, transportation and reprocessing of spent fuel elements. However, taking into account the fact that the matrix (aluminium) is the same in both cases, the solution of these problems should be not too difficult.

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FUEL STORAGE HANDLING AT THE R2 REACTOR AT STUDSVIK

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Abstract

The Swedish 50 MW(th) high flux reactor R2 has been in operation since 1960. The reactor has during these 33 years used more than 1300 fuel assemblies. The HEU (high enriched uranium) fuel is gradually replaced by LEU (low enriched uranium) fuel. From the operating point of view the same rules on transport and storage are valid for both HEU and LEU assemblies. Before, during and after irradiation the fuel assemblies are stored in a fresh assembly dry storage vault, in the reactor pool and core and in an intermediate storage pool outside the containment, respectively. Final disposal of the fuel assemblies has been through reprocessing by USDOE. It is essential for a safe storage of the assemblies that they are: kept in a critically safe configuration; effectively separated from the public (radiation protection); kept under good ageing conditions; easily safeguard controlled.

1. REGULATIONS FOR HANDLING OF HEU AND LEU FUEL ASSEMBLIES

In Sweden, the nuclear operator is responsible for the safe operation of the plant. SKI, the Swedish Nuclear Power Inspectorate, is the Swedish government regulatory body under the Nuclear Activities Act and as such it is the authority which ensures that the nuclear operator assumes this responsibility.

SKI also supervises the accounting for and control of nuclear material within Sweden and ensures that national and international safety regulations are followed. This is done in cooperation with the IAEA.

The handling, storage and use of the fuel assemblies at the R2 site complies with the SKI rules.

2. THE ARRIVAL AND THE DRY STORAGE FOR FRESH FUEL

The fresh fuel assemblies, which normally arrive in batches corresponding to a one year supply, are immediately transferred to a dry storage vault. The vault can contain up to 168 assemblies plus 12 control rods with mounted control rod followers. The vault has good physical protection, as it is situated in a controlled area. The activity in the vault is not monitored. Critical safe configuration during transport to and from the vault is kept by restricting the number of assemblies allowed to be moved at one time to 4. Normally the transport is made on a small trolley.

The assemblies are kept lying down on shelves in the vault. The safe critical distance between them is maintained by fixed aluminium spacers. Only one assembly at a time can be placed in each compartment. The shelves are covered with 1 mm Cd to secure the criticality margin in case of a collapsed shelf system together with a flooding water system. Moderator materials such as D_2O, graphite or beryllium are not allowed in the vault. Only two persons are allowed to be simultaneously in the vault to reduce the amount of moderating material.

The assemblies are kept in sealed plastic bags to avoid humidity, dirt or dust which might affect them.

The fresh assemblies are checked in a special calibration room just outside the vault. This room is equipped with a meter frame (rig) for checking the outer dimensions of the assemblies. There are also checking gauges for measurements of the cooling channel width. Possible contamination of uranium is checked by smear tests.

Procedures for safeguard and bookkeeping of the assemblies follow international standards. All fuel handling is supervised by appropriate personnel.

3. THE FUEL STORAGE IN THE REACTOR POOL

The reactor pool is divided into three compartments, separated by large removable ports. The first compartment, pool 1, is mainly occupied by the reactor tank, its tubing and operational equipment. Pool 2 is used for storage of the fuel assemblies and core equipment. This pool is also used for loading and unloading of spent fuel into the transportation cask. Pool 3 contains a zero power research reactor, R2-0, with storage racks for that reactor. There is also a four story fuel rack for fully burned (and cut) R2 assemblies in pool 3. Figure 1 gives an overview of the reactor hall with the pools.

The fuel assemblies are normally taken from the vault directly to the reactor pool top and then loaded into the core. After each cycle, the core is unloaded into fuel storage racks in the reactor pool. These storage racks in pool 2 contain all of the spent fuel in current use and are dimensioned for three core loadings of fuel, that is at least 190 assemblies plus 18 control rods. A schematic picture of a standard fuel rack is given in Figure 2. One of these racks is specially constructed for gamma irradiation purposes. All pool fuel racks have a covering net to avoid damage from dropped equipment.

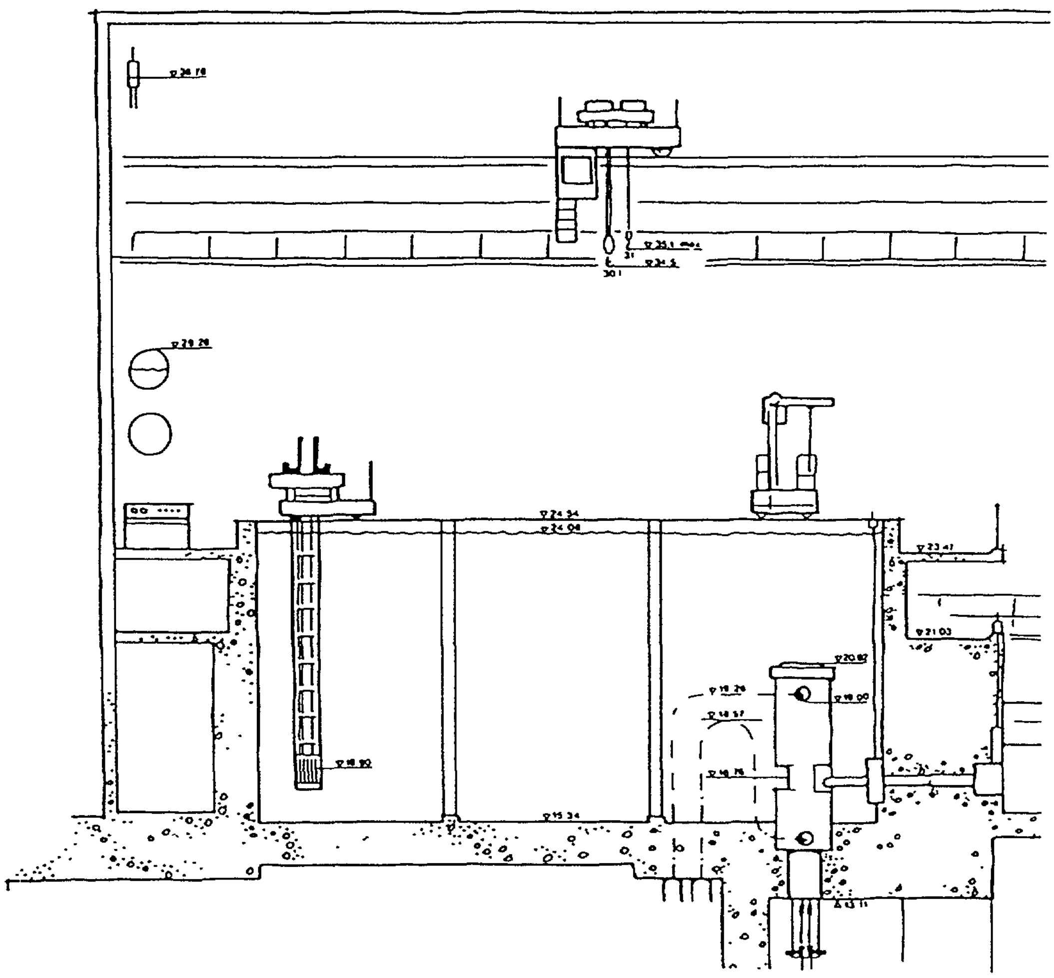


FIG. 1. R2 reactor hall overview.

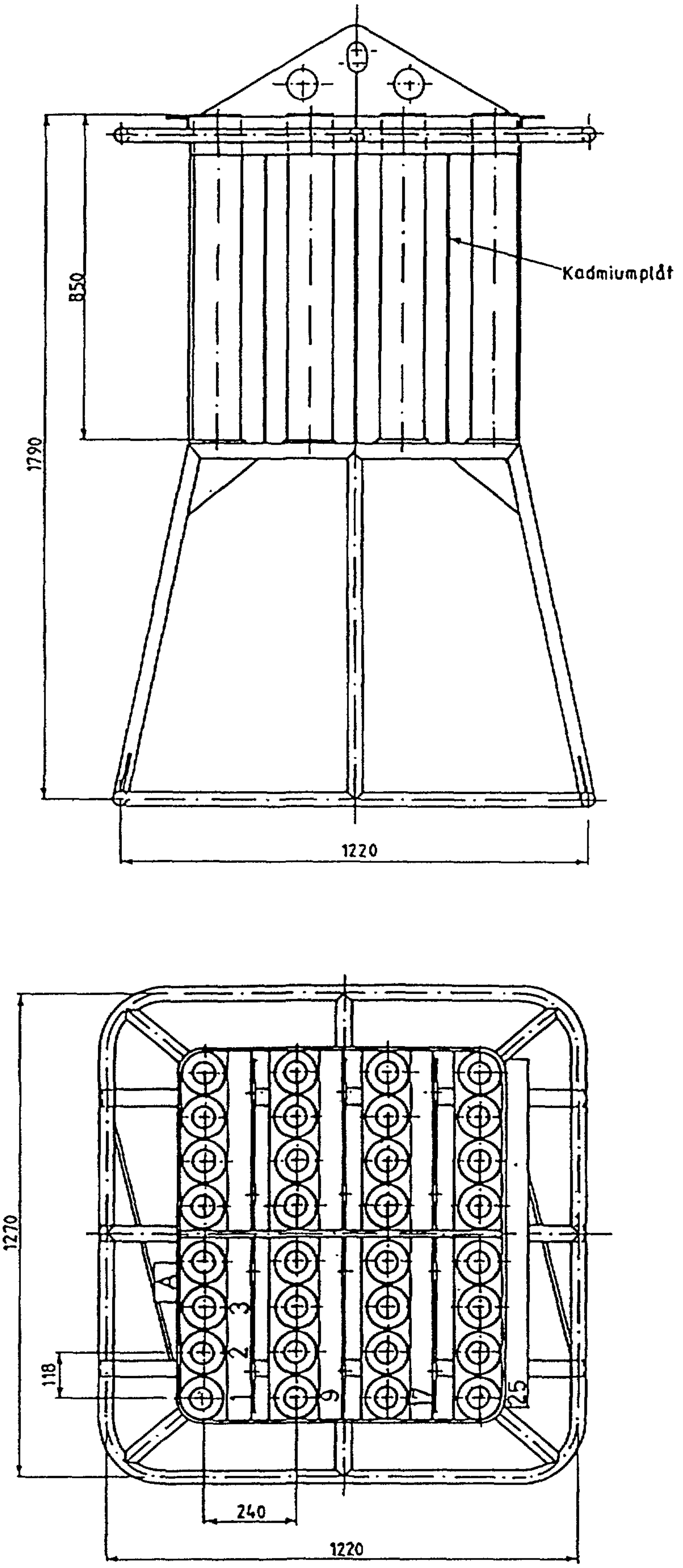


FIG. 2. Schematic view of fuel storage rack.

The storage rack in pool 3 contains 8 cages in four levels, each with 9 cut assemblies. The whole rack is secured to the pool wall and special rails prevent other assemblies or rack from coming close to them. A simplified drawing of this rack is given in Figure 3.

The pool system is fully instrumented for automatic and safe operation. The water system contains several forced circulation systems with cooling and ion exchange cleaning. The water level is automatically adjusted by adding deionized water from a large tank system. Radioactivity and conductivity of the water are monitored. There are alarms for exceeding the maximum allowable limits.

The fuel handling between the core and the storage racks in pool 2 is done by hand. Movements of fuel bundles are only made after written requests, and a maximum of two assemblies are allowed to be handled simultaneously. The fuel assembly name is read before each move and the new position is recorded on the transfer document as well as on a core and rack panel.

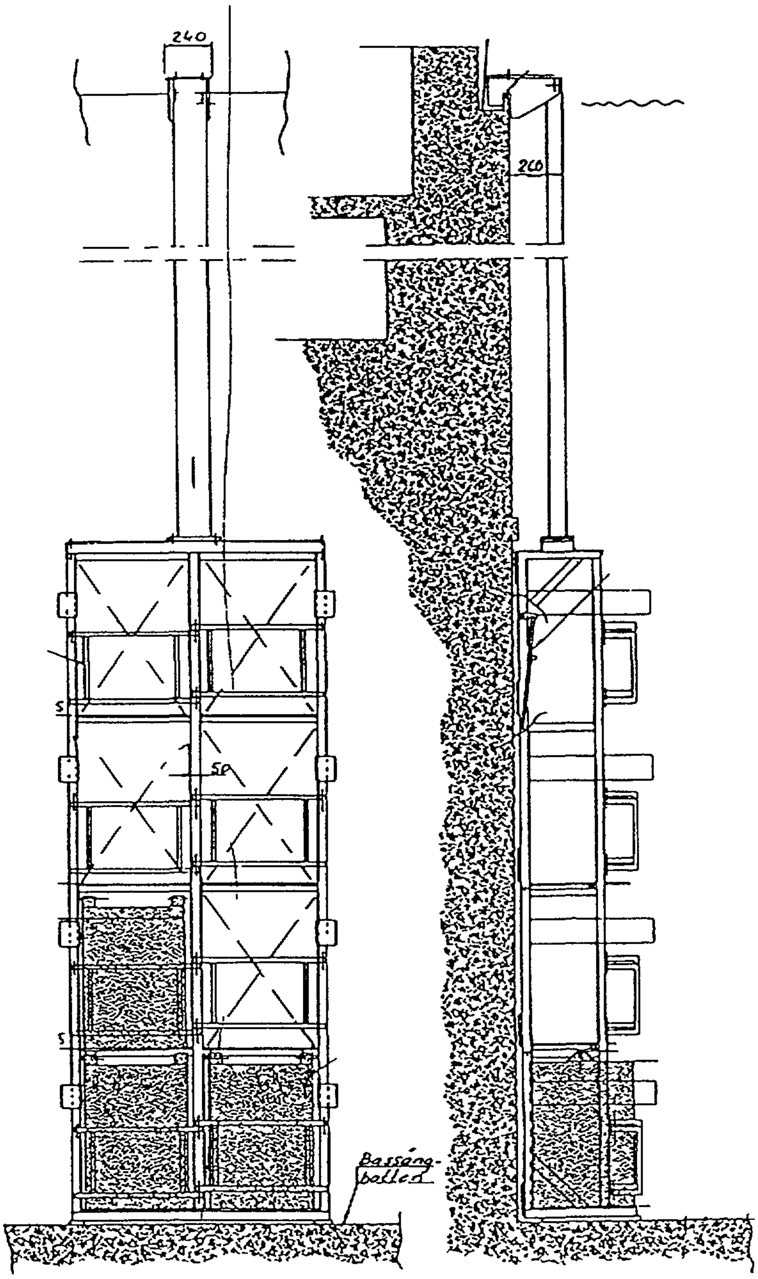


FIG 3 Schematic view of multi-level storage rack

4. THE INTERMEDIATE STORAGE IN THE FA POOL

The spent HEU fuel was up until 1988 sent back to the US for reprocessing. However, since DOE stopped further return of fuel at that time all the spent fuel assemblies stored are at the site waiting for final disposal. This has caused much concern with the authorities due to the large number of highly enriched assemblies stored in one place. The normal operation of the reactor would have been impossible with all spent fuel residing in the pools, so additional storage space was urgently needed. Studsvik had from earlier research reactors a storage facility, FA, with three reloading pools. FA is situated on the Studsvik site close to the reactor, but transport must occur outside of the containment.

The concrete walls of one of the three pools was retrofitted with an epoxy layer for better water conditions. The pool was then ready for use as a short term storage for the R2 spent fuel assemblies after they had been cut. The pool was designed with a bottom layer storage rack with 145 positions for fuel assemblies (or cut control rod follower fuel). A schematic overview of the pool is given in Figure 4. Soon it became clear that a second layer of fuel racks would be required and that was accomplished by construction of a movable rack system. It had to be removable to allow for the safeguard inspection of the lower level.

The rack system has a construction with an overmoderated lattice to ensure subcriticality. An absorption plate (1 mm Cd) was added between the two layers. The plates take care of the improbable case where the rack or part of it is compacted by some external force.

This pool system is much simpler in design than the reactor pool. The pool temperature is only 4 to 6 °C above room temperature so there is no need for a pool cooling system. A simple ion exchange system was installed to improve the water quality as described below in section 5 "water chemistry". The pool is covered with a lid to minimize fouling and growth of algae in the water.

5. WATER CHEMISTRY

The aluminium clad MTR fuel assemblies are sensitive to contamination in the water as well as its acidity. The water chemistry of the pool water is therefore crucial.

Special care has been taken at the FA pool to maintain a good water chemistry. The FA pools have a conventional water cleaning system, and a simple submersible ion-exchange system has been added to clean the water. This has been done to reduce the consequences of a possible accident with water leakage or pipe break. The system consists of a small submersible water pump, which forces the water through a small container box with a disposable ion exchange cartridge. The instrumentation of the system is reduced to pressure and conductivity gauges. There is also a water sample tap. The layout of the system is shown in Figure 5. This system has been in use for three years and the conductivity and activity levels of the water are effectively reduced as can be seen in Figure 6. The ion exchange cartridge was renewed in February, August and November of 1992 and the positive influence on both specific activity and conductivity can be seen in this Figure.

This system can be recommended for a small storage facility as it is simple, easily built from standard parts and thus quite cost effective. Maintenance needs of the system are minor. Circulation flow, temperature and conductivity are checked only once a week.

6. CRITICALITY

The pool storage racks are constructed for critical safe storage under all conditions. That requirement is met by using large spacing between the assemblies (over-moderated lattice). The racks have a geometrical configuration that makes it impossible to move the racks close to each other and thus obtain the critical mass. A typical storage rack for 36 assemblies is shown in Figure 3.

The reactor operator uses the well-known reactor physics codes CASMO and SIMULATE to verify safe handling and storage of the fuel with regard to criticality.

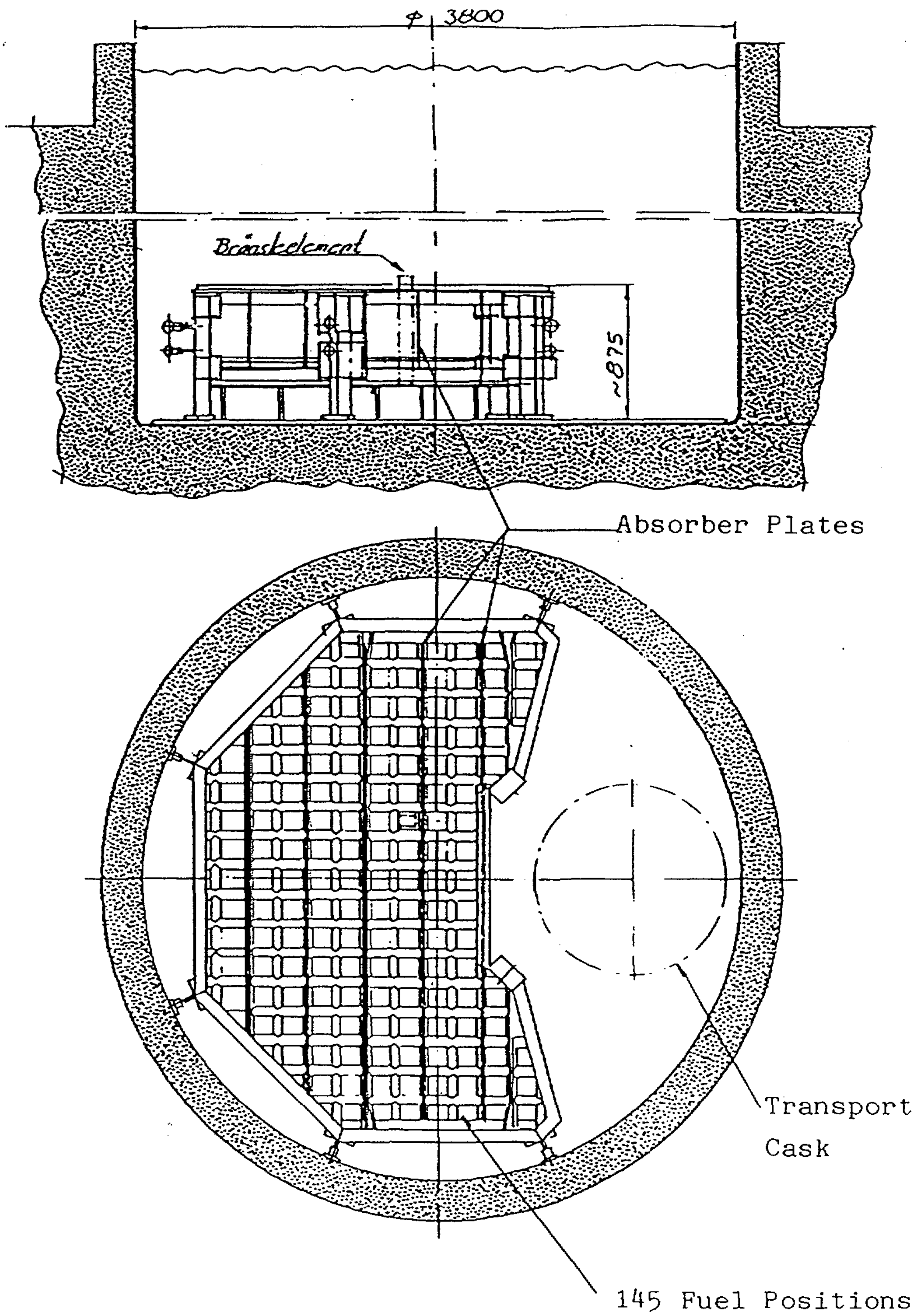


FIG. 4. Schematic view of the FA storage pool.

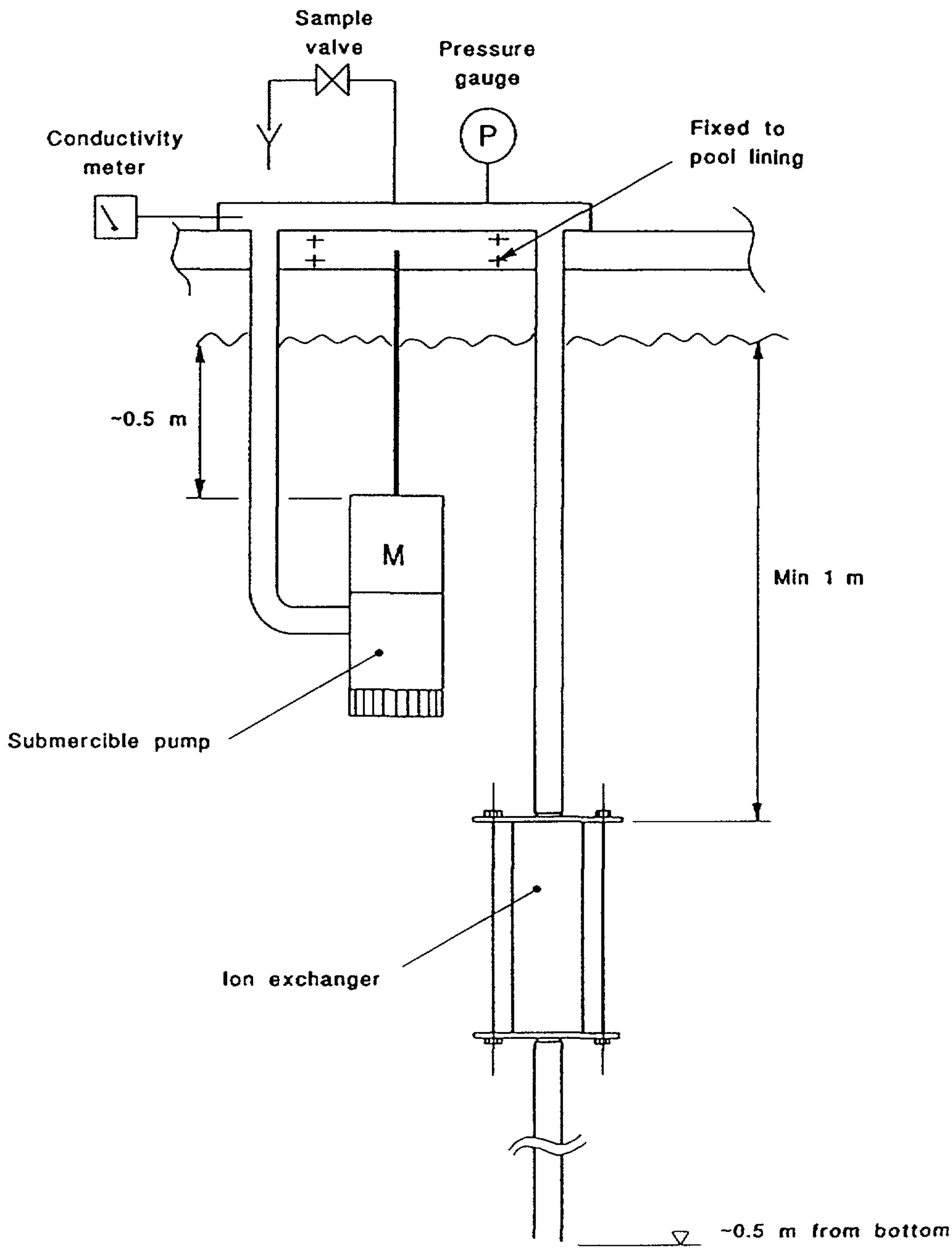
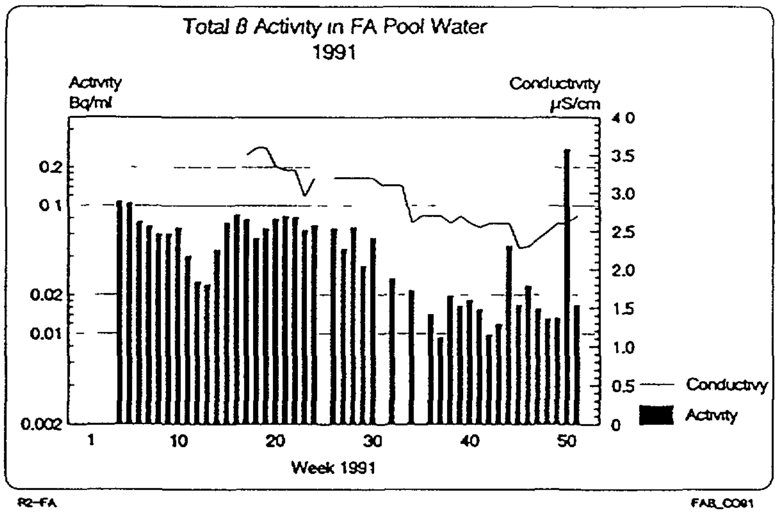


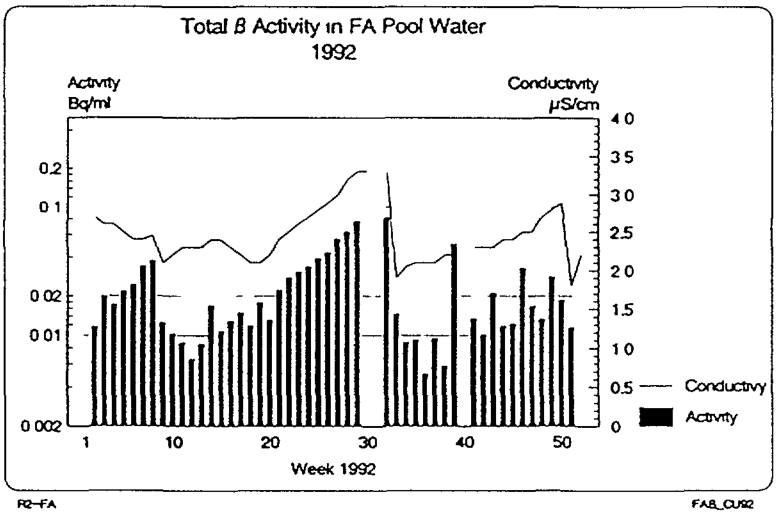
FIG. 5. Simple water cleaning system with ion exchanger.

7. TRANSPORT

Transfer of fuel assemblies within the R2 building and to the FA storage are only made after written requests. The documentation and the procedure for transport follow well established routines laid down by SKI. The fuel assembly name is read before each move and the new position is verified on the transfer document. It is also recorded in a computerized database and on a rack and/or core panel in the control room.

The typical transfer document contains data on the responsible sending and receiving persons, the weight, activity and heat dissipation of the fuel.





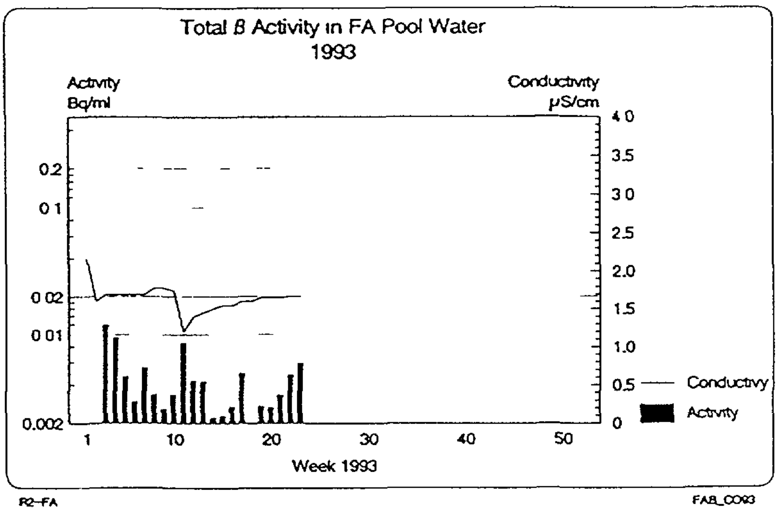


FIG. 6. Pool water activity and conductivity 1991-1993.

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