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# Storage of spent fuel from power reactors

*Proceedings of a symposium held in Vienna, 9–13 November 1998*

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# FOREWORD

An International Symposium on Storage of Spent Fuel from Power Reactors was held in Vienna from 9 to 13 November 1998. The symposium was organized by the International Atomic Energy Agency in co-operation with the OECD Nuclear Energy Agency.

The symposium gave an opportunity to exchange information on the state of the art and prospects of spent fuel storage, to discuss the worldwide situation and the major factors influencing the national policies in this field and to identify the most important directions that national efforts and international co-operation in this area should take.

Dominant messages retrieved from the symposium are that the primary spent fuel management solution for the next decades will be interim storage, the duration time of interim storage becomes longer than earlier anticipated and the storage facilities will have to be designed for receiving also spent fuel from advanced fuel cycle practices (i.e. high burnup and MOX spent fuel).

It was noted that the handling and storage of spent fuel is a mature technology and meets the stringent safety requirements applicable in the different countries. The changes in nuclear policy and philosophy across the world, and practical considerations, have made interim storage a real necessity in the nuclear power industry. This is being addressed adequately by utilities, vendors and regulators alike.

The IAEA wishes to express appreciation to all authors for their presentations to the symposium and papers included in these proceedings. The IAEA officer responsible for organizing the symposium and preparing this publication was M.J. Crijns of the Division of Nuclear Fuel Cycle and Waste Technology.

## *EDITORIAL NOTE*

# CONTENTS

## OPENING SESSION



## SESSION 1. NATIONAL PROGRAMMES

**SESSION 2. TECHNOLOGIES**



**SESSION 3. EXPERIENCE & LICENSING**

## SESSION 4. R&D AND SPECIAL ASPECTS



## CLOSING SESSION



## POSTER SESSION

# OPENING SESSION

**Chairman**

**V. MOUROGOV**
IAEA

# SPENT FUEL MANAGEMENT OVERVIEW: A GLOBAL PERSPECTIVE

XA9951768

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**Abstract**

The paper defines the main spent fuel management strategies and options, highlights the challenges for spent fuel storage and gives an overview of the regional balances of spent fuel storage capacity and spent fuel arising. The relevant IAEA activities in the area of spent fuel management are summarised.

## 1. INTRODUCTION

Spent fuel management has always been an important stage in the nuclear fuel cycle and stands among the most vital challenges, common to all countries operating nuclear reactors. It comprises the technical operations that begin with the discharge of spent fuel assemblies from an operational reactor and end either with the direct disposal of the spent fuel assemblies in a geological repository (open, once-through cycle) or with the reprocessing of spent fuel, in view of the recycling of plutonium and uranium in new mixed oxide fuel and the disposal of the remaining waste (closed cycle). Other technical operations in spent fuel management are: treatment, transportation and storage of spent fuel.

A third approach presently considered, is the deferral of the decision to choose between the open or closed cycle and is called the "wait and see" strategy resulting in the decision to only consider interim storage of the spent fuel. This strategy provides the ability to monitor the storage continuously and to retrieve the spent fuel later for either direct disposal or reprocessing. At present, most countries with nuclear programmes are using the "wait and see" strategy. The practical approach proposed may be different depending on the type of fuel concerned, more specifically the material behaviour of the various fuels (e.g. Magnox fuel cladding is highly corrosive, reason why Magnox fuel is reprocessed after short cooling times). Additionally, some countries follow one approach while evaluating the other approaches that might be applied in the future.

Today, the world-wide reprocessing capacity is only a fraction of the total spent fuel arising and since no final repository has yet been constructed, there will be an increasing demand for interim storage. Depending on the strategy, the interim storage period can vary. World-wide a large amount of fuel is stored in the pools of the NPPs and several wet and dry storage facilities are in operation. Spent fuel can be safely stored for long periods of time (some spent fuel has now been stored for over 30 years), depending on the materials used. One of the challenges in spent fuel management is to demonstrate and enhance the capability of storing spent fuel safely for several decades.

## 2. HISTORICAL DEVELOPMENT

In the early nuclear power era, the general philosophy was to close the nuclear fuel cycle. Spent fuel was generally stored in the fuel storage pool at the reactor (AR) and, after transportation, in the spent fuel storage pool of a reprocessing plant which served as a wet buffer store. Because there was not enough storage capacity at the reprocessing plants in that period and because some countries decided not to close the fuel cycle, the utilities started to rerack their reactor pools and using neutron absorbers to expand the pool storage capacity.

Additional pool type storage facilities away from the reactor (AFR), either at the reactor site (RS) or at other sites (OS), were built. Pool type facilities, in which the spent fuel is submerged under water, are usually referred to as wet storage facilities. Because many countries deferred the decision as mentioned earlier, and the lack of final repositories, the storage time of the spent fuel is increasing compared to the original foreseen storage duration, making new developments necessary. The dry storage technology has been developed for longer-term AFR storage. This type of storage has many benefits, including the possibility of passive cooling and a reduced need for service (e.g. water chemistry).

Since the early 1980's, countries such as France commenced to apply burnup credit in their criticality safety analysis of spent fuel systems, which traditionally assumed that the fuel is fresh. This to facilitate a further increase in existing spent fuel capacity, by allowing smaller center-to-center distances in fuel storage systems, and to reduce the number of spent fuel shipments.

## 3. CHALLENGES

Many at-reactor (AR) storage pools have been used to full capacity in recent years, threatening the continued routine operation of the power plant, in some cases. Recent designs of reactors have in fact now incorporated pools that can accommodate arisings for periods of up to 40 years. However, most older operational plants, due to their limited AR capacity, have necessitated away-from-reactor (AFR) storage to maintain their operationability. Timely construction of new storage facilities and implementation of burnup credit (Section 5) is a prerequisite to provide adequate capacity for the increasing future spent fuel arisings (Section 4.2).

The trend to higher fuel burnup, and consequently higher enrichment of the fresh fuel, and the use of plutonium in mixed oxide fuel, leads to other spent fuel characteristics (i.e. higher decay heat and flatter downward curve over time). This demands in its turn a longer storage period than for the present spent fuel with burnup lower than 40 GWd/t.

In many Member States, the lack of final repositories and the deferral of the decision to choose between the open or closed fuel cycle will lead automatically to long storage periods even of uncertain duration. The lifetime of many existing storage facilities will be extended and new facilities serving long-term storage have to be built. For the existing facilities it will be necessary to review the safety and in some cases improvements to upgrade the safety have to be performed in order to meet the more stringent requirements. The design of new facilities has to take into account not only the fuel behaviour during long-time storage but also the behaviour of the materials, equipment and installation used for safe storage over very long periods, this might imply changing to dry storage technology.

With respect to operating experience, spent fuel can be safely stored for long periods of time (some spent fuel has now been stored for over 30 years), as was mentioned earlier. However, the possible storage duration for different fuel types is dictated by the corrosion resistance of the cladding material used (for example, nodular corrosion between the materials of the spacer grid and cladding in RBMK fuel, limits the wet storage period. The latter problem is solved by using zircaloy as spacer material in new fuel assemblies). There is a scientific and technical consensus that the present technologies of spent fuel storage give adequate protection to population and the environment, but there is a strong interest to see whether further improvements can be achieved in spent fuel storage. An example of successful storage is the experiment in Greifswald, Germany, where it has been demonstrated that also defective WWER fuel can be dried effectively and stored safely under dry conditions without any release of water or vapour.

The uncertainties in future policies make it necessary to guarantee a safe and long-term storage enabling retrievability. The technology has to be adapted to this requirement and challenge.

A typical challenge exists in Eastern Europe. The break up of the former Soviet Union and the political and economical changes in Central and Eastern Europe have created problems in spent fuel management in this region. In the past most of those countries relied heavily on the Soviet Union. More than 30 WWER and 4 RBMK reactors are operating outside the Russian Federation. Return agreements signed in the past with the former Soviet Union were voided or amended on a commercial basis. Due to economic constraints most countries did not opt for the commercial contracts. As a result, many NPPs are faced with a shortage of spent fuel capacity.

## 4. BALANCE OF SPENT FUEL ARISINGS AND STORAGE CAPACITY

### 4.1. Status of Nuclear Power

Today the growth of nuclear power is at a standstill in Western Europe and North America, while expanding in parts of Asia and Eastern Europe. At the end of 1997, 437 nuclear reactors operating in 31 countries provided about 17 per cent of global electricity supply, slightly lower than the previous year [1]. The total installed nuclear capacity was 352 GWe, whilst 36 reactors are under construction with a total additional capacity of 28 GWe. Table I shows the status for four world regions, i.e. West Europe, Asia and Africa, East Europe and North and South America. Several existing reactors are now approaching the end of their design life. In order to keep the electricity production at the needed level, decisions must be made to extend their time in service, to replace them with new plants or to find other options.

TABLE I. STATUS OF NUCLEAR POWER IN WORLD REGIONS

[GWe]

| Regions | Installed Nuclear Capacity | |
|---|---|---|
| | Operating | Under construction |
| West Europe | 127.0 | 1.4 |
| Asia & Africa | 64.3 | 12.2 |
| East Europe | 46.3 | 11.2 |
| North & South America | 113.9 | 2.9 |
| World | 351.6 | 27.7 |

### 4.2. Spent fuel arising

In 1997, the annual spent fuel arising from all types of power reactors (i.e. of 352 GWe) world-wide amounted to about 10,500 tonnes of heavy metal (t HM). About 35% came from each of the two regions West Europe and North and South America and 15% of each of the two regions East Europe and Asia and Africa. Fig. 1 shows the current and projected regional spent fuel arisings.

The total amount of spent fuel accumulated world-wide at the end of 1997 was about 200,000 t HM, of which about 70,000 t HM were reprocessed. Hence, about 130,000 t HM of spent fuel is presently being stored in at-reactor (AR) and away-from-reactor (AFR) storage facilities awaiting either reprocessing or final disposal (Table II and Fig. 2). Table II gives not only the breakdown of the spent fuel presently stored according to the type of facility, but also in world regions. On a regional basis, the picture for spent fuel to be stored in 1997 looked different than the annual spent fuel arisings. About 50% is stored in North and South America, 25% in West Europe and the remaining part in East Europe and Asia and Africa . More recent information on national data will be given in the various national presentations.

Projections indicate that the cumulative amount generated in the world by the year 2010 may surpass 340,000 t HM and by the year 2015 395,000 t HM. In 2010 about 225,000 t HM of spent fuel has to be stored, in 2015 more than 260,000 t HM. Of this total amount in 2015, the amount in West Europe will remain about the same (because of reprocessing spent fuel) and will four fold in Asia and Africa. The ratios of spent fuel to be stored in 2015 are projected to be 49% for North and South America, 14% for West Europe and 19% for both East Europe as well as for Asia and Africa (Table III).

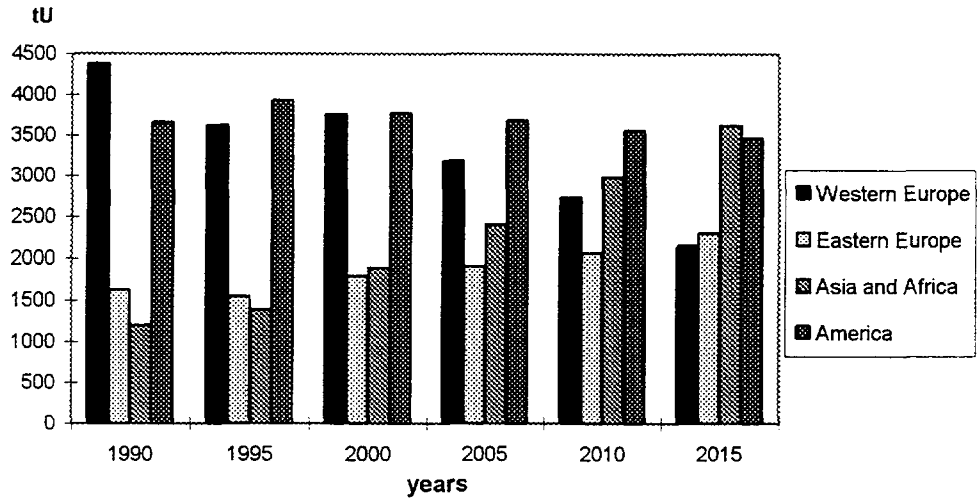


*FIG. 1. Annual spent fuel arisings in world regions*

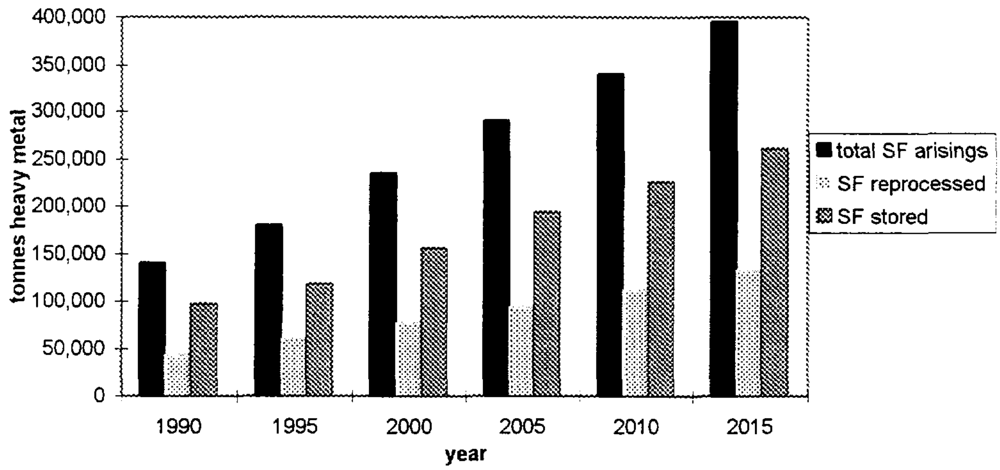


*FIG. 2. World spent fuel arisings and amounts of spent fuel reprocessed and stored*

TABLE II. STATUS OF SPENT FUEL STORED AT YEAR-END 1997

[kt HM]

| Regions | AR | AFR Wet | AFR Dry | Total |
|---|---|---|---|---|
| West Europe | 13.9 | 19.3 | 1.0 | 34.2 |
| Asia & Africa | 11.6 | 0.2 | 0.7 | 12.5 |
| East Europe | 7.8 | 9.9 | 0.3 | 18.0 |
| North & South America | 59.8 | 1.5 | 3.3 | 64.6 |
| World | 93.1 | 30.9 | 5.3 | 129.3 |

TABLE III. PROJECTIONS OF SPENT FUEL STORED IN WORLD REGIONS

[kt HM]

| Regions | 1997 | 2005 | 2010 | 2015 |
|---|---|---|---|---|
| West Europe | 34.2 | 40.1 | 38.9 | 36.4 |
| Asia & Africa | 12.5 | 27.6 | 38.6 | 50.2 |
| East Europe | 18.0 | 31.1 | 39.4 | 47.9 |
| North & South America | 64.6 | 91.3 | 108.4 | 125.9 |
| World | 129.3 | 190.1 | 225.3 | 260.4 |

### 4.3. Spent fuel storage capacity

Nearly all countries operating nuclear power plants have increased their original AR storage capacity by reracking the spent fuel pools with high density racks and by implementing burnup credit or by commissioning additional AR and AFR storage facilities. The AR spent fuel storage facilities are wet storage type facilities, as were also most of the AFR spent fuel storage facilities away from reactor built in the past. Most of the newly built and future planned facilities are dry storage type facilities. Dry facilities involve storage of spent fuel in a gaseous environment, such as an inert gas or air, and include casks and vaults. A cask is a massive container which may or may not be transportable. Vaults consist of above or below ground reinforced concrete buildings containing arrays of storage cavities suitable for containment of one or more fuel units.

Various types of wet and dry storage facilities are in operation. The current world storage capacity is about 231,000 t HM, of which 46% is situated in North and South America, 30% in West Europe, 15% in East Europe and the remaining part in Asia and Africa (Table IV and Figure 3). The storage capacity of facilities under construction world wide is about 12,000 t HM.

TABLE IV. STATUS OF SPENT FUEL STORAGE CAPACITY IN WORLD REGIONS

[kt HM]

| | In operation | | | | Under construction | | |
|---|---|---|---|---|---|---|---|
| Regions | at NPP | Wet | Dry | Total | Wet | Dry | Total |
| West Europe | 26.1 | 31.7 | 9.2 | 67.0 | | 0.8 | 0.8 |
| Asia & Africa | 20.0 | 1.9 | 0.7 | 22.6 | 0.7 | 0.8 | 1.5 |
| East Europe | 14.3 | 19.6 | 0.8 | 34.7 | 0.8* | 1.6 | 2.4 |
| North & South America | 94.9 | 1.8 | 10.0 | 106.7 | | 6.8 | 6.8 |
| World | 155.3 | 55.0 | 20.7 | 231.0 | 1.5 | 10.0 | 11.5 |

* by reracking AFR storage capacity

### 4.4. Balance of arising and capacity

In 1997, the spent fuel storage capacity world wide exceeded the amount of spent fuel to be stored by about 100,000 t HM and in all types of storage facilities there was excess capacity available. Fig. 4 compares the capacities of the various storage types with their current inventories.

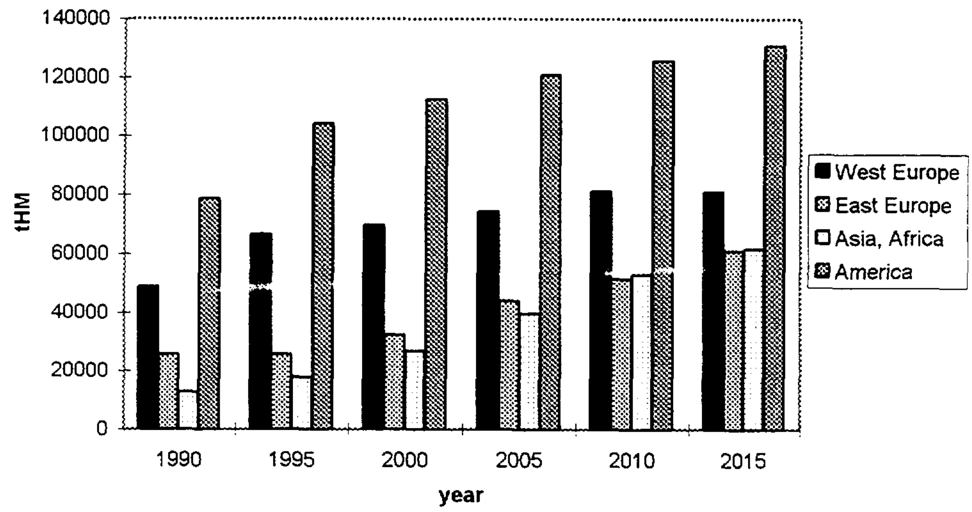


*FIG. 3. Spent fuel storage capacity in world regions*

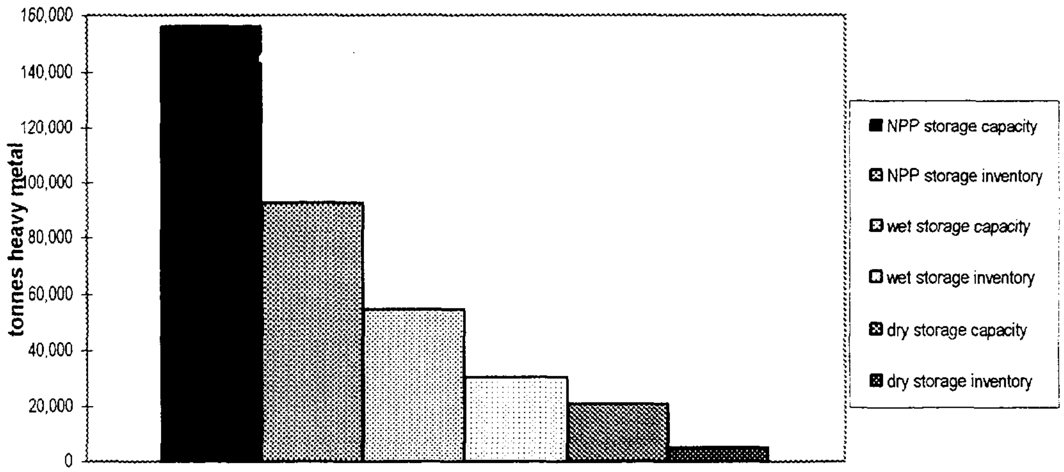


*FIG. 4. Comparison of capacities and inventories of different types of spent fuel storage*

On a world basis, the spent fuel arisings will fill the existing storage facilities and those under construction by around 2010, if no new additional facilities are built by that time. This implies that decisions should be taken for timely expansion within the next decades. However, if new storage capacities are launched at the rate of the last 10 years, no shortage is envisaged globally, see Figure 5. A world wide approach is of course somewhat unrealistic and a look from the national perspective is recommended.

Nationally, the situation differs from country to country and sometimes even from utility to utility. In some cases, the storage pools are fully occupied by spent fuel allowing emergency core unloading only by special measures like in Armenia, else, additional storage capacity has to be installed in time, to avoid this problem. In other cases, also additional storage capacity has to be installed timely to replace wet storage facilities which can not be refurbished, as is the case in Chernobyl. In particular in some countries of Eastern Europe, plant operation might be jeopardized if additional storage capacity cannot be installed in time.

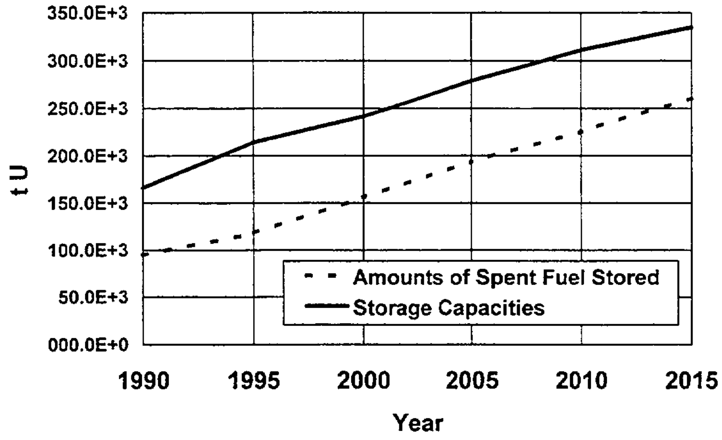


*FIG. 5. Comparison of projected spent fuel storage capacities and amounts of spent fuel stored*

If there will be a delay in MOX fuel utilisation (e.g. delay in reprocessing), more spent fuel needs to be stored. This situation would demand more storage capacity. Further postponement of the decision related to the development of final repositories will also lead to a higher demand in spent fuel storage capacity.

## 5. TRENDS

There is a potential advantage of implementing burnup credit at the different stages of the back-end of the fuel cycle, such as at storage, transportation and disposal. Various burnup credit levels are under study or implemented and allow to take a credit for the reactivity reduction in the spent fuel associated with the use of the fuel in the reactor. In first instance, burnup credit assists in expanding the capacity of a facility/device (e.g. France, Germany, Korea, Lithuania, Russia, Spain, Switzerland and USA) by allowing smaller center-to-center distances in the fuel storage system and consequently, it can reduce the number of shipments needed. Further, burnup credit allows the introduction of initially higher enriched fuel in the existing storage, reprocessing or transport systems.

Other motivations for using burnup credit are based on economics, public health and safety, resource conservation and environmental quality. In the area of spent fuel storage, burnup credit can avoid or minimize the environmental impacts associated with expanding or building new storage pools or dry storage facilities. For dry cask storage and transportation, higher capacity casks result not only in fewer shipments, but also to less worker and public exposure, and lower risk both radiological and non-radiological.

Burnup credit can be used to maintain production rates at existing reprocessing facilities even while fuel enrichments increase, thus avoiding the environmental impacts of constructing new facilities, or expanding old ones. Many experts believe that burnup credit is a necessity for any viable disposal scheme of spent fuel. Ignoring the reduced reactivity from burnup credit would lead to expanded disposal sites and unnecessary use of land and money.

As mentioned in Section 2, dry storage has gained more interest, because of the possibility of passive and thus simpler cooling system.

## 6. IAEA ACTIVITIES

### 6.1. General

The IAEA is collecting a variety of information on nuclear energy for its various assessments. This information is normally stored in data bases, which are supporting these assessments. In particular, we should mention here the data bases Nuclear Fuel Cycle Information System (NFCIS), International Nuclear Event Scale (INES) and Incident reporting System (IRS).

NFCIS is an international directory of civilian nuclear fuel cycle facilities. Its purpose is to provide the IAEA and its Member States with current, consistent, and readily accessible information on existing and planned nuclear fuel cycle facilities throughout the world, including spent fuel storage facilities. It is expected that some information from this data base will be available on the internet in the near future. *[For long-term projections, a new model (Nuclear Fuel Cycle Requirements Simulation System, VISTA) is used for the estimation of uranium and fuel cycle service requirements. It has been developed by the IAEA to assist in the analysis of both nuclear fuel cycle service requirements and actinide generation based on different realities. VISTA is a long-term scenario-based tool which allows the calculations to be carried on for a given set of assumptions.]*

INES and IRS serve to promote safety culture in nuclear facilities. The particular purpose of INES is to facilitate prompt communication between the nuclear community, the media and the public and that of IRS to analyse causes of significant incidents.

### 6.2. Spent Fuel Management

The Agency conducts a variety of projects on spent fuel management and elaborates guidelines for the storage of spent fuel in order to provide Member States with information on the development of safe, reliable and economical technical solutions in spent fuel management and to provide advisory services to Member States for the implementation of the best internationally agreed methods for storing and handling spent fuel from power and research reactors [2,3]. The following summarises the main activities and projects related to spent fuel storage:

*6.2.1. Advisory Groups*

- *Regular Advisory Group (RAG) on the Status and Prospects on Spent Fuel Management.*

  The RAG overlooks the whole spent fuel area including spent fuel storage. The objectives of the RAG are to review the situation in spent fuel management in the Member States, to identify the most important directions of national efforts and international co-operation in this area, and to assist the Agency in formulating the future programme in the subject field.

*6.2.2. Co-ordinated Research Projects*

- *The CRP on Spent Fuel Performance Assessment and Research (SPAR).*

  This CRP is to carry out research work which will evaluate and justify the storage of spent fuel for long periods of time (more than 50 years).

*6.2.3. Review Studies*

- *"Survey of Experience with Wet and Dry Storage of Spent Nuclear Fuel".*

  This survey describes the world-wide experience that has been gained over the last decade with the storage of spent fuel.

- *Review of the Technologies and Safety Aspects of a Regional Spent Fuel Storage Facility.*

  The purpose of this project is to collect and evaluate information on the technologies and safety aspects of a regional spent fuel storage facility. For countries with small nuclear program, regional spent fuel storage facilities would be very attractive. For countries with only research reactors without the possibility to send their fuel back to the provider, a regional storage is essential.

- *Review of the Implementation of Burnup Credit in Spent Fuel Management Systems.*

  The purpose of the activity is to describe the current and future aspects of burnup credit implementation in spent fuel management systems and to provide an overview of the status of national practices, in particular on spent fuel storage and transportation.

*6.2.4. Topical Issues*

- *Decontamination and Modification of Spent Fuel Storage Casks for Transport & Storage.*

  Dedicated casks will be needed to transfer fuel to off-site interim storage, reprocessing and final disposal. The contamination of casks during fuel handling operations, and the potential for subsequent release, is an inevitable consequence of reactor operations and will continue to require careful management as it is a topic of direct concern to the public. In some cases, casks have been developed both for transport of spent fuel and for storage.

- *Improvement of the Safety of Spent Fuel Storage in WWER and RBMK*

  An extrabudgetary project sponsored by the Japanese Government began in 1995 with the aim to improve the safety of spent fuel storage in countries operating WWERs and RBMKs. In the framework of this project, three different tasks have been performed.

  - The workshops on „Spent fuel dry storage technologies“ have been held, to bring together experts from representative industrialised countries and from Member States, operating WWER and RBMK type plants, in order to exchange information on the safety of spent fuel storage technologies.

  - A dry spent fuel storage test is under way in the Novo Voronezh NPP, where pre-characterized rods will be stored for 3, 6 and 12 months and than post-characterized to evaluate the performance under dry storage conditions at a temperature of 350 to 400°C (see poster 15P).

  - The computer codes COBRA-SFS for thermal-hydraulic analysis of storage facilities and SCALE for criticality and shielding analysis were tested on instructive WWER problems. Input data for WWER were evaluated and handbooks for WWER users were written (see posters 12P and 13P).

- *Activities of the Co-operation Forum for WWER Regulators.*

  Under an extrabudgetary activity sponsored by the US, a Working Group of the Co-operation Forum for WWER and RBMK was founded to exchange the opinion and experience of the regulatory bodies in the field of licensing of spent fuel storage facilities. The Working Group envisages to prepare three documents on:
  - advisory comments to the Safety Analysis Report of a dry storage facility;
  - experience in the evaluation and re-evaluation of existing spent fuel storage facilities;
  - information for the public in relation to the construction of new storage facilities.

## 7. CONCLUSIONS

On the basis of information received from and exchanged among Member States, the following conclusions can be drawn:

- the spent fuel arisings expected over the next 15 years are quite well known; the total amount of spent fuel accumulated at the end of 1997 world wide, was about 130,000 t HM and the projected amount to be in store by the year 2010 is about 225,000 t HM (assuming that part of the spent fuel will be reprocessed, as is current practice).
- currently, there is sufficient spent fuel storage capacity on a world wide basis. However, nationally the situation can be different and might need urgent attention;
- more spent fuel storage capacity is required than earlier anticipated, because most countries deferred their decision to choose between the open and closed fuel cycle;
- the first geological repositories for the final disposal of spent fuel are not expected to be in operation before the year 2010 and many countries did not yet start investigations. Thus, the use of interim storage will be the primary operational spent fuel management option for the next decades in many Member States.
- the storage duration becomes gradually longer than earlier anticipated, because of the selection of the "wait-and-see" option chosen by many nuclear power countries and the use of high burnup and MOX fuel; these fuels will lead to higher residual heat and will require long heat decay times, implying longer interim storage period before final disposal.
- experience exists in long-term storage of about 30 years without any problems. However, much longer storage periods are expected.
- major challenges are the timely construction of new spent fuel storage facilities, more stringent requirements for existing facilities, the longer term storage of the spent fuel (material aspects and type of storage) and the implementation of burnup credit in spent fuel storage facilities;
- the Agency will continue to assist the Member States in developing and maintaining appropriate approaches, policies, strategies, and technologies in spent fuel management; in particular, it will continue the discussions on regional spent fuel storage facilities, the storage of advanced spent fuel and the requirements for long term storage.

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# SESSION 1

# NATIONAL PROGRAMMES

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Germany

IAEA-SM-352/2

# SPENT NUCLEAR FUEL IN BULGARIA

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**Abstract**

The development of the nuclear energy sector in Bulgaria is characterized by two major stages. The first stage consisted of providing a scientific basis for the programme for development of the nuclear energy sector in the country and was completed with the construction of an experimental water-water reactor. At present, spent nuclear fuel from this reactor is placed in a water filled storage facility and will be transported back to Russia. The second stage consisted of the construction of the 6 NPP units at the Kozloduy site. The spent nuclear fuel from the six units is stored in at reactor pools and in an additional on-site storage facility which is nearly full. In order to engage the government of the country with the on-site storage problems, the new management of the National Electric Company elaborated a policy on nuclear fuel cycle and radioactive waste management". The underlying policy is de facto the selection of the "deferred decision" option for its spent fuel management.

## 1. STATUS AND NEW REALITIES

The development of the nuclear energy sector in Bulgaria is characterised by two major stages that are inextricably bound up with the prospects for use of atomic energy for peaceful purposes in the former socialist system and is based on the technical and economic support provided by the former Soviet Union.

The first stage consisted of providing a scientific basis for the ambitious programme for development of the nuclear energy sector, elaborated by the former communist Government in the country and was completed with the construction of an experimental heterogeneous, water-water reactor (IRT-type reactor with a heat output of 2 MW) at the Institute of Physics of the Bulgarian Academy of Sciences (BAS). The reactor core consists of forty-eight aluminium-coated assemblies which contain 10% or 36% enriched uranium oxide fuel. The reactor was commissioned in September 1961 and operated for 24,600 hours. In July 1989, the reactor was shut down for the implementation of a modernisation and refurbishment programme.

The spent nuclear fuel (SNF) stored during that period of operation amounts to 73 assemblies (totalling to 184 kg heavy metal) of which 57 assemblies are of the type EK-10 with an enrichment of 10% and 16 assemblies of the type C-36 with an enrichment of 36%. At present, this spent fuel is placed in a water filled storage facility, built within the reactor biological shield. The facility can store 112 assemblies of the EK-10 and C-36 type. According to information received from the governing body of the Institute for Nuclear Research and Nuclear Energy to the BAS, the spent fuel from this reactor will be transported back to Russia, within the scope of the contract for transportation of Kozloduy NPP spent nuclear fuel.

The second stage of that programme started with the commissioning of the first unit of the NPP at Kozloduy in 1974 and is continuing up to now. At present, six VVER-type reactors are being operated on the site. The first four units have each a capacity of 440 MW(e) and the other two units each 1000 MW(e). In addition, the construction of another 1000 MW(e) reactor is being planned.

According to their design, the reactor core of the VVER-440 units consists of 312 fuel elements and 37 control rods. Measures for reactor vessel protection against neutron embrittlement were taken and the fuel assemblies along the core periphery of the first three units were replaced by dummy assemblies. That resulted in a reduction of the number of the reactor assemblies by 36. On the average, 400 spent fuel assemblies are annually discharged from the four reactors and are stored according to their design in at-reactor pools for three years. Pursuant to an intergovernmental agreement, they are subject to return to the country of origin, i.e. the former Soviet Union. 3,086 VVER-440 spent nuclear fuel assemblies were sent back under this agreement on a zero value basis during the period 1979-1988.

Already in 1979 through the COMECON, we were notified of a change in the storage period of the spent fuel from 3 to 5 years and, due to that reason, an additional pool-type storage facility of Soviet design was built (Figs. 1 to 3). Some of the operational safety features are shown in Fig. 4. In 1989, the first assemblies were placed in that. storage facility, but yet it has not been licensed by the Bulgarian Nuclear Regulatory Agency owing to a number of discrepancies with the new safety requirements. Spent nuclear fuel storage is authorised through temporary permits which are renewed annually. For the current year, a temporary permit was granted provided the amount of spent fuel stored will not increase. This means that the same quantity of spent fuel to be taken out of the at-reactor pool and transported to the spent fuel storage facility must be pulled out of the spent fuel storage facility and sent back to Russia for reprocessing. The second condition is to start the implementation of its modernisation and refurbishment programme envisaging seismic stabilisation, installation of additional equipment needed for handling spent nuclear fuel generated by the VVER-1000 units and possibly maximize the packing of the basins. Both conditions are met and a Framework Agreement was concluded for the return of 480 VVER-440 assemblies to Russia by the end of this year and the implementation of the programme for modernisation and refurbishment of the spent fuel storage facility has started.

The core of the VVERs-1000 consists of 163 fuel assemblies which have a 3.3% enrichment and a mass of 430 kg HM. Unit No.5 has been in operation since 1987 and Unit No. 6 - since 1991. Each year, half of the core is discharged and the spent fuel assemblies are stored in compact racks in the at-reactor basins. Last year, refuelling of the reactors has started on a three year cycle basis which will reduce the quantity of spent fuel to be stored by 30%.

Table I shows the quantity of spent nuclear fuel stored to date at the plant site and the forecasted quantities until the end of the units' design life assuming an optimum use of the fuel and conversion to high burnup fuel.

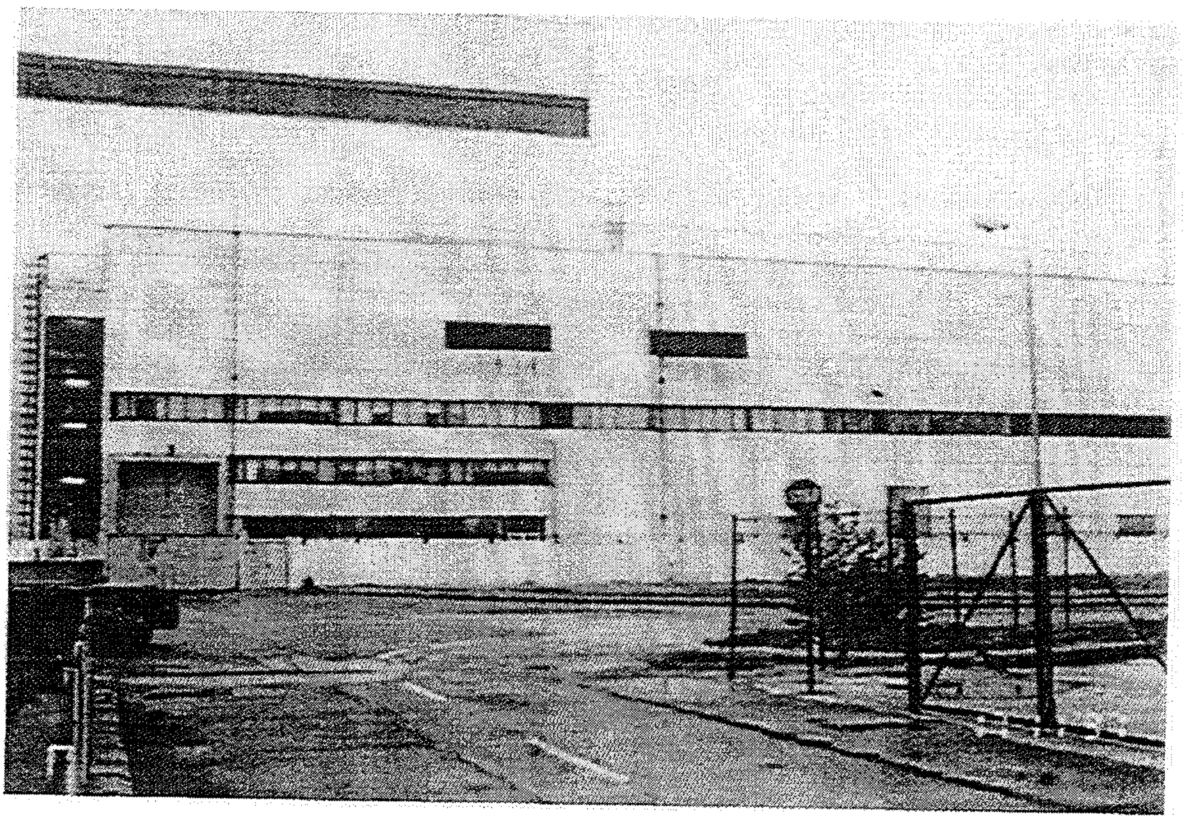

*FIG.1. On-site pool-type storage facility*

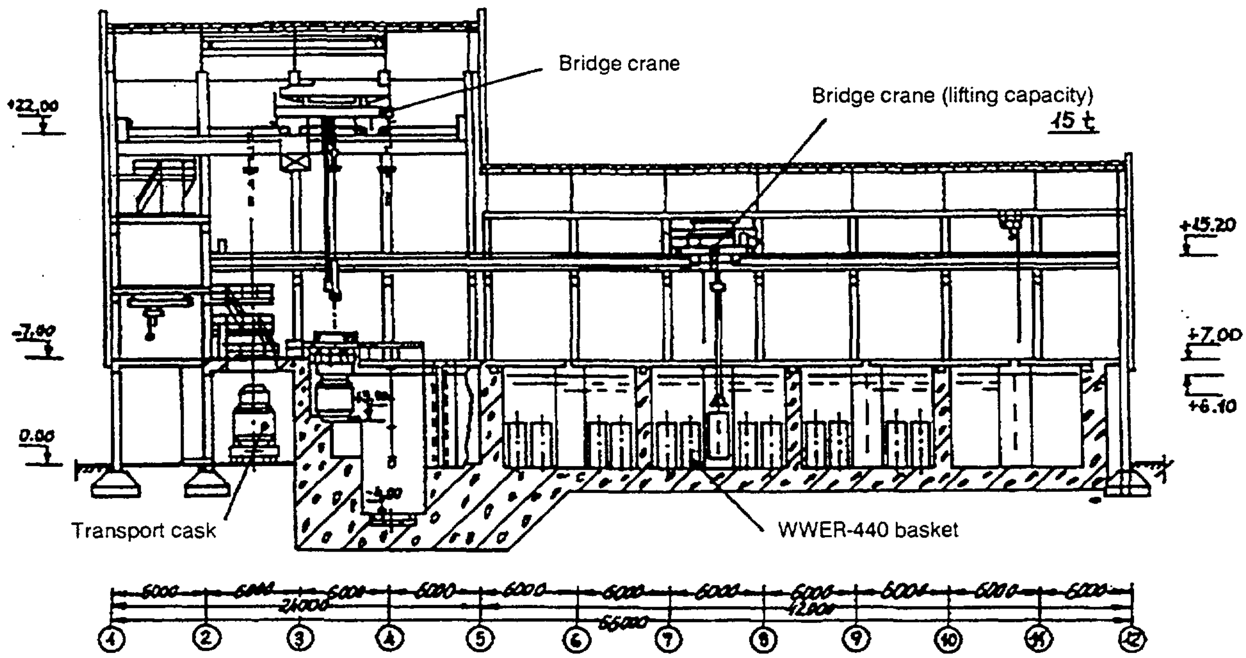


*FIG.2. Vertical cross section of the pool-type storage facility*

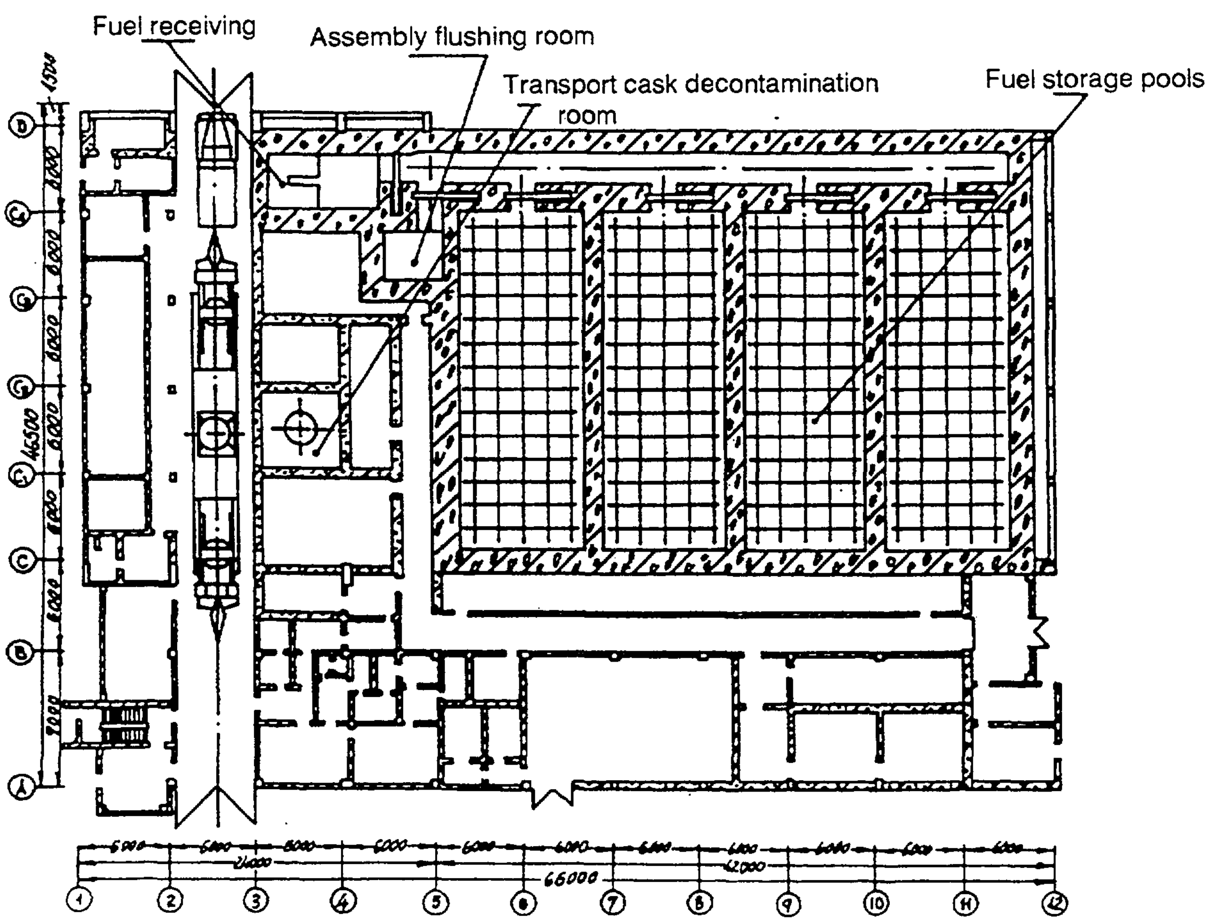


*FIG.3. Horizontal cross section of the pool-type storage facility*

## ¤ Operational Safety

- Protective layer of demineralized water - 2950 mm
- Distance between the baskets - 1600 mm
- Prevention of occasional water discharge
- $t \leq 45°C$
- Water purification system
- Leak detection system
- Remote temperature, water level, radiation measurement
- Decontamination equipment
- Transport in containers
- Two independent safety assessment reports

*FIG. 4. Operational safety features*

After facing the problem of licensing the additional storage facility and proceeding with transportation of fuel back to Russia at international prices, the management of the National Electric Company (NEK), which is the plant's operator, has tried to solve the problems by inviting bids for construction of a dry spent nuclear fuel storage facility. The bidding process is not finalised yet because of some organisational reasons, one of which is the absence of an agreed government strategy for the SNF management.

In order to engage the government of the country with this serious problem and meet the new requirement resulting from the conclusion of the Convention on "Safe spent nuclear fuel and radioactive waste management", the new management of NEK elaborated a "Policy on nuclear fuel cycle and radioactive waste management" which outlines the trends in this field till 2010 in prospect. This policy considers the prospects of development of the nuclear energy sector during the period when the first units of Kozloduy NPP will be decommissioned and replaced by new capacity.

Under the effective legislation in Bulgaria the spent nuclear fuel is not treated as waste as a result of which all activities related to its management in the future should provide for its re-extraction from reprocessing. This means that for the time being the country takes the so called suspended solution underlying the policy on its management.

The policy on spent nuclear fuel sets as primary objectives the requirements to use new, improved fuel of higher reliability and quality, prolonged service life and fuel burnup that would provide a possibility of maximum reduction of costs for storing the spent nuclear fuel on the plant site. This will result in asking for new manufacturers of fresh nuclear fuel in Russia and in the world.

Adoption of such policy by the national Government would allow for accelerated completion of the actions taken with regard to the construction of a new dry spent nuclear fuel storage facility on the plant site and starting the project for construction of a new national storage for the radioactive wastes from the spent nuclear fuel having been returned to Russia for reprocessing.

TABLE I. PROJECTED NUMBER OF DISCHARGED SPENT NUCLEAR FUEL ASSEMBLIES OVER THE DESIGN LIFE OF THE NPPs
(3-year fuel cycle)

| Year | VVER-440 | VVER-1000 |
|---|---|---|
| prior years and 1997 | 4,300 | 619 |
| 1998 | 415 | 132 |
| 1999 | 430 | 109 |
| 2000 | 429 | 108 |
| 2001 | 429 | 109 |
| 2002 | 430 | 109 |
| 2003 | 429 | 108 |
| 2004 | 429 | 109 |
| 2005 | 638 | 109 |
| 2006 | 533 | 108 |
| 2007 | 221 | 109 |
| 2008 | 221 | 109 |
| 2009 | 220 | 108 |
| 2010 | 221 | 109 |
| 2011 | 221 | 109 |
| 2012 | 429 | 108 |
| 2013 | 349 | 109 |
| 2014 | | 109 |
| 2015 | | 108 |
| 2016 | | 109 |
| 2017 | | 109 |
| 2018 | | 108 |
| 2019 | | 218 |
| 2020 | | 54 |
| 2021 | | 54 |
| 2022 | | 55 |
| 2023 | | 163 |
| **Total assemblies:** | **10,034** | **3,468** |
| **Total mass tHM** | **1,200** | **1,400** |

At present, action is being taken in three directions as follows:

- reconstruction of the additional storage facility of basin type in order to receive a licence from the National Regulatory Agency;
- maximum packing of the at-reactor basins of the first four units;
- return of limited quantities of spent nuclear fuel to Russia.

On the first direction, a project of EBRD through the Phare Programme is under way which is implemented by Bulgarian firms and its completion is planned by the end of the following year by getting a licence from the Bulgarian Regulatory Agency.

A decision on whether to pack the basins to the highest possible degree within this storage facility or to extend it with new adjacent basins is at hand.

On the second issue, a tender for a contractor who will pack the at-reactor basins of the first four units is in progress. The difficulties, we face, are only organisational since the plant management is not certain in the expedience of this work.

On the third direction, there is already a Framework Agreement for return of 480 assemblies from VVERs-440 to Russia in place and at the end of September the first shipment of 8 TK-6 canisters containing 240 assemblies will be transported. The canisters will be loaded on a barge following the existing transport route - Kozloduy NPP port, Reny port where the canisters will be reloaded on railway wagons and transported across the territories of Moldova and Ukraine to the reprocessing plants in Russia. Recently, with the establishment of Moldova and Ukraine as independent states, Bulgaria encounters serious difficulties in settling the problems with the transit of the nuclear materials via their territories. With regard to this, the policy sets as an objective to change the transport route envisaging delivery of SNF canisters to a sea port on the territory of Russia.

The economic analysis of the various options for implementing the new policy shows that the costs are within a range of 200-800 million USD. In case of maximum use of the at-reactor basin capacity of the first four units and of the additional on-site storage facility after their refurbishment and packing, the costs will be the lowest. Return of the total quantity of SNF from the first four units (10,166 assemblies) for reprocessing in Russia and construction of a new dry spent nuclear fuel storage facility for VVER-1000 will entail the greatest expense. The last option will be implemented only if the other options cannot be realised within the set time frame and temporary suspension of plant operation is enforced.

The policy does not consider an option of direct disposal of SNF, because it is regarded that there are no suitable geological formations on the Bulgarian territory where such a repository could be build. The fate of leaking SNF assemblies is not decided either and there is no decision taken for their return to Russia for reprocessing. Most probably they will have to be disposed as radioactive waste.

In conclusion one might say that it will be difficult to make up within the next year and a half before the new century for the time lost in the last 10 years when problems with the SNF were not resolved. Such speedy actions will require significant amount of financial and material resources which the plant does not have now and we hope that by approving the Policy on Nuclear Fuel Cycle and Radioactive Waste management developed by NEK, the Government will do everything in its power to settle these difficult problems.

IAEA-SM-352/3

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# SPENT FUEL MANAGEMENT IN CANADA

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Presented by M.J. CRIJNS

## 1. INTRODUCTION

The current status of the Canadian spent fuel storage methods is presented. This includes wet and dry interim storage. Extension of wet interim storage facilities is not planned, as dry storage technologies have found wide acceptance.

The Canadian nuclear programme is sustained by commercial CANDU type reactors, which have been in operation since 1971. Ontario Hydro's CANDU reactors, representing 13,600 MW(e) of installed capacity are capable of producing about 92,000 spent fuel bundles (1,800 tU) every year. Hydro Québec and New Brunswick Power each have a 685 MW(e) CANDU reactor, generating about 100 tU of spent fuel annually.

The typical CANDU fuel bundle contains zircaloy clad - natural uranium dioxide elements and weighs about 24 kg. Due to its relatively small size, its low burnup characteristics and the absence of criticality hazard in light water, CANDU fuel is managed relatively easily at the back end of its cycle compared with LWR fuel. The spent fuel bundles are kept in Reactor-Site (RS) interim storage facilities.

The implementation of various interim (wet and dry) storage technologies resulted in simple, dense and low cost systems. Table I provides a summary of the installed wet and dry storage capacities in Canada as of January 01 1998. A summary of the spent fuel arisings is given in Tables II and III.

Economical factors determined that the open cycle option be adopted for the CANDU reactors rather than recycling the spent fuel. Research and development activities for immobilization and final disposal of nuclear waste are being undertaken in the Canadian Nuclear Fuel Waste Management Programme.

## 2. CURRENT STATUS OF DRY SPENT FUEL STORAGE IN CANADA

Spent fuel originates from research reactors, decommissioned prototype CANDU reactors and operating commercial CANDU Generating Stations. Virtually all spent fuel is stored on site, either in primary/secondary reactor pools or in Away-From-Reactor (AFR) type dry storage facilities.

Between 1974 and 1989, concrete canisters (silos) were used by Atomic Energy of Canada Ltd. (AECL) to store the spent fuel generated by decommissioned research and prototype reactors. As the primary reactor pools of the commercial reactors filled up, secondary pools were built by the largest Canadian utility, Ontario Hydro. Subsequently, Ontario Hydro switched to dry storage technology rather than expanding its wet storage capacity.

TABLE I. SPENT FUEL STORAGE CAPACITIES AT CANADIAN NUCLEAR STATIONS

as of 1 January 1998

| Station Name | Installed Wet Storage Capacity (tU) | Installed Dry Storage Capacity (tU) |
|---|---|---|
| **Operating Stations** | | |
| Pickering 1 to 8 | 8,591 | ~ 1424 |
| Bruce 1 to 8 | 14,231 | - |
| Darlington 1 to 4 | 6,650 | - |
| Point Lepreau 1 | 930 | 1,026 |
| Gentilly 2 | 1,005 | 687 |
| Sub-Total | 31,407 | 3,137 |
| **Decommissioned Stations** | | |
| Whiteshell WR1 | - | 25 |
| Gentilly 1 | - | 67 |
| Douglas Point | - | 298 |
| Nuclear Power Demonstrator | - | 75 |
| Sub-Total | 31,407 | 465 |
| TOTAL | 31,407 | 3,602 |

Between 1985 and 1995 non-transportable silos were designed and built by AECL. Since 1995, vault and cask type storage methods have also been in use. The various dry storage methodologies used at present in Canada are summarized in Figure 1.

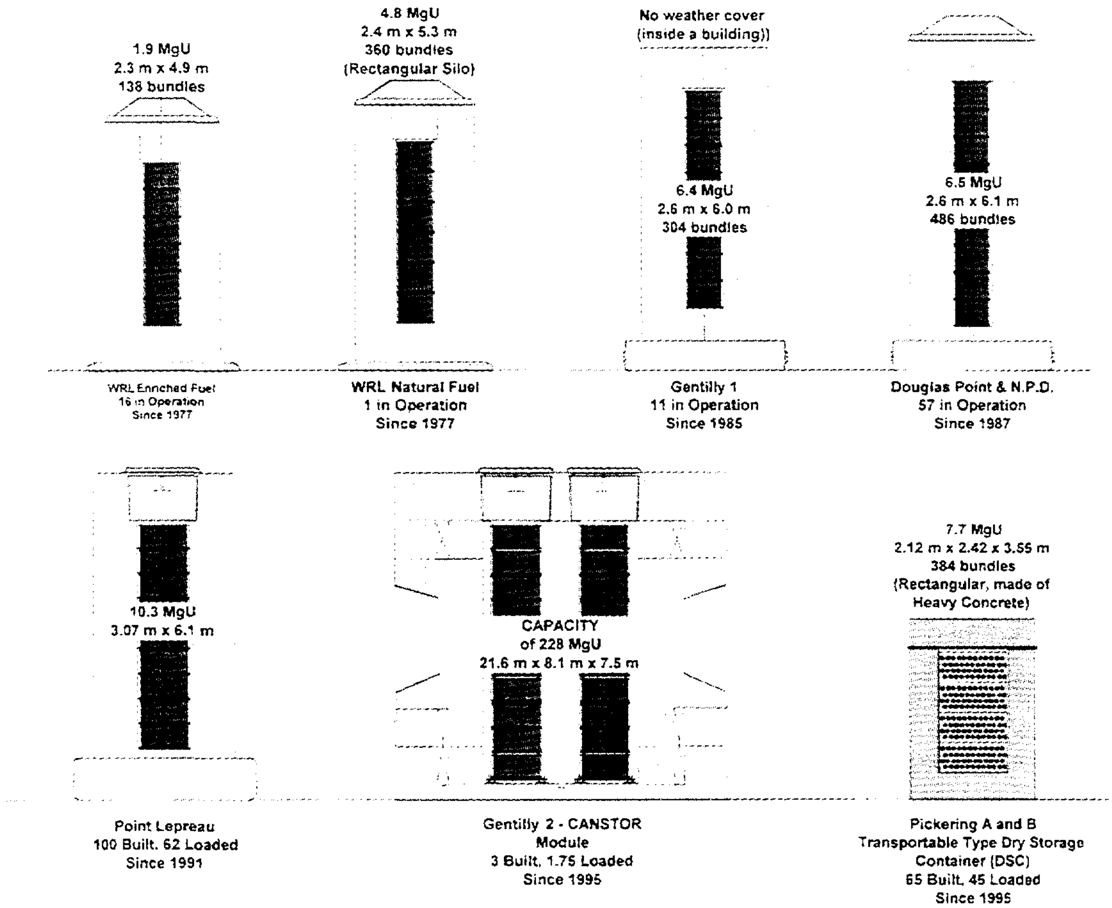


*FIG. 1. Canadian concrete silos, module and cask*
*(data as of December 1997)*

TABLE II. CANADIAN “AT REACTOR” WET SPENT FUEL STORAGE FACILITIES

(Status of wet storage facilities as of 1 January 1998)

| *Name of Facility* | *Owner and Location* | *Fuel Type (natural U)* | *Design Capacity (tU)* | *Current Inventory (tU)* | *Planned Operating Life* | *Operating Period* |
|---|---|---|---|---|---|---|
| Gentilly 2 primary pool | New-Brunswick Power /Point Lepreau New-Brunswick | CANDU 37 elements | Bundles: 52,896 (1,005 tU) | Bundles: ~ 50,000 (~ 950 tU) | 30 years | In operation since 1983 |
| Point Lepreau primary pool | Hydro-Québec/ Gentilly, Québec | CANDU 37 elements | Bundles: 48,960 (930 tU) | Bundles: ~ 40,000 (~ 760 tU) | 30 years | In operation since 1984 |
| Pickering A primary pool | Ontario-Hydro Pickering | CANDU 28 elements | Bundles: 78,978 (1,580 tU) | Bundles: 74,033 (1,484 tU) | 44 years | In operation since 1972 |
| Pickering A secondary pool | Ontario-Hydro Pickering | CANDU 28 elements | Bundles: 204,288 (4,094 tU) | Bundles: 193,500 (3,878 tU) | 37 years | In operation since 1978 |
| Pickering B primary pool | Ontario-Hydro Pickering | CANDU 28 elements | Bundles: 146,304 (2,932 tU) | Bundles: 146,000 (2,926 tU) | 43 years | In operation since 1983 |
| Bruce A primary pool | Ontario-Hydro Bruce | CANDU 37 elements | Bundles: 31,680 (609 tU) | Bundles: 23,544 (450 tU) | 50 years | In operation since 1977 |
| Bruce A secondary pool | Ontario-Hydro Bruce | CANDU 37 elements | Bundles: 340,992 (6,557 tU) | Bundles: 310,344 (5,930 tU) | 50 years | In operation since 1978 |
| Bruce B primary pool | Ontario-Hydro Bruce | CANDU 37 elements | Bundles: 42,432 (816 tU) | Bundles: 25,000 (480 tU) | 40 years | In operation since 1985 |
| Bruce B secondary pool | Ontario-Hydro Bruce | CANDU 37 elements | Bundles: 333,888 (6,420 tU) | Bundles: 232,590 (4,470 tU) | 40 years | In operation since 1987 |
| Darlington primary pool | Ontario-Hydro Bowmanville | CANDU 37 elements | Bundles: 350,000 (6,650 tU) | Bundles:~ 106,308 (2,044 tU) | 40 years | In operation since 1990 |

TABLE III. CANADIAN DRY SPENT FUEL STORAGE FACILITIES

(Status of dry storage facilities as of 1 January 1998)

| *Name of Facility* | *Location and Owner* | *Fuel Type* | *Design Capacity (tU)* | *Built Storage Capacity (tU)* | *Current Inventory (tU)* | *Planned Operating Life* | *Operating Period* |
|---|---|---|---|---|---|---|---|
| Whiteshell | Pinawa, Manitoba AECL | CANDU WR-1 (enr. and nat. U) | 24.6 (17 silos) | 24.6 (17 silos) | 24.6 (17 silos) | 50 years | 1977 to present |
| Gentilly 1 | Gentilly, Québec AECL | CANDU 18 elements | 67 (11 silos) | 67 (11 silos) | 67 (11 silos) | 50 years | 1985 to present |
| Douglas Point | Bruce, Ontario AECL | CANDU 19 elements | 298 (47 silos) | 298 (47 silos) | 298 (46 silos) | 50 years | 1987 to present |
| Chalk River | Chalk River, Ontario AECL | CANDU/NPD 19 elements (enr. and nat. U) | 75 (12 silos) | 75 (12 silos) | 75 (11 silos) | 50 years | 1989 to present |
| Point Lepreau | Point Lepreau, New-Brunswick NBP | CANDU 37 elements | 3,078 (300 silos) | 1,026 (100 silos) | 636 (62 silos) | 50 years | 1991 to present |
| Gentilly 2 | Gentilly, Québec Hydro-Québec | CANDU 37 elements | 3,648 (16 modules) | 684 (3 modules) | 401 (1.75 modules) | 50 years | 1995 to present |
| Pickering-Phase 1 | Pickering, Ontario Ontario-Hydro | CANDU 28 elements | 5,390 (700 DSCs) | ~ 1,424 (~ 185 DSCs) | ~ 585 (76 DSCs) | 50 years | Stage I Licensed, oper. since Jan./96 |
| Pickering-Phase 2 | Pickering, Ontario Ontario-Hydro | CANDU 28 elements | 6,160 (800 DSCs) | 0 | 0 | 50 years | To be determined |
| Bruce | Bruce, Ontario-Hydro | CANDU 37 elements | 14,900 | 0 | 0 | 50 years | Currently being designed/licensed |
| Note: All facilities are of AFR (RS) type except for Chalk River which is an AFR (OS) facility | | | | | | | |

[1] *Natural U unless specified*

## 3. SPENT FUEL STORAGE AT ONTARIO HYDRO'S STATIONS

### 3.1. Pickering Nuclear Generating Station

The fuel bays at Ontario Hydro's Pickering Nuclear Generating Station filled close to capacity in 1995. A Used Fuel Dry Storage Facility (UFDSF) has, therefore, been constructed on-site to transfer 10 year-cooled and older fuel from the fuel bays to this facility. The UFDSF is an indoor facility which houses a workshop and a storage area. Spent fuel is stored in the storage area, in steel-lined concrete containers called Dry Storage Containers (DSCs). Stage I of the facility has been operational since January 1996 and is designed to store 185 DSCs (~ 1,424 tU). Stage II of the facility is undergoing licensing at present and is expected to be in service by July 2000. Stage II will accommodate 500 DSCs (~ 3,750 tU).

The Pickering DSC, illustrated in Figure 2, has a capacity for 384 spent fuel bundles. The container is wet-loaded in the fuel bay, drained, vacuum-dried, back-filled with helium, seal-welded, and helium leak tested before placement in the storage area. The UFDSF has 76 DSCs stored to date. No surface contamination has been detected on any DSC and dose rates have been found to be less by a factor of 4-6 than those predicted, mainly because of a higher than minimum specified concrete density.

The following technical issues emerged during the development and licensing of the Pickering DSCs: fuel integrity; retention of helium over the DSC's design life; weld integrity; facility temperatures; long-term durability. Further studies and analyses resolved most of these issues.

To resolve the thermal issues, an experimental thermal performance verification programme was committed as a license condition. This programme consisted of monitoring the thermal performance of an instrumented DSC loaded with 6 year old fuel over a 3 month period in summer. The programme has been carried out during summer 1998 and has demonstrated the actual temperatures for the DSC inner and outer liners to be well below the predicted temperatures.

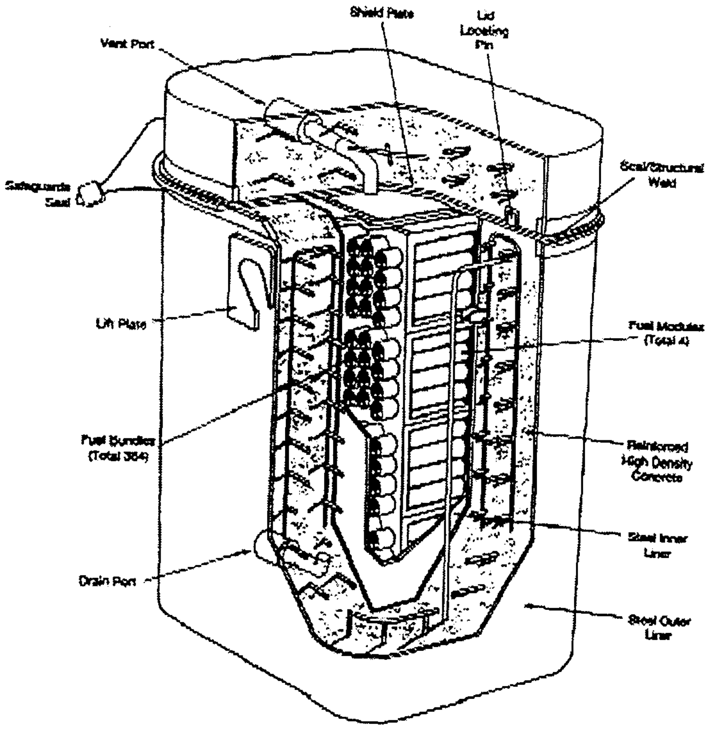


*FIG. 2. Pickering dry storage container (DSC)*

### 3.2. Bruce Nuclear Generating Stations

A dry storage programme has also been developed to meet the additional spent fuel storage needs of Bruce NGS A and B. A design had earlier been developed to meet the storage needs of Bruce A, but since the lay-up of Bruce A recently, the focus has shifted towards meeting the storage needs of Bruce B. The required in-service date for Bruce Used Fuel Dry Storage Facility is 2002.

As part of the licensing process initiated in 1997 to construct Bruce UFDSF, an Environmental Assessment (EA) was carried out under the Canadian Environmental Assessment Act. EA submissions were made to the Atomic Energy Control Board (AECB) in 1997 and 1998. A decision on the acceptability of this EA is expected from the federal authorities later this year. Meanwhile, based on a system design study conducted for different options, a suitable design option has been selected. The design will be based on wet-loading of Pickering type DSCs, as opposed to dry transfer of spent fuel from the fuel bays into larger 600-bundle capacity DSCs, contemplated in the earlier version of the system design in 1997.

A smaller third bay will be built next to the secondary bay at Bruce B for wet-loading and decontaminating a Pickering type DSC. This new bay will be connected to the existing bay by an interface. The container will be drained, vacuum dried, clamped, safeguarded and transported to the UFDSF. The UFDSF will be an indoor facility with provisions to process the container for storage. The DSC will be seal-welded, back-filled with helium, helium leak tested, safeguarded and stored in an indoor storage area. The transporter will be functionally similar to the one currently in use at Pickering.

Currently, work is underway to obtain the construction approval for Bruce UFDSF and the necessary approvals for the required station modifications.

### 3.3. Darlington Nuclear Generating Station

Darlington station is not expected to require additional fuel storage capacity until 2004. There are no dry storage plans currently under development at present for Darlington station.

## 4. SINGLE UNIT CANDU REACTORS

### 4.1. The Gentilly 2 Experience

Dry storage at Hydro Québec's Gentilly 2 Nuclear Generating Station is the first nuclear project to be submitted for review under the provincial environmental regulations in Canada. Public hearings were held to assess the project and their success was crucial to the acceptance of the project.

The CANDU version of the MACSTOR module, being the first of its kind to be built in Canada or elsewhere, became the subject of a complete review by the Atomic Energy Control Board (AECB). All licensing activities involving the Québec's Ministry of Environment (MENVIQ) and the AECB (including the Federal Environmental Review Assessment Office - FEARO) during all phases of design, construction, commissioning and operation have been successfully executed.

In October 1995, the first vault type storage concept was implemented at Gentilly 2 on the basis of the MACSTOR (Modular Air-cooled Canister Storage) technology. Its application to store CANDU fuel resulted in the building of the CANSTOR version of the storage module. This concrete vault like structure stores 12,000 bundles, which represents more than 2 years of reactor operation. The obtained economy of scale reduces the annual operating expenditures. The storage system also sharply economizes on storage area requirements. This is particularly useful for relatively small single reactor sites such as that available at Gentilly 2.

The implementation of a dry storage system for CANDU fuel requires more fuel handling equipment than for an equivalent LWR system. This is due to the necessity to package the fuel bundles in a storage basket that is sealed in a shielded work station (hot cell) prior to transfer into dry storage. The module design is such that none of the already developed site equipment requires modification ( gantry crane, transfer flask etc.). The methodology of fuel handling for the CANSTOR system remained similar (refer to Figure 3) and all safety and mechanical features vital to the existing licensed concrete canister storage system were retained.
Project Milestones.

- Contract signed with AECL in 1993. The facility became operational within 30 months in September 1995;
- The licensing and public hearing process took 18 months. It required a sustained effort from both Hydro Québec and AECL;
- The storage site construction including the first module, pool area modifications, equipment installation and commissioning was completed in 7 months;
- Second Module was built and 50 % loaded in 1996.
- Third Module was completed in 1997

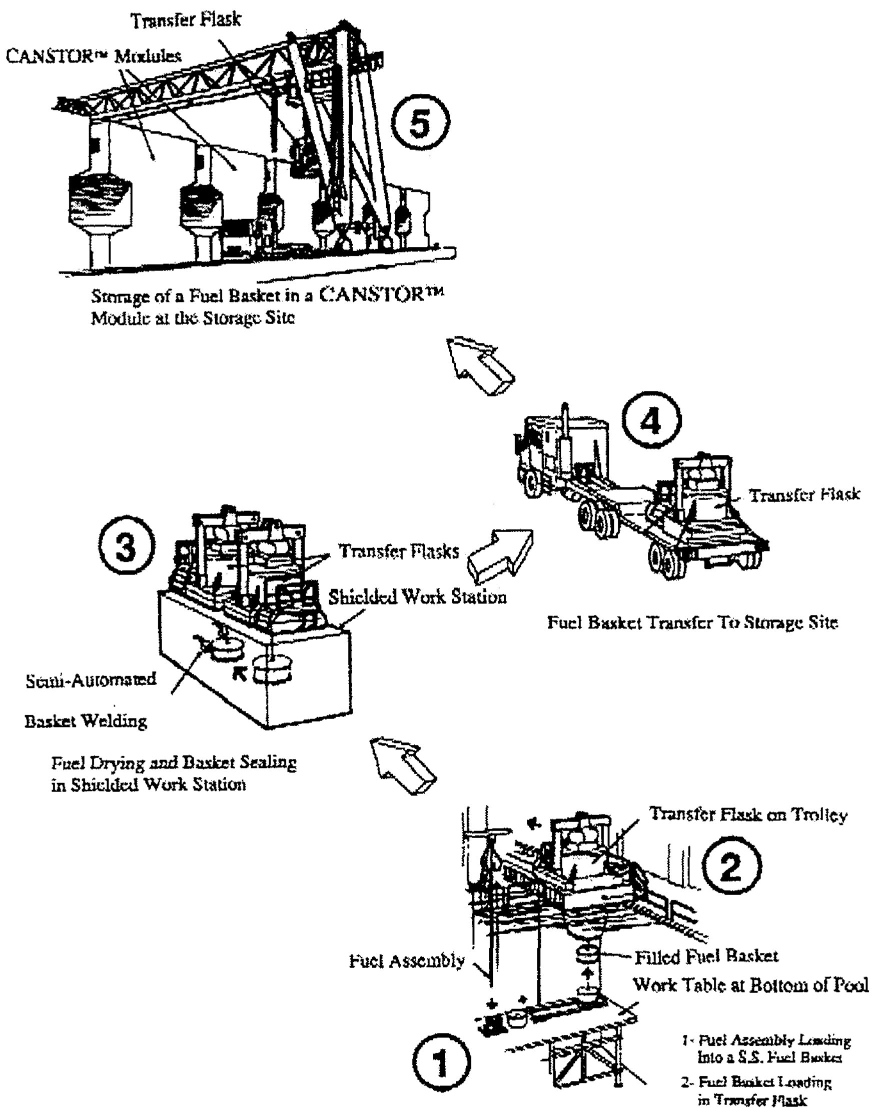


*FIG. 3. Fuel handling operations for dry storage at Gentilly 2*

At the Gentilly 2 plant, Hydro-Québec will continue the transfer of spent fuel to its MACSTOR modules. Plans are to build two new modules in 1999, bringing the installed capacity to 1,140 tU.

### 4.2. Point Lepreau Dry Storage Programme

At the Point Lepreau plant, New Brunswick Electric Power Commission will proceed with its annual dry storage campaign. The existing dry storage facility using concrete canisters is sufficient for three years of operation. Plans for capacity extension, with concrete canisters or MACSTOR modules, are expected to be developed next year.

IAEA-SM-352/4

# CZECH INTERIM SPENT FUEL STORAGE FACILITY: OPERATION EXPERIENCE, INSPECTIONS AND FUTURE PLANS



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**Abstract**

The paper describes the situation in the spent fuel management in the Czech Republic. The interim Spent Fuel Storage Facility (ISFSF) at Dukovany, which was commissioned in January 1997 and is using dual - transport and storage CASTOR - 440/84 casks, is briefly described. The authors deal with their experience in operating and inspecting the ISFSF Dukovany. The structure of the basic safety document "Limits and Conditions of Normal Operation" is also mentioned, including the experience of the performance. The inspection activities focused on permanent checking of the leaktightness of the CASTOR 440/84 casks, the maximum cask temperature and inspections monitoring both the neutron and gamma dose rate as well as the surface contamination. The results of the inspections are mentioned in the presentation as well. The operator's experience with re-opening partly loaded and already dried CASTOR-440/84 cask, after its transport from NPP Jaslovské Bohunice to the NPP Dukovany is also described. The paper introduces briefly the concept of future spent fuel storage both from the NPP Dukovany and the NPP Temelín, as prepared by the ČEZ. The preparatory work for the Central Interim Spent Nuclear Fuel Storage Facility (CISFSF) in the Czech Republic and the information concerning the planned storage technology for this facility is discussed in the paper as well. The authors describe the site selection process and the preparatory steps concerning new spent fuel facility construction including the Environmental Impact Assessment studies.

## 1. INTRODUCTION

Fresh nuclear fuel was supplied to the former Czechoslovakia from the previous U.S.S.R. and spent fuel was originally planned to be re-exported after a 5 year cooling period. The transport of spent nuclear fuel from the Czechoslovak NPPs was suspended in 1988 and the spent nuclear fuel assemblies from the NPP Dukovany were (after a 3 year cooling period) transported to the wet interim storage facility at Jaslovské Bohunice in the Slovakia. These transports were halted in 1992.

The necessity to handle a growing number of spent fuel assemblies in the NPP Dukovany and the need to transport the 1,176 spent fuel assemblies stored at Jaslovské Bohunice (Slovakia) back to the Czech Republic, have led to the construction of an away-from-reactor spent fuel storage facility at Dukovany. The Interim Spent Fuel Storage Facility (ISFSF) at Dukovany was commissioned in January 1997 and uses dual - transport and storage CASTOR - 440/84 casks (see the Figures 1 and 2). The transport of the loaded cask into the storage building is shown in the Figure 3.

## 2. STORAGE OPERATION, ITS EVALUATION AND INSPECTION

The process of ISFSF commissioning brought neither major problems nor surprisingly new experience, more details can be found in [1]. There were only two issues connected with the original design, which had to be solved before permanent operation of the storage was approved. The first

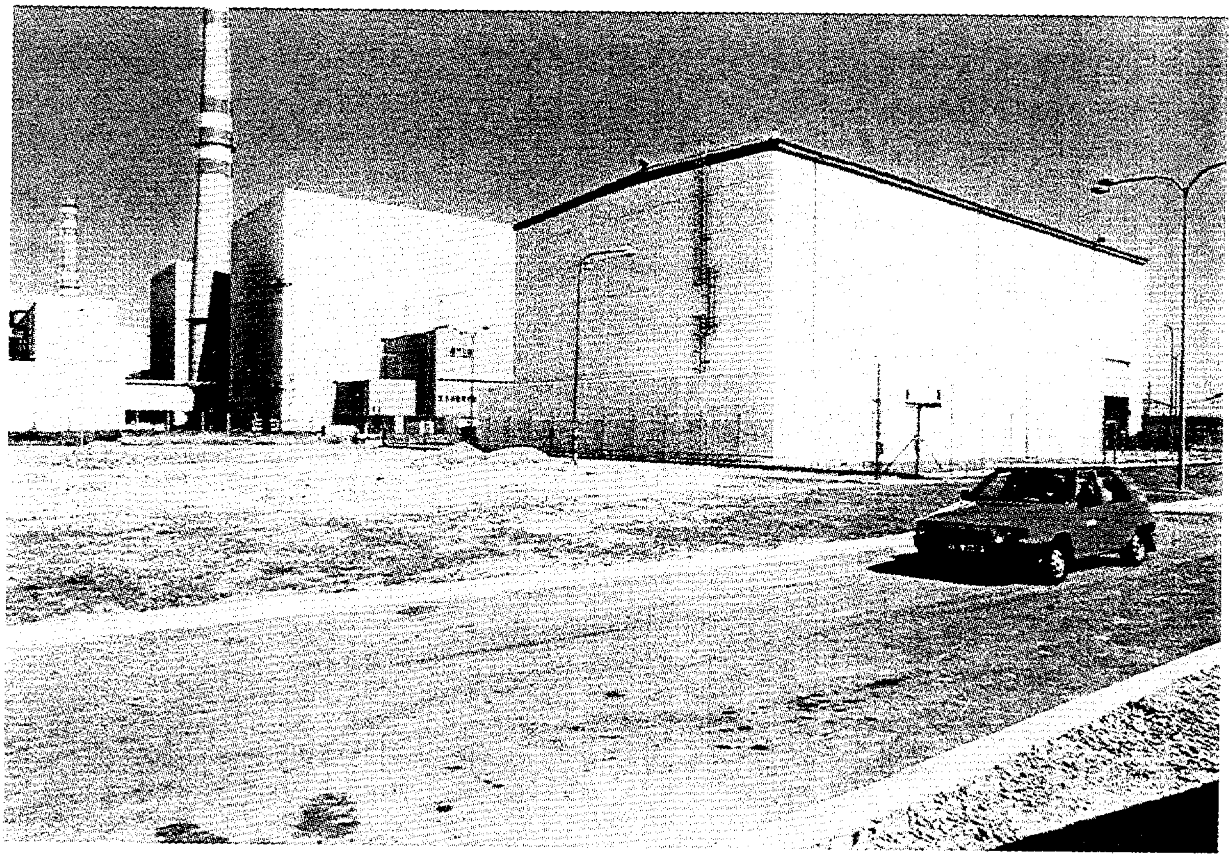

*FIG. 1. Interim Spent Fuel Storage Facility at Dukovany*

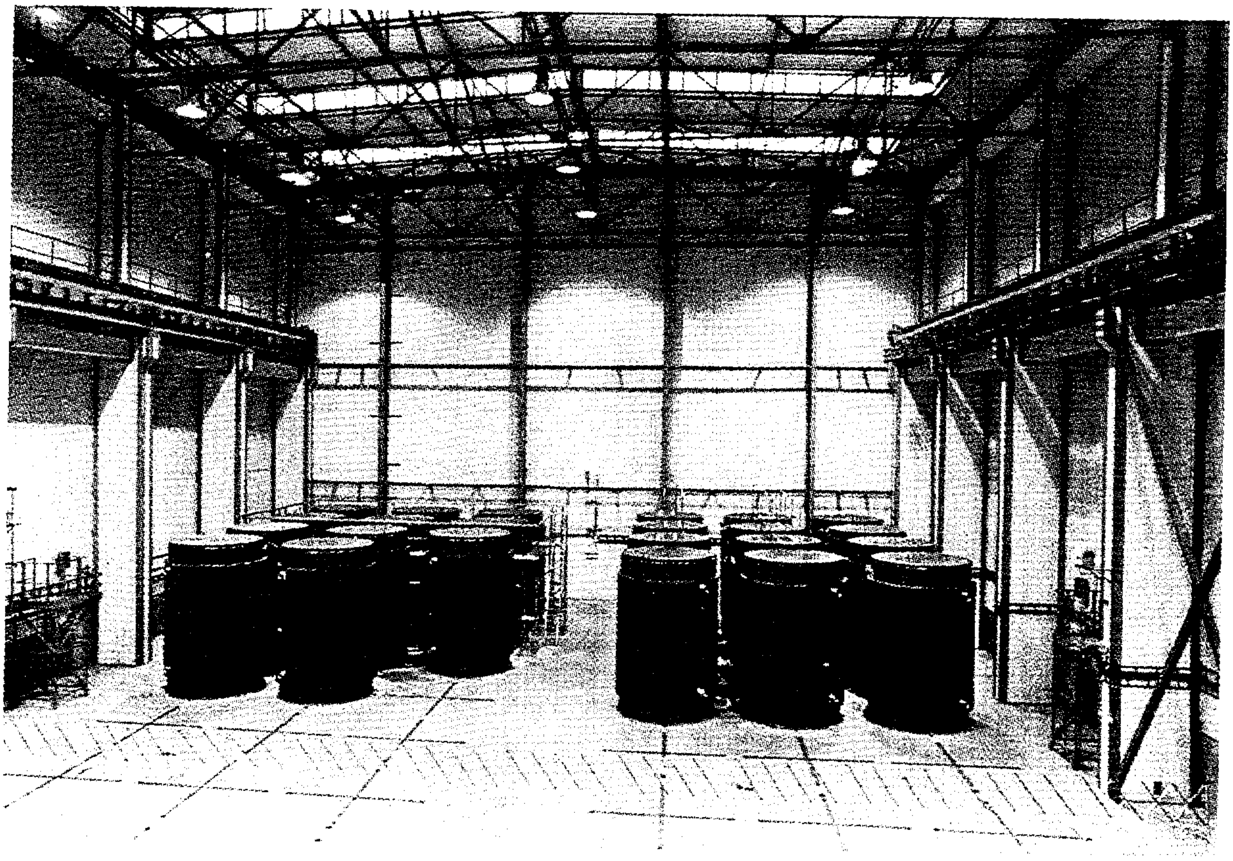

*FIG. 2. Inside view of the Interim Spent Fuel Storage Facility*

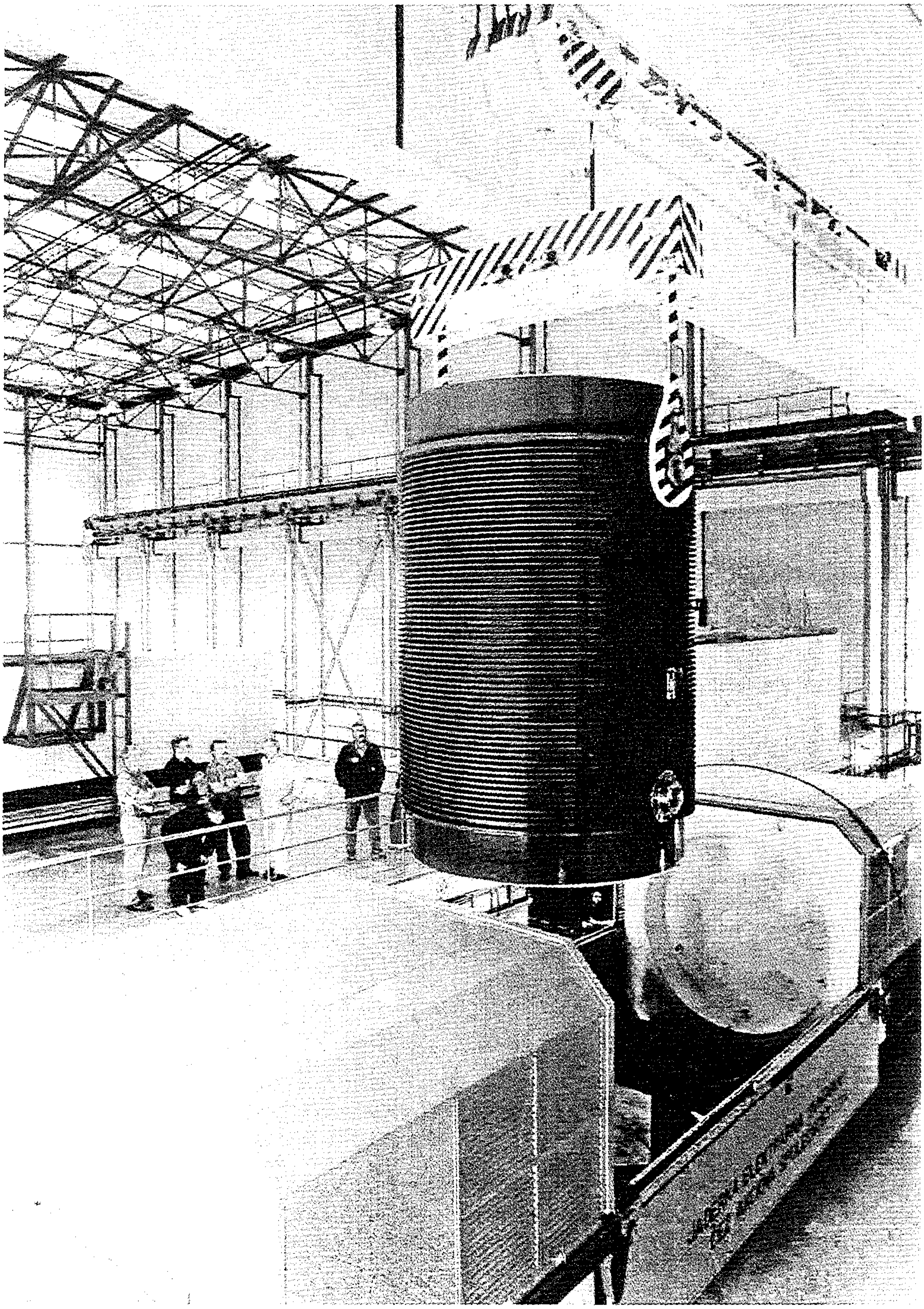

FIG. 3. Cask transfer into the storage building

(minor) one was insufficient thermal insulation of a wastewater tank collecting wastewater from staff's showers, causing the real risk of the water freeze in the tank during the winter time. Other consequences of unexpectedly low temperature in the storage hall in the winter period were snowdrifts, glaze and slides in the hall, complicating the working conditions of the storage staff. While the former problem was quite easily overcome before commissioning ISFSF Dukovany, by insulating the tank and equipping it by its own heating, the final solution of the latter took almost two years. The experiment took a year (mostly during the trial operation) when the jealousies in the input openings were practically completely blocked. The temperature of defined points on each cask surface was continuously monitored in this period as well as the air temperatures in the storage hall and outside.

Parallel to the measurements in the storage, calculations of the thermal removal from the casks in the storage were prepared by the operator in April 1997 [2]. The calculations took into an account the quantity of the loaded casks in the storage hall in given period and conservatively considered each cask loaded to its maximum allowed thermal output of 21 kW. For example, in December 1997 there were 23 casks with a total thermal output about of 197 kW, while the model was calculating with 483 kW. In October 1998, there were 28 casks with total thermal output about 248 kW. The outside temperature and the area of the vent openings were other imputes of the calculation model, which calculated the outgoing air temperature. Both, the results of calculations and experiment, supported the request of the operator to approve two regimes of the jealousies in the input openings - open in the summer and practically closed in the winter period. These design changes have been accepted by the SÚJB and several conditions concerning the way of monitoring temperatures of the casks surface and the temperature of the cooling air have been set in order to assure timely corrective measures - if necessary.

Above mentioned issues led to two changes in the original storage design. Both changes were carefully examined by the regulatory body and consulted with the vendor. The experience gained until today has confirmed the safety of the changes and was a useful lesson in understanding better some potential problems of a conservative approach in the storage design.

During more than one year trial operation of the ISFSF, the basic safety document "Limits and Conditions of Normal Operation" gradually developed to its recent structure, following the operational experience, results of inspections and independent research. It is necessary to stress, that because of the experience gained during consultations with colleagues from the US NRC and the application of the methodical guidance in the IAEA Practice on Safety Operation of Spent Fuel Storage Facilities [3], the first version of the Limits and Conditions of Normal Operation was basically adequate. The modifications which followed, reflected either above mentioned design changes or were motivated by the effort to express more clearly or simply the obligations of the storage staff. The recent version of Limits and Conditions of Normal Operation of ISFSF Dukovany has the structure given in Table I.

The SÚJB planned inspection activity has been mainly concentrated on the performance of Limits and Conditions of Normal Operation and especially on the performance of the storage monitoring systems. Each inspection tested the function of the leaktightness monitoring system (measuring the pressure of Helium between primary and secondary cask lids), by simulating the pressure drop or by the disconnecting the transfer cables. Both the SÚJB inspections and operator experience with the pressure monitoring system have not indicated any real problem. In the year 1997, one drop out of data transfer (lasting less than three hours and concerning both pressure and radiation situation monitoring systems) from the storage to the central dosimetry control room occurred.

Regular operator and SÚJB attendance has also been devoted to the inspections of the maximum cask temperature, neutron and gamma dose rate, as well as results of the surface contamination of the loaded casks. After evaluating the results of continuous measurements during the trial period, when the surface temperature of each cask was monitored. Taking into an account the results of monitoring the process of establishing the cask surface temperature equilibrium in the period from spent fuel loading into the CASTOR-440/84 casks to final - third lid attachment the SÚJB requirements on the cask surface temperature monitoring were defined. Recently, continuous temperature monitoring of only the three hottest casks is required. These casks are selected on the basis of their thermal output, however, their position among the cask array has also a slight influence on their surface temperature.

TABLE I. THE STRUCTURE OF THE SAFETY DOCUMENT "THE LIMITS AND CONDITIONS OF NORMAL OPERATION" OF ISFSF DUKOVANY

| **Area** | **"Limits and Conditions of Normal Operations" focuses on** |
|---|---|
| Residual heat removal | ▪ maximum capacity of the storage<br>▪ geometry of the cask positioning<br>▪ minimum cooling period before loading the SF assemblies into the cask (storage module)<br>▪ maximum allowable burnup, maximum enrichment, maximum residual heat generation per assembly<br>▪ maximum cask surface temperature<br>▪ availability of the cooling system |
| Radiation protection/containment | ▪ maximum allowable burnup, maximum enrichment, allowed geometry of SFAs loading<br>▪ minimum cooling period before loading the SF assemblies into the cask (storage module)<br>▪ maximum allowable dose rate on the surface and in a defined distance from the cask surface (1-2m)<br>▪ maximum allowable surface contamination<br>▪ radiation monitoring system<br>▪ leaktightness monitoring system |
| Subcriticality | ▪ maximum fuel enrichment<br>▪ fuel assembly characteristic<br>▪ licensed cask only |
| Retrievebility | ▪ licensed cask only<br>▪ standard handling tools only |
| Other | ▪ obligation of timely information concerning any breaking of approved limits to the regulatory body |

In 1997, the thermal output of the casks varied from 12 to 13 kW, their maximum temperature did not exceed 60°C, while the maximum cooling air - (input) temperature was 24.5°C (recorded in June 1997). The temperature of selected casks was measured in the middle of their sides. The operator also measured the other casks but the frequency of these measurements was only once a week. The standard uncertainty of the temperature measurements represents 0.5°C.

Systematic monitoring of the radiation situation in the storage facility and its neighbourhood belongs among the most serious tasks of the operator. The extent of the monitoring is as follows:

- Dose rate, at the cask surface, non-fixed contamination of the cask surface;
- Dose rate, non-fixed contamination of the place in selected areas of ISFSF building;
- Radiation monitoring of personnel, collective and maximum individual equivalent dose;

- Radiation monitoring of ISFSF surrounding (NPP area), equivalent dose rate, surface activity ($^{58}Co$, $^{60}Co$, $^{110}Ag$, $^{134}Cs$, $^{137}Cs$ and others), volume activity of underground waters ($^{3}H$, $^{60}Co$, $^{90}Sr$, $^{110}Ag$, $^{137}Cs$ and others), aerosol activity ($^{51}Cr$, $^{54}Mn$, $^{60}Co$, $^{110}Ag$, $^{134}Cs$, $^{137}Cs$ and others)
- Radiation monitoring of neighbouring villages: equivalent dose rate.

All the results confirmed that values of the dose rates inside the ISFSF are within the approved limits and the dose rates in the storage facility surrounding comparable with the background dose rates. Consequently, the contribution of the interim storage facility to the radiation level at the Dukovany site is negligible.

The data describing the radiation situation are carefully recorded and regularly inspected by the SÚJB inspectors. In addition, systematic measurements were carried out by experts of Faculty of Nuclear Engineering of Czech Technical University in the years 1996 and 1997 on the basis of a SÚJB contract. The measurements discovered, among others, an interesting contribution of neutron flux to the integral equivalent dose rate. As a consequence of this finding, the regulatory body requested the introduction of personnel neutron dosimetry. Each member of the storage staff is obliged to have (his/her) own personal neutron dosimeter while working inside the ISFSF.

Initiated by the loud international public discussions concerning problems with cask surface contamination in some European countries, a thorough operator inspection of the loaded casks surface contamination was carried out in July 1998. The NPP staff used for these inspections more measurement points than for the routine inspections and the SÚJB inspectors examined selected casks. The maximum discovered non-fixed contamination on the CASTOR-440/84 cask surface averaged over 300 cm$^2$ has been within the limits and did not exceed 4 Bq/cm$^2$ for beta and gamma emitters and low toxicity alpha emitters, or 0.4 Bq/cm$^2$ for all other alpha emitters.

In spite of the fact that no significant problem have occurred since the beginning of the ISFSF Dukovany operation the SÚJB required the operator's ability to re-open already loaded casks. Such operation demonstrates the ability to repair a failed cask as well as the possibility to move the fuel from the storage cask to other place.

Based on a proposal of the Gesellschaft für Nuklear Behälter mbH (GNB), the operator developed a procedure for back filling the dried cask already loaded with spent fuel with water. At the beginning of 1997, GNB sent equipment and a manual for cask handling under these circumstances to the NPP Dukovany. The GNB documents had to be modified to the needs of the NPP Dukovany.

The first operation with the new device and procedure was realised at the end of 1997. During the planning of spent fuel shipment from Slovakia back to the NPP Dukovany, it was known that one of the casks would not be filled to its full capacity wih spent fuel. This situation led the operator to the decision to use newly installed equipment for reloading this cask. The last cask containing 48 spent fuel assemblies was transported to the reactor hall where was opened. The second lid was removed and directly at the cask handling point, the equipment for back filling with water was used for filling and cooling of the partially loaded container. The cask was filled with additional 36 spent fuel assemblies and further handled by the routine procedure and transported to the ISFSF. These activities were carried out on 6 November 1997, in the presence of the SÚJB inspectors. The handling was carried out in 10 hours without problems and the equipment was used without difficulties. An opened and empty CASTOR-440/84 cask can be seen in Figure 4.

## 3. PREPARATION OF FUTURE SPENT FUEL STORAGE FACILITIES

The Czech utility – Czech Energy Board (ČEZ) started its activity focused on the Central Interim Spent Fuel Storage Facility (CISFSF) in 1992. The utility top management decided to use the dry storage technology as the storage option. It was also decided to study both the surface and near surface variants of the storage facilities. The starting position for this task was as follows:

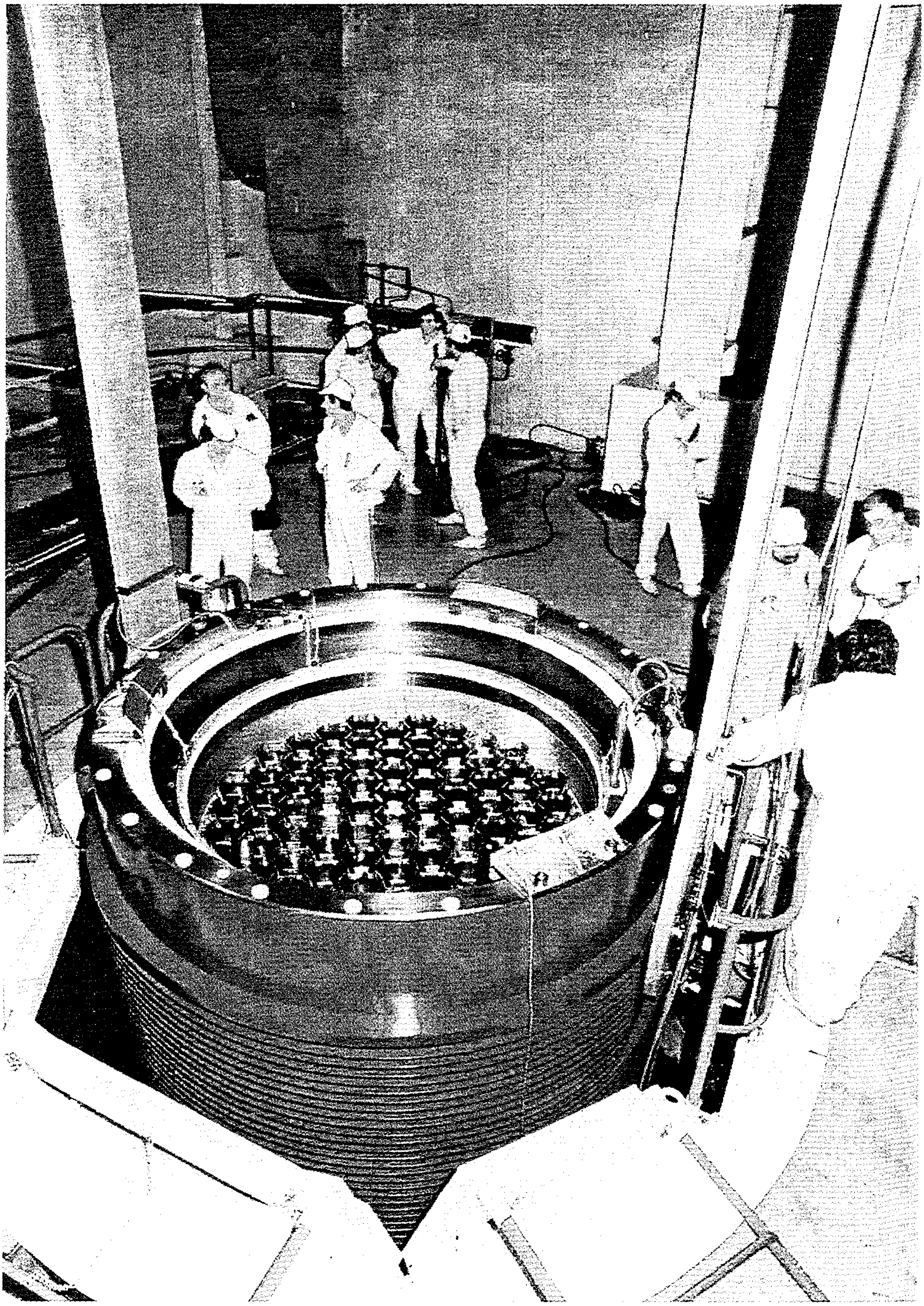

*FIG. 4. Open CASTOR-440/84 cask*

- Government Resolution No. 213 limited the capacity of the ISFSF Dukovany to 600 t of heavy metal;
- The ČEZ top management did not agree with locating the CISFSF on the NPP Temelín site, which was under construction;
- The promise of ČEZ, given to the public during the ISFSF Dukovany public hearing, to examine the possibility of siting a new storage facility inside the whole territory of the Czech Republic;
- The capacity of the new storage should be sufficient for the spent fuel generated during the lifetime both of the Temelín and Dukovany NPPs;
- The new storage facility should be available in 2005, when the capacity of the ISFSF Dukovany is exhausted.

### 3.1. Bidding process

The utility started a preliminary bid for the storage technology in 1995. Sixteen potential vendors were requested to participate in the bid. The following companies answered and sent their proposals: AECL CANDU (Canada), DIAMO (Czech Republic), GEC ALSTHOM (Great Britain), VOEST ALPINE/NAC (Austria/USA), NUKEM (Germany), SGN (France), BNFL/SIERRA NUC (Great Britain Great, USA), ŠKODA (Czech Republic), TRANSNUCLEAIRE (France).

A broad spectrum of storage technologies were presented by the above mentioned vendors. All major existing dry spent fuel storage technologies were presented: dual purpose (storage and transport) casks, vault type storage, concrete modular storage and storage containers. The evaluations of the tender were independently carried out in the Czech Republic, Spain and Belgium. The dual purpose (transport and storage) cask was selected. Four potential vendors have been shortlisted: NUKEM, ŠKODA, TRANSNUCLEAIRE and VOEST ALPINE/NAC.

### 3.2. Site selection process

The process of the site selection consisted of two steps. The first step was based on comparing the archived data describing the Czech Republic territory with the legislative criteria valid for nuclear facility siting. This search identified 10 appropriate or conditionally appropriate sites. The second step consisted of the field investigations in the sites. The most prospective sites underwent a detailed geological investigation. At the Skalka site (see Figure 5) were about 750m of corridors excavated (see Figure 6). Two IAEA missions have been involved in the evaluation of the site selection process. The described site selection process resulted in selection of the following three sites:

- Skalka – near surface (underground) storage;
- Dolní Cerekev – surface storage;
- Batelov – both possibilities.

### 3.3. New Realities in 1996

In 1996, the Ministry of Industry and Trade started discussions about some parts of the Government Resolution No. 213 concerning the Dukovany 600 t storage limit. Under a Government's decision, a study of five different variants of possible sites for spent fuel storage facilities was performed including the NPP sites. The safety, technology, ecological and economical evaluation resulted in the five following variants:

- storage facilities at Dukovany and Temelin NPPs;
- a central storage facility at Skalka;
- a central storage facility at Temelin;
- a central storage facility at Dukovany;
- a central storage facility at Batelov.


*FIG. 5. Skalka site*

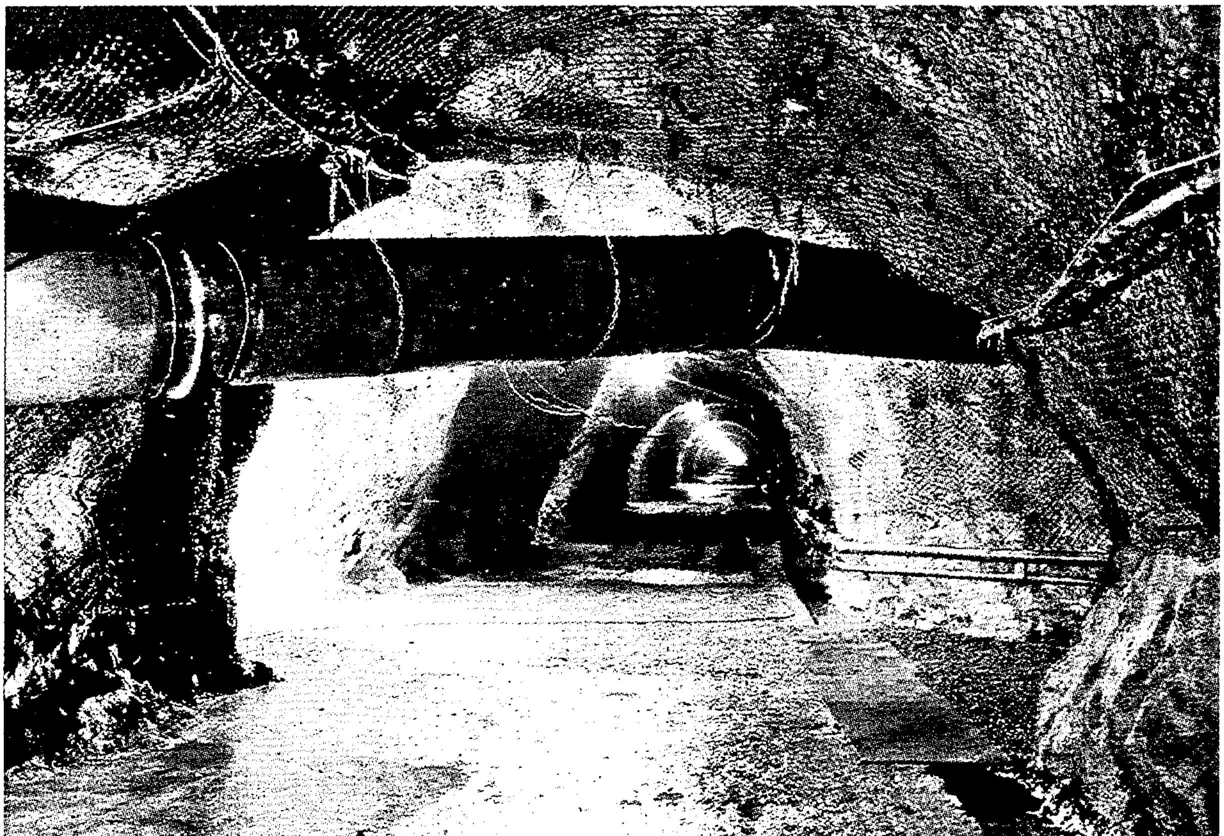

*FIG. 6. Excavated corridors at Skalka*

### 3.4. Present Situation

In March 1997, the Government has cancelled some parts of Government Resolution No. 213 concerning the Dukovany 600 t storage limit and recommended to prepare new storage capacities at the Dukovany NPP and Temelín NPP sites as priority sites and at Skalka (under ground alternative) as a stand by site.

ČEZ has prepared a feasibility study for the priority and stand by sites in the first half of 1997. The content and scope of the additional information requested from the bidders for the purpose of the Environment Impact Assessment (EIA) Study, the Safety Report and the Site Permit Documentation for Dukovany and Skalka site was done subsequently.

The EIA Study, Safety Report and Site Permit Documentation for both the Dukovany (main option) and Skalka sites (back up option) were performed in 1998. After their publication, some comments have been collected and the Ministry of Environment requested ČEZ to amend the reports. Recently, the EIA process was done and the modified versions of both EIAs have been published. In August 1998, the siting analysis reports of both the Dukovany and Skalka sites were submitted to SÚJB together with the request for siting approval.

According to the existing Czech legislative principles, the Environmental Impact Assessment under the supervision of the Ministry of Environment and the siting approval to be issued by SÚJB, are necessary preconditions for official starting the siting process.

### 3.5. Future steps

In order to minimise the risk in case of failure of the utility plan concerning timely commissioning of the new storage facility, the ČEZ prepares two parallel storage facilities one at the Dukovany and one at the Skalka site. The next steps for preparation of the above mentioned projects will depend on the development of the real situation.

The ČEZ management supposes that the parallel preparation of both above described projects will continue with minimal efforts unless the siting permits are issued. After that the top management will decide whether to accept the risk connected with the Skalka (back up) project preservation or to continue until constructions permits for both sites are issued. When the construction permit for the Skalka site is issued, the utility will not start the construction, but will preserve and prepare this site as a back up solution for the NPP Temelín spent fuel storage facility.

On the contrary, if the construction permit for the Dukovany storage facility is not issued on time, the Skalka facility will become the main variant for the NPP Dukovany spent fuel. The Skalka project can play the same role for the NPP Temelín. Consequently, steps to be taken by ČEZ in the future are as follows:

- Preparation of the EIA Study, Safety Report, Site Permit Documentation for both the Dukovany and Skalka site in 1998;
- Request for site permit for both the Dukovany and Skalka in 1999, to be followed by a decision of the ČEZ top management to stop/continue the preparation of the Skalka facility;
- Final bidding process for the Dukovany facility and/or Skalka facility in 1999 - 2000;
- Construction permit for the Dukovany facility and/or Skalka central facility in 2000;
- Construction of the Dukovany facility or Skalka central facility in 2002 - 2004;
- Commissioning of the Dukovany facility or Skalka central facility in 2004 - 2005;
- Preparatory works for the Temelín facility (when the Skalka central facility is not under construction) in 2002 - 2008;
- Commissioning of the Temelín facility in 2011 (when the central Skalka facility is not in operation).

## 5. CONCLUSION

Since 1995, a significant development in the area of spent fuel management in the Czech Republic could be observed. The SÚJB staff has gained new experience with the evaluation of the safety analyses for both spent fuel storage casks and facilities. A dry interim storage facility in Dukovany has successfully been constructed, commissioned and safely operated. The Czech industry started to be involved not only in the production of licensed storage technologies, but also attempts to develop its own storage and transport cask for spent fuel, e.g. at SKODA Nuclear Machinery. The existence of a national spent fuel facility has also led to establishing new research tasks. Some results of the research are presented at this Symposium, e.g. results on the behaviour of spent fuel cladding during long-term (dry) storage, both under normal and accident conditions. Other tasks focuse on long-term behaviour, ageing, maintenance and in service inspections. The development of criteria for a dual use cask intended to be transported after a long storage period should also be addressed in the future.

Significant development has also been achieved in the legislative area. New legislation was adopted in 1997. First of al it was the „Atomic Act" - law No. 18/1997, issued in January 1997. The Atomic Act is based on the internationally adopted principles of nuclear safety and radiation protection, which are implemented in recommendations of the IAEA, ICRP and WHO. The Atomic Act includes provisions declaring, in compliance with the Vienna Convention, the licensee's responsibility for any nuclear damage resulting from an accident. This law also establishes the basic principles of safe spent fuel management. The Atomic Act does not proscribe whether to reprocess or dispose the spent fuel. According to this law, spent fuel is not considered as waste, but both the operator and the State Office for Nuclear safety have the right to declare spent fuel as waste. Following this law, the Ministry of Trade and Industry has had to establish a Radioactive Waste Repository Agency. This Agency, financed by funds from the waste generator, has overtaken the activities connected with the waste management in Czech Republic, including the research and development.

The safe operation of both Dukovany and Temelín will require new spent fuel storage facilities in the near future. The evaluation of the safety analyses for such facilities and the competent preparation for their construction, undoubtedly represent the main challenges both for the operator and regulator.

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IAEA-SM-352/5

# EDF's PROGRAMME FOR SPENT FUEL MANAGEMENT



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**Abstract**

The French policy is presently to reprocess spent fuel and before any fuel assembly can be used, it has to be proven that it is reprocessable after irradiation. Nevertheless, in order to maintain the separated plutonium inventory, reprocessing is only implemented for producing the quantity of plutonium to be recycled. That strategy leads to slowly increase the inventory of spent fuel since the number of authorized reactors for recycling is presently limited. This fact and the French law (of 30 December 1991), which demands to study various fuel cycle back end strategies, led to develop an extended reflexion concerning spent fuel storage and disposal.

## 1. HISTORICAL BACKGROUND

The presently implemented strategy for EdF's Spent Fuel Management is a logical consequence of the past. The energy production needs for France were founded in the mid seventies due to the lack of fossil resources and the possible limited availability of uranium in the long term. Thus, the nuclear programme was launched with the concept of plutonium recycling in breeders. Consequently, industrial tools for reprocessing and recycling were developed. The decision, ten years later, to postpone the breeder development for decades, led to development of plutonium recycling in PWRs.

## 2. THE PRESENT SPENT FUEL MANAGEMENT OF EDF

### 2.1. Data and facts

At the beginning of 1998, the French PWRs had produced about 14,900 t spent fuel, of which 5,300 t has been reprocessed, 6,650 t is stored at the La Hague reprocessing plant and 2,950 t is stored at reactor sites, waiting for transfer to La Hague. Each year, about 1,200 t spent fuel is unloaded from the PWRs and the same quantity is transferred to La Hague, where 850 t is reprocessed each year. All the metal spent fuel from the gas-graphite reactors has been reprocessed. 17 Reactors are loaded with MOX fuel (Fig. 1) and, from 1987, almost 1,000 MOX fuel assemblies (more than 400 t) have been introduced in the 900 MW PWRs (Fig. 2). The separated plutonium stockpile is maintained at a level necessary for the MELOX (MOX fuel fabrication plant) working inventory.

### 2.2. Present programme

EdF maintains the above indicated annual quantities for production of spent fuel (the burnup increase compensates this through energy production), transportation and reprocessing. Each MOX reload contains 16 MOX fuel assemblies associated with 28 UOX fuel assemblies. MOX fuel stays 3 years in the core and UOX stays there for four years. The reactor contains permanently about 30 % of MOX fuel. The MOX fuel use is perfectly satisfactory for reactor operation, behaviour and economics. More than 100 batch/reactor·year have been accumulated (Fig. 3). 20 Reactors are presently authorized to use MOX fuel. Reprocessing of UOX spent fuel allows a reduction of the weight and volume of the radioactive inventory (Fig. 4).

## 3. THE PROSPECTS FOR THE FUTURE

### 3.1. Short term

During 1997, a complete analysis of the French policy concerning the back end of the fuel cycle was realized. The conclusions were presented to the Government and can be summarized as follows:

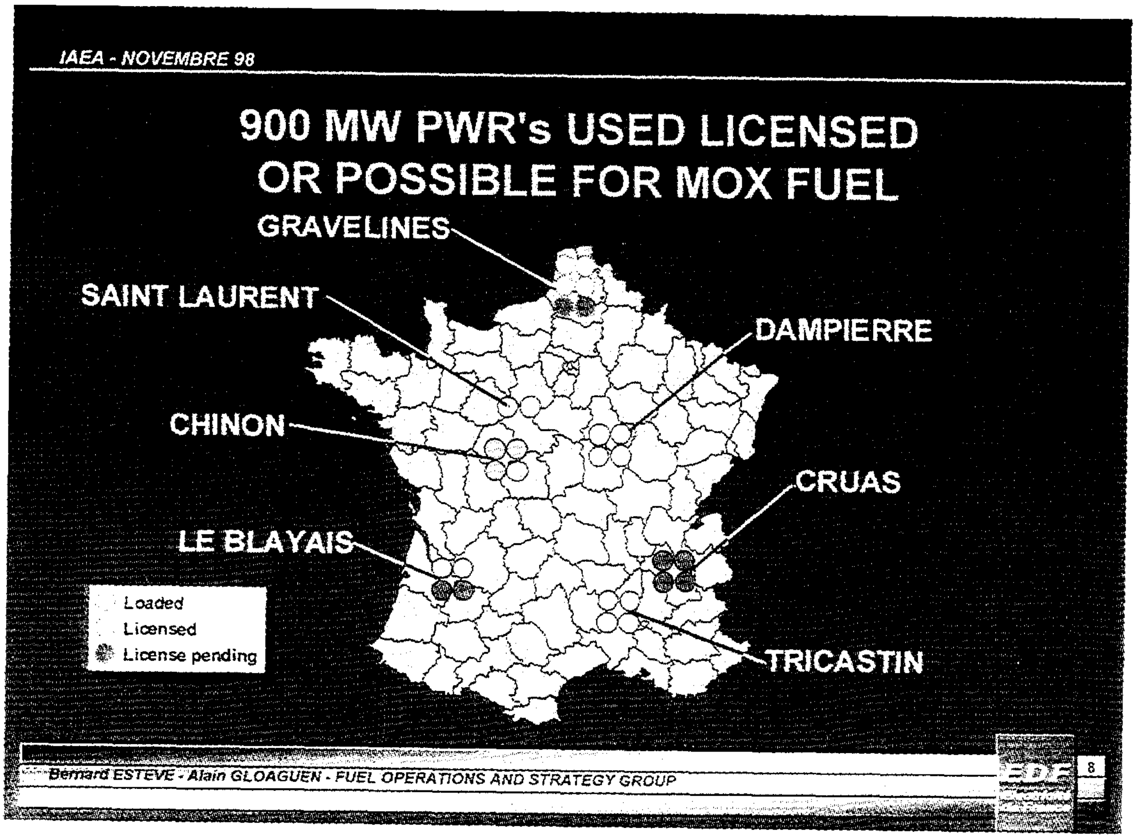


*FIG. 1. Status of MOX fuel utilization in the 900 MW PWRs*

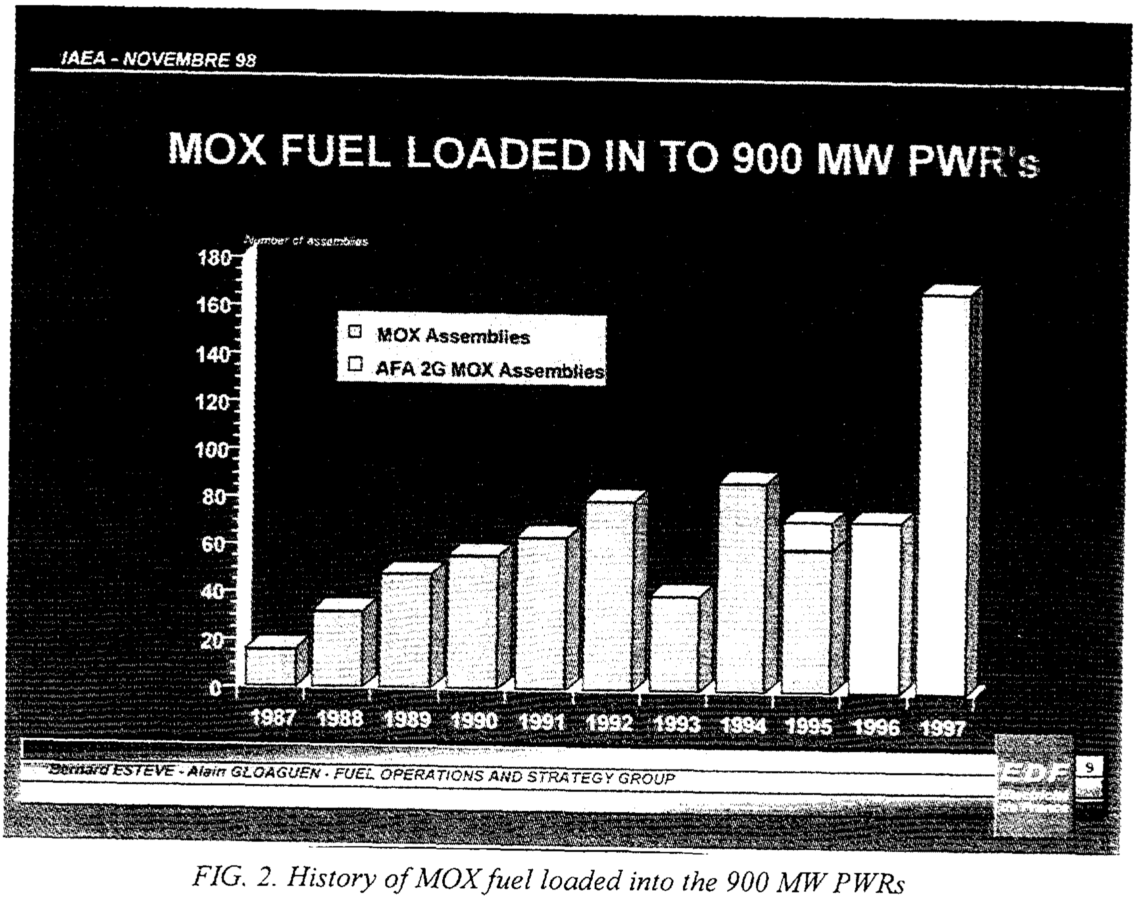


*FIG. 2. History of MOX fuel loaded into the 900 MW PWRs*

IAEA - NOVEMBRE 98

## STATUS OF MOX RELOADS

| | 1987 | 1988 | 1989 | 1990 | 1991 | 1992 | 1993 | 1994 | 1995 | 1996 | 1997 | 1998 | TOTAL cycles | TOTAL MOX |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| BLAYAIS 1 | | | | | | | | | | | C15-16 | C16-16 | 2 | 32 |
| BLAYAIS 2 | | | | | | | | C12-8 | C13-8 | C14- 0 | C15-0 | C16-16 | 5 | 32 |
| CHINON-B 3 | | | | | | | | | | | | | 0 | 0 |
| CHINON-B 4 | | | | | | | | | | | | C11-16 | 1 | 16 |
| DAMPIERRE 1 | | | | C9-16 | C10- 0 | C11-16 | C12- 0 | C13- 0 | C14-16 | C15-8 | C16-16 | C17-16 | 9 | 88 |
| DAMPIERRE 2 | | | | | | | C12-16 | C13-16 | SH | C14-8 | C15- 0 | C16-16 | 5 | 56 |
| DAMPIERRE 3 | | | | | | | | | | | C15- 0 | C16-16 | 1 | 16 |
| DAMPIERRE 4 | | | | | | | | | | | C15- 0 | C16-16 | 1 | 16 |
| GRAVELINES 1 | | | | | | | | | | | C15-16 | C16-16 | 2 | 32 |
| GRAVELINES 2 | | | | | | | | | | | | C16-16 | 1 | 16 |
| GRAVELINES 3 | | | C8-16 | C9-16 | C10-16 | C11-16 | C12-8 | C13-16 | C14-16 | C15-8 | C16-16 | C17-16 | 10 | 144 |
| GRAVELINES 4 | | | C8-16 | C9-8 | SH | C10-16 | C11-16 | C12-16 | C13- 0 | C14- 0 | C15-8 | C16-16 | 8 | 96 |
| ST LAURENT-B 1 | C5-16 | C6-16 | SH | C7-16 | C8-16 | C9-16 | C10-16 | C11-16 | SH | C12-16 | C13-16 | C14-16 | 10 | 160 |
| ST LAURENT-B 2 | | C6-16 | C7-16 | SH | C8-16 | C9-16 | C10- 0 | C11-16 | C12-16 | C13-16 | C14-16 | C15- 0 | 10 | 128 |
| TRICASTIN 1 | | | | | | | | | | | C16-16 | C17-16 | 2 | 32 |
| TRICASTIN 2 | | | | | | | | | | C15-16 | C16-16 | C17-16 | 3 | 48 |
| TRICASTIN 3 | | | | | | | | | | C15-16 | C16-16 | C17-16 | 3 | 48 |
| TRICASTIN 4 | | | | | | | | | | | C15-16 | C16-16 | 2 | 32 |
| TOTAL CYCLES | 1 | 2 | 3 | 4 | 4 | 5 | 6 | 7 | 5 | 9 | 13 | 17 | 76 | 76 |
| TOTAL MOX | 16 | 32 | 48 | 56 | 48 | 80 | 56 | 88 | 56 | 88 | 168 | 256 | 992 | 992 |

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*FIG. 3. Detailed status of MOX reloads*

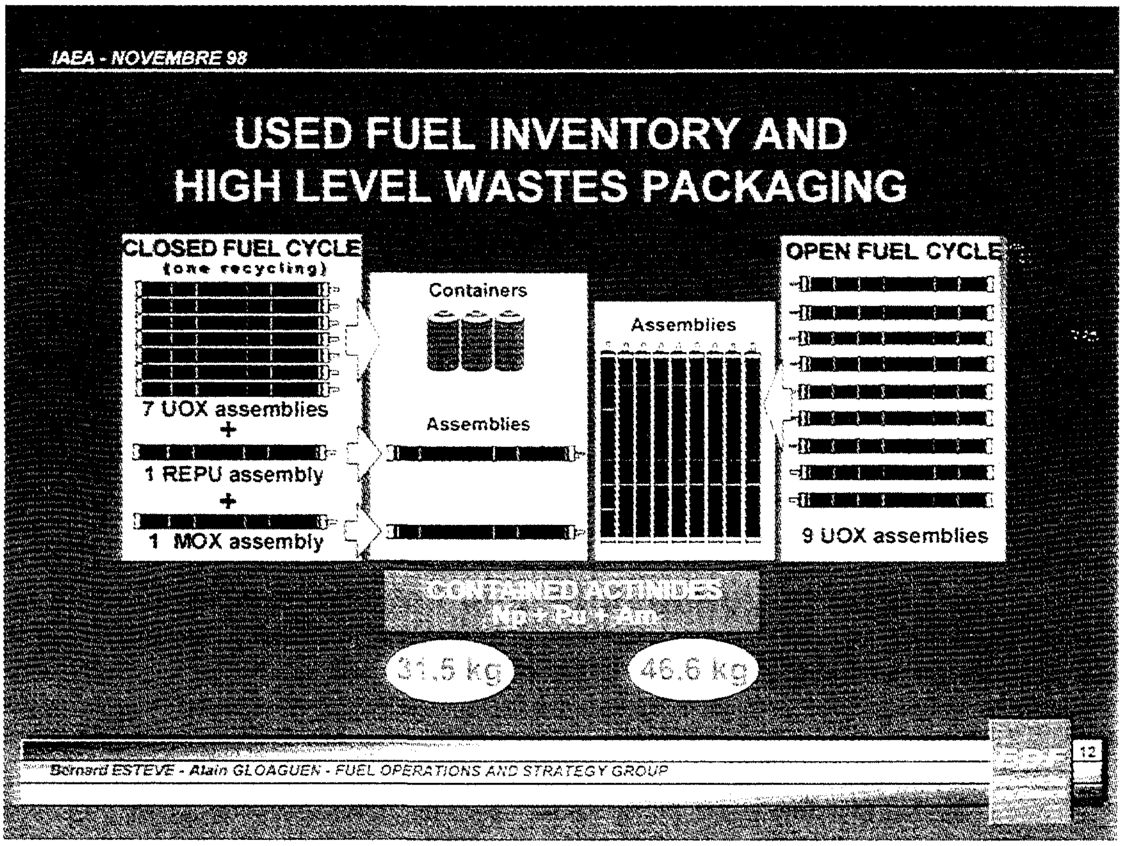


*FIG. 4. Comparison of weight of radioactive inventory of the open and closed fuel cycle*

- the nuclear electricity production will be stable for some years (there is no need for a new reactor before about 2003);
- the reprocessed quantities should be maintained to about 850 t/a, which:
  - permits to master the spent fuel inventory under the existing storage capacities;
  - permits to use the full capacity of MOX fuel fabrication;
  - permits to implement up to 2010 to recycle plutonium, which is already contained in spent fuel assemblies and has well known characteristics.
- the industrial tools exist, are improved and can be used under economical conditions;
- EdF's strategy is open i.e. any future decision concerning the back end could be implemented since present operations do not jeopardize other merging policy,
- no decision is necessary before 2006 (term fixed by the 1991 law to have results) and then there would be time to adapt the strategy to eventually a different one;
- authorizations must be delivered to load MOX in 8 more PWRs (900 MW), which are conceived to receive that type of fuel, in order to give industrial flexibility to recycling (4 cycle core-management).

### 3.2. The future

The future will depend of the results of the studies required by the 1991 law (and awaited by 2006) concerning:

- the separation and transmutation of long-lived radioactive wastes;
- the possibilities of disposal in deep geological repositories (reversible or not) using underground laboratories;
- the possibilities of long term surface storage.

The spent fuel management is deeply concerned by the studies and resulting future decision. Moreover, pursuing after 2006 the present policy will lead to some reflections. Firstly, it is necessary to consider that the spent fuel stored under water in 2010 will be in that situation since 10 years at least. Secondly, the MOX fuel is not planned to be reprocessed at that time since there is enough UOX spent fuel to produce the recycled plutonium and because it is considered that MOX reprocessing is especially interesting in the case of breeder construction. Thirdly, although every fuel assembly is conceived (and this is imposed by the Safety Authority) to be reprocessed, technical or economic reasons can lead to delay in reprocessing. Consequently, EdF is interested in studies concerning the long term behaviour of fuel in wet or dry conditions and in the studies related to direct disposal.

The studies which are launched concern mainly:

- possible concepts of dry storage : cask, vaults;
- behaviour of the fuel rods in wet or dry environment;
- confinement of radioactivity in the fuel pellets with various scenarios of environment;
- analysis of various types of confining materials.

That large scope of investigations has not yet be reduced by results which would permit to define any preliminary option. Various solutions will probably be technically acceptable and the final choice will be made on the basis of economics and political interest.

## 4. CONCLUSION

Time is needed to define a new policy which could be less reversible than the present one and that time is available now, taking into account the available storage capacity and the good behaviour of the spent fuel in the storage pools.

IAEA-SM-352/6a

# LICENSING OF SPENT FUEL STORAGE FACILITIES IN GERMANY



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**Abstract**

An outline of the structure of regulations and the licensing procedure is given, both in terms of requirements and the administrative procedure. The current status of interim storage in Germany is described. Experiences during the licensing procedures and the legal proceedings are mentioned as well as possible license changes.

## 1. INTRODUCTION

Since 1994, the German Atomic Energy Act opens the possibility in the field of spent fuel management to choose between reprocessing and direct disposal. The latter option requires long-term interim storage of spent fuel. In addition, there is a commitment of the Federal Republic of Germany to allow the transport of canisters of vitrified high active waste (HAW) back from foreign reprocessing. Also in this case, there is a need for interim storage until final disposal.

## 2. REGULATION STRUCTURE

Spent fuel storage facilities are divided into At-Reactor Storage Facilities and Away-from-Reactor Storage Facilities.

The At-Reactor Storage Facilities:

- are part of a nuclear power plant;
- need to be licensed according to § 7 Atomic Energy Act;
- need to be licensed by the state authorities as competent authorities e.g. by Bavaria, Hesse or Lower Saxony;
- are supervised also by the state authorities.

The At-Reactor Storage Facilities for the 20 power reactors in Germany are of the wet storage type. The total licensed capacity is about 6,700 t HM.

The Away-from-Reactor Storage Facilities:

- are independent of nuclear power plants and are normally centralized installations;
- need to be licensed according to § 6 Atomic Energy Act;
- need to be licensed by the Federal Office for Radiation Protection (Bundesamt für Strahlenschutz, BfS) as competent authority;
- are supervised by the state authorities of the respective facility location.

For the Away-from-Reactor Storage facilities, Germany has adopted the philosophy of dry interim storage in combined transport and storage casks. Consequently, with the exception of the ZAB Greifswald, all existing or planned Interim Storage Facilities are of the „dry storage“ type.

In the legal frame of § 6 of the Atomic Energy Act, the interim storage of the following amounts has been licensed:

- 8,322 t of heavy metal in spent fuel;
- in 790 casks for dry storage of LWR spent fuel;
- up to 420 casks of these may be used for HAW;
- 463 casks for graphite elements from pebble bed reactors.

These numbers have to be compared to the annual discharge of the German nuclear power plants of about 470 t heavy metal. Contracts for reprocessing in France and the United Kingdom cover about 400 t heavy metal per year with a decreasing tendency.

## 3. THE LICENSING PROCEDURE

The general base for the licensing and operation is the Atomic Energy Act (Atomgesetz). Its original date is 1959; the last change was in April 1998. In paragraph 6, the basic prescriptions for storage of nuclear fuel and high-active waste (HAW) from reprocessing are written down.

### 3.1. Requirements

– *A Need for the Facility*

In the application the need for storing the nuclear material must be demonstrated. There are general arguments, e.g. the interim storage facility may be listed in the Spent Fuel Management Report of the Federal Government, and there are more specific arguments for example the amount of nuclear material from a reactor to be decommissioned. The licensing authority has to take these arguments into account for the decision.

– *Reliability and Knowledge by the Applicant*

The reliability and knowledge of the applicant and the responsible personnel is verified through informations from the respective competent authorities. The appropriate training of the responsible personnel is verified using certificates of education and professional career. The level of requirement is fixed in guidelines concerning the competence in radiation protection and physical protection.

– *Precautions against Injuries*

Precautions according to the „development of science and technology" have to be taken against possible injuries caused by the storage. This general formulation is the base of the technical requirements, which are written in more detail in subordinate regulations and technical standards. The objective is to keep the safety goals.

- Confinement of the Radioactive Inventory
- Radiation Shielding
- Criticality Safety
- Removal of Decay Heat.

The actual dose limits are given in the radiation protection ordinance. The dose limits under normal operation conditions from airborne activity or liquid effluents are 0.3 mSv/year for people outside the storage facility. For direct radiation in the control area outside the facility, a dose limit of 1.5 mSv/year is permitted including radiation from airborne and liquid effluents. Both, direct radiation and radiation doses from effluents have to be minimized according to a principle similar to ALARA. Fulfilment of the requirements is achieved by the low hypothetical leakage rate of the lid system and the shielding effect of the casks and of the building. The surface doses (of the cask) are limited to 0.1 - 0.2 mSv/h for gamma radiation and 0.1 - 0.325 mSv/h for neutron radiation (ICRP60). The effective dose for the population outside the plant is limited to 1 mSv/year from all sources according to ICRP60. This is part of the Directive 96/29/EURATOM from 1996 which has to be transformed into national regulations by the year 2000.

For design base accidents, it has to be shown that the calculated cumulative dose (for 50 years) will not exceed 50 mSv. In the accident analysis, the possible radiological consequences of external and internal events are evaluated. Typical conditions for accident analyses are handling errors like dropping of a cask from a crane, impact of one cask to another, fire, explosion, earthquake, storm, snow or other extreme weather conditions.

For severe external caused accidents, e.g. aeroplane crash, the probability is determined. If the probability is lower than ~ $10^{-6}$ per year, the event is treated as a „rest risk event“. For this case, an estimation of possible radiological consequences is made. Estimating the probability and the consequences, the licensing authority makes a decision, which measures have to be provided for risk reduction.

– *An adequate liability insurance contracted by the applicant*

For storage facilities the upper limit is 500 MDM. The actual value depends on the radioactive inventory and will be determined by the authority.

– *Physical protection against unauthorized access*

The detailed arrangements for this topic are classified as confidential.

### 3.2. Administrative procedure

The procedure to ensure the public participation during the licensing procedure is outlined in the following (typical time intervals in brackets):

| | | |
|---|---|---|
| 1) | Application and submission of:<br>- a safety analysis report;<br>- supporting documents, plans and drawings;<br>- documents proving the accomplishment of the licence requirements;<br>- a short description of the facility. | |
| 2) | Check, if the following is correct and complete:<br>- application;<br>- safety analysis report;<br>- short description. | |
| 3) | Public announcement in the „Bundesanzeiger“. | (1 day) |
| 4) | Public announcement in regional newspapers. | (about 1 week) |
| 5) | Public presentation of:<br>- application;<br>- safety analyses report;<br>- short description.<br><br>at the location of the competent authority and near the facility site. | (during 2 months) |
| 6) | Objections may be raised by every body. | (at least 1 month) |
| 7) | Public announcement of the project hearing in the „Bundesanzeiger“. | (1 day) |
| 8) | Public announcement of the project hearing in regional newspapers. | (about 1 week) |
| 9) | Non-public project hearing with:<br>- authority and experts;<br>- applicant;<br>- persons having raised objections. | (1 - 4 days)<br>(for Interim Storage Facilities) |
| 10) | Proceed the results of the project hearing. | |

11) If all licence requirements are fulfilled and the objections raised have been taken into consideration or rejected: Draft writing.

12) Issue of licence.

13) Public announcement in the „Bundesanzeiger". (1 day)

14) Public announcement of the enacting part of the licence document in regional newspapers. (1 day)

15) Public presentation of the licence document at the location of the competent authority and near the facility site.

The number of persons raising objections and the times for the project hearing were as follows:

- **Ahaus** licence 1997
  2,317 objections, 4 days project hearing

- **Gorleben** licence 1995
  about 14,000 objections, 4 days project hearing

- **Jülich** licence 1993
  925 objections, 1 day project hearing

- **ZLN Greifswald** licensing procedure
  about 15,000 objections, 3 days project hearing

4. CURRENT STATUS OF INTERIM STORAGE

At present, four Away-from-Reactor interim storage facilities are licensed and in operation at the locations Ahaus, Gorleben, Jülich and Greifswald

- **Ahaus (Nordrhein-Westfalen)**

  For the „Transportbehälterlager Ahaus" (TBL-A) 420 cask storage positions corresponding to 3,960 t HM are licensed. The total inventory is $2 \times 10^{20}$ Bq and the maximum heat release is 17 MW. In this licence, an option for 305 casks for storage fuel from the gas cooled graphite moderated prototype reactor THTR is included. The pebble bed fuel elements are graphite spheres containing several grams of high enriched uranium and thorium. So the specific inventory of this fuel is low compared to LWR fuel. The space required for the 305 casks corresponds to 50 cask positions for LWR fuel elements.

  The licensed casks are of the CASTOR type, the permitted maximum burnup for LWR fuel is 65 GW·d/t HM. Now 3 CASTOR V/19 casks, 3 CASTOR V/52 casks and 305 casks with THTR fuel are stored in the Ahaus facility.

- **Gorleben (Niedersachsen)**

  The Gorleben cask storage facility (TBL-G) received its approval for 1500 t HM in 1983, but it could not go into hot operation before 1995. The storage hall is of the same design as that for the Ahaus facility.

In 1995, the approval for an increased capacity of 3,800 t HM corresponding to 420 casks was issued. The licence includes the permission for storage of LWR fuel elements up to 65 GW·d/t HM burnup and for vitrified high-active waste (HAW). The permitted total inventory is 2 x $10^{20}$ Bq, the maximum permitted heat release is 16 MW. Now 5 casks with spent LWR fuel of about 38 t HM in total and 3 casks with vitrified HAW each containing 28 canisters are stored at Gorleben.

- **Jülich (Nordrhein-Westfalen)**

  The storage facility Jülich was built for interim storage of spent fuel from the gas cooled test reactor AVR, which is shut down for decommissioning. The licence for the storage facility comprises 158 casks of the CASTOR type. At the moment 106 casks are stored.

- **Greifswald (Mecklenburg-Vorpommern)**

  The site of Greifswald has two storage facilities one in operation and one under construction. The ZAB (Zwischenlager für abgebrannten Brennstoff) started operation in 1986 in the former GDR. It is a wet storage facility with four pools for VVER type fuel elements from the NPPs Greifswald and Rheinsberg. The total storage capacity is 4,680 fuel elements respectively 562 t HM and a maximum heat release of 0.9 MW. Now the ZAB contains 4,547 fuel elements. The permission for operation was limited until June 2000 and was extended by law for 5 years. When the permission ends, all the fuel must be removed from the storage facility. Therefore, the dry storage facility ZLN (Zwischenlager Nord) was applied and is now under construction. The applied capacity is 585 t HM, respectively 80 cask positions of the type CASTOR 440/84, with a total inventory of 7.5 x $10^{18}$ Bq and a maximum heat release of 0.6 MW. The licence for starting operation is expected in 1999.

## 5. LEGAL PROCEEDINGS AND LICENCE CHANGES

Against a former Ahaus licence from 1997 (1,500 t HM) proceedings were instituted. The court (Oberverwaltungsgericht Münster) rejected the 1996 suits and the highest court (Bundesverwaltungsgericht) rejected the 1998 appeal. Also against the Ahaus licence from 1997 proceedings are instituted.

Against the Gorleben licence of 1995 proceedings were instituted. The court (Oberverwaltungsgericht Lüneburg) rejected the 1996 suits and the highest court (Bundesverwaltungsgericht) rejected the 1997 appeal.

No proceedings were instituted against the Jülich licence from 1993. Consequently, these three mentioned licences are valid in law.

Future licence changes concern for:

- **Ahaus**
  the interim storage of spent fuel elements from the research reactor Rossendorf and changes in the inventory of the CASTOR V/19 cask;

- **Gorleben**
  the CASTOR V/52 cask for BWR fuel elements, non standard fuel elements and administrative changes;

- **Jülich**
  2,400 low enriched uranium pebble bed fuel elements.

IAEA-SM-352/6b

# PRACTICAL EXPERIENCE IN SPENT FUEL MANAGEMENT FOR GERMAN NUCLEAR POWER PLANTS



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**Abstract**

The paper describes the practical experience in spent fuel management gained in the past by using the traditional route of reprocessing and, since the amendment to the Atomic Law in 1994, by using also direct disposal via interim storage.

## 1. INTRODUCTION

After the amendment to the Atomic Law in 1994, reprocessing and direct disposal of spent fuel have been accepted as equal options for nuclear waste management. With the introduction of the so-called Omnibus Bill, the opening of this disposal route besides the traditional disposal by reprocessing was an essential part of this amendment. Since 1980, in accordance with the decision of the Federal and State governments, these disposal concepts have been examined with regard to their technical and safety-relevant comparability, and positive results were confirmed as early as 1985.

In order to ensure the operation of their nuclear power plants (NPPs), the German plant operators have to demonstrate their spent fuel management concepts six years in advance by reprocessing contracts, interim storage contracts or on-site storage capacities.

## 2. REPROCESSING

From the late seventies on, the German utilities have concluded Service Agreements with COGEMA and BNFL, the so-called old contracts. After the abandonment of the Wackersdorf reprocessing project, new reprocessing contracts with COGEMA and BNFL have been signed. Under these contracts the major part of the spent fuel arisings is covered until about 2005.

## 3. RETURN OF RESIDUES

Like the other baseload customers of COGEMA and BNFL, the German utilities have contractual obligations to take back the residual arising from reprocessing of their spent fuel. The major amount of the radioactive inventory (the fission products) is contained in the vitrified waste and the return of those residues has already begun: three casks of vitrified high-level waste (HLW) have been sent back to Germany during the last two years, the total amount to be returned under the Service Agreements will be some 130 casks.

## 4. DIRECT DISPOSAL

In view of the new situation due to the amendment to the Atomic Law, RWE Energie as the operator of the Biblis, Gundremmingen and Mülheim-Kärlich NPPs, terminated at the end of 1994 parts of their reprocessing contracts and, in agreement with the competent licensing authority, partly based the waste management provisions on direct disposal. This has led to considerable cost reductions with positive effects on the balance sheet and the fuel cycle costs.

According to the planning of the utilities, spent fuel arising from the German NPPs is to be disposed of by reprocessing as well as direct disposal. An increasing shift from reprocessing – having been practised for many years – to direct disposal is aimed at. The reasons for this shift are the well-known technical and economic advantages of direct disposal in Germany. A current assessment of the utilities shows that from the year 2003 about 50 % of the spent fuel arising from German NPPs will be disposed of by reprocessing and about 50 % by interim storage/direct disposal (Figure 1). There is agreement between the German utilities to avail themselves of both disposal routes, enabling them to choose one of these disposal alternatives as the circumstances may require, and to further optimize the costs. In this context, it should be pointed out that the interim storage of spent fuel leaves the final disposal route open, and a decision on the further treatment of the spent fuel can be taken during the interim storage period of about 40 years.

At present, two interim storage facilities – TBL-G at Gorleben and TBL-A at Ahaus – are being operated with an overall capacity of about 4,000 t HM each. These facilities will cover the interim storage requirements for spent fuel and vitrified high-level waste arising from reprocessing until the year 2010. If some utilities continue to have their spent fuel reprocessed, these capacities will even cover a longer period, and new interim storage facilities will not be needed in the near future. However, the interim storage of spent fuel is absolutely necessary due to the spent fuel management provisions required under the Atomic Law and the unavailability of sufficient storage pond capacity at the NPPs.

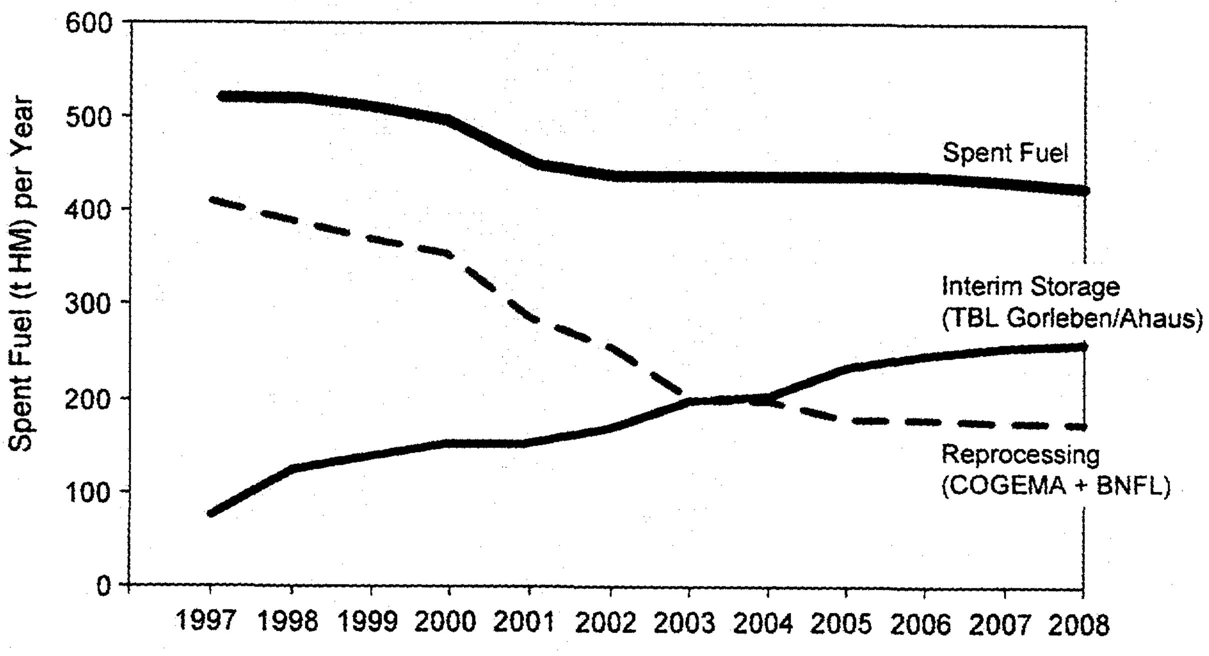


*FIG. 1. Spent fuel arisings*

Direct disposal of spent fuel in Germany requires long-term interim dry storage in transport and storage casks prior to its final disposal. Although there has been strong opposition by anti-nuclear groups, five CASTOR casks containing spent fuel from the Philippsburg, Gundremmingen and Neckar NPPs and three casks containing HLW canisters from La Hague have been stored in the Gorleben storage facility (TBL-G) (Figure 2) and six CASTOR casks containing spent fuel from the Gundremmingen and Neckar NPPs in the Ahaus storage facility (TBL-A) to date. According to the utilities' waste management schedule, the Gorleben storage facility which, in contrast to the Ahaus storage facility, has also been licensed for high-level waste shall, in general, be used for the storage of vitrified residues.

The casks currently used are the CASTOR V/19 for PWR fuel, CASTOR V/52 for BWR fuel and CASTOR HAW 20/28 for HLW canisters. The CASTOR V casks have a capacity of about 10 t HM and the CASTOR HAW cask can accommodate up to 28 HLW canisters being equivalent to about 40 t HM.

In the course of the previous spent fuel transports to Gorleben and Ahaus practical experience and know-how has been gained regarding the loading and handling of CASTOR transport and storage casks (Figure 3).

## 5. SPENT FUEL TRANSPORT

The main strategy of the anti-nuclear groups is to make the spent fuel management impossible or, at least, difficult and expensive. As the interim storage represents a reliable and economical solution for spent fuel management, the CASTOR transports are facing a violent opposition by anti-nuclear activists. During the last three years, tens of thousands of police had to be engaged to escort each of the CASTOR transports and to guarantee a safe arrival at the interim storage facility.

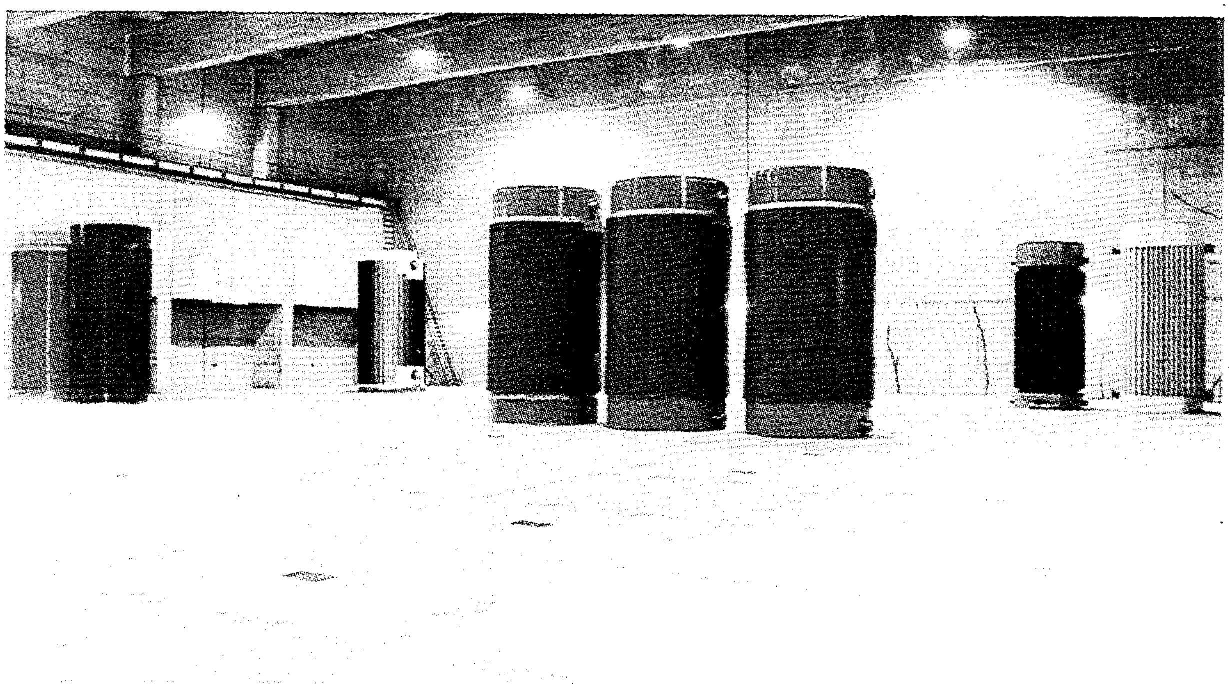

*FIG. 2. Gorleben interim storage facility*

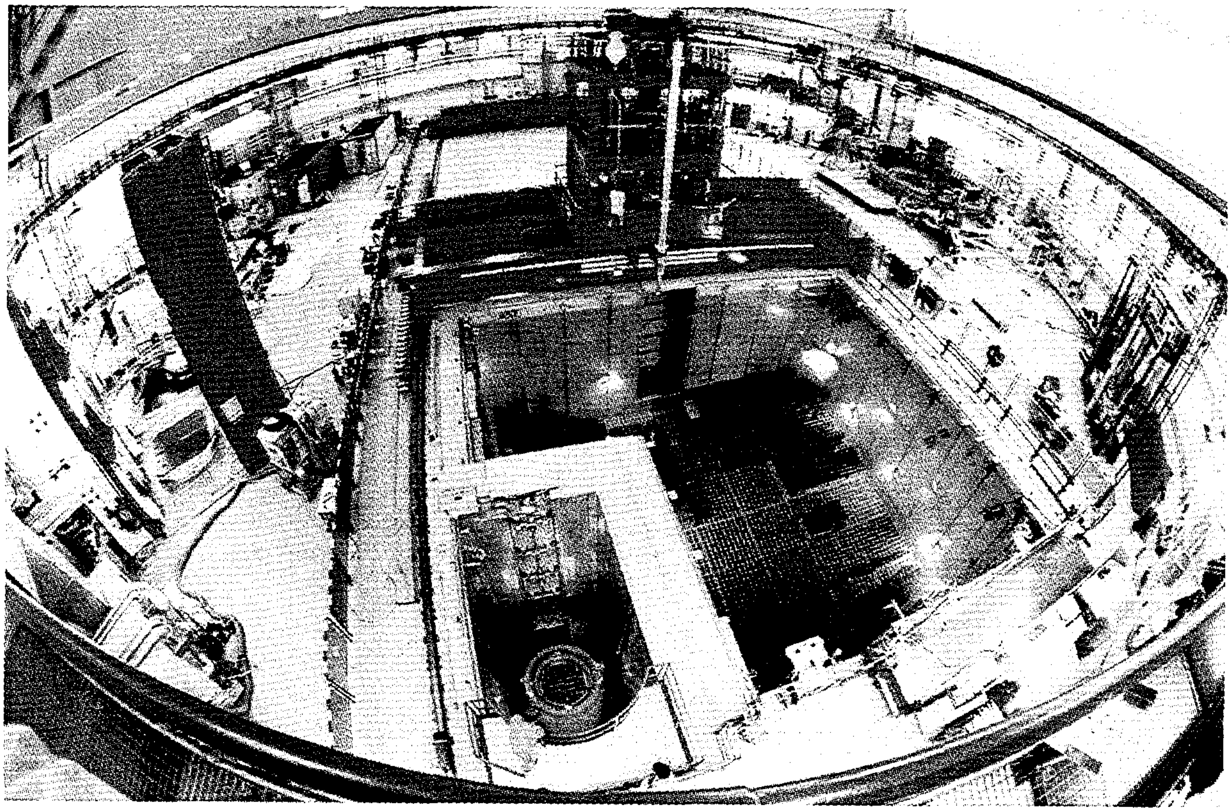

*FIG. 3. Loading of CASTOR V/52 cask at Gundremmingen NPP*

While over the last two decades a steadily increasing number of casks has been transported to the reprocessing plants every year, an event which became known at the end of April 1998. Through an official message of the French Environmental Minister, it became known that a number of transport casks were found to have local surface contamination above the regulatory value of 4 $Bq/cm^2$ on arrival at the Valognes terminal near La Hague, although a review of the transport documentation showed, that the casks had been clean when leaving the NPPs in Germany. This caused a suspension of the spent fuel transports in France, Germany and Switzerland.

In the run-up to the federal elections, this contamination phenomenon became a major political issue. All spent fuel transports from German NPPs have been halted, and it is expected that they will be resumed as soon as the requirements of the Federal Ministry for the Environment, the so-called 10-Points-Plan, have been fulfilled.

In this context, the utilities have proposed to take the following measures (Figure 4):

(1) Optimization of the technical measures to prevent contamination and extension of the radiation protection measurements;

(2) Establishment of an information and reporting system;

(3) Improvement of the organisational structure within the utilities;

(4) Restructuring of the transport organisation.

***(1) Optimization of the Technical Measures to prevent Contamination and Extension of the Radiation Protection Measurements;***

***(2) Establishment of an Information and Reporting System;***

***(3) Improvement of the Organisational Structure within the Utilities;***

***(4) Restructuring of the Transport Organisation***

*FIG. 4. Measures to be taken by the utilities before resuming transports*

## 6. FUTURE PROSPECTS

The waste management has to ensure that spent fuel assemblies and spent core components are disposed of in due time under economic aspects and in compliance with the licensing conditions. With regard to an efficient plant operation, practical measures have to be planned for the disposal of the uranium, MOX and reprocessed uranium fuel assemblies, the control and absorber elements as well as the fuel assembly channels. The synergy effects between fuel management and waste management are of great advantage.

Under the current interim storage licence spent uranium fuel assemblies and MOX fuel assemblies can be disposed of in the medium term by using certain optimization possibilities for cask loading. The efforts of the NPP operators to achieve even higher fuel assembly burnup will require in the long term an optimum utilization of the physical-technical potential of existing and future cask types. Therefore, a licence is required which basically is geared to the achievement of the primary safety objectives by applying advanced methods of calculation. The approved generic licence for core reloads could serve as a model for such a licence (Figure 5).

The pilot conditioning plant at the Gorleben site, which is currently in its cold start-up phase, is of great importance with regard to interim storage and direct disposal. Its functions such as optimization of interim storage, preparation for direct disposal, cask handling and waste treatment are currently determined in extensive research and development programmes. The hot start-up of this plant is scheduled for mid-1999. Special attention is given to the development of a new waste disposal concept with considerable cost saving potential.

- Primary criteria ensure that all safety-relevant objectives are met
- Secondary criteria reduce the calculation effort

*In-core fuel management*

- Primary criteria
  - fuel rod integrity
  - neg. moderator temp. coeff.
  - minimum DNB
  - . . .
- Secondary criteria
  - control rod worth
  - burnup
  - linear heat flux
  - . . .

*Cask loading*

- Primary criteria
  - subcriticality
  - decay heat removal
  - dose rate
  - fuel rod integrity
  - . . .
- Secondary criteria
  - n- and γ source strength
  - burnup
  - minimum cooling time
  - cladding temperature
  - . . .

*FIG. 5. Generic specifications for in-core fuel management serving as a model for an advanced cask licence*

## 7. CONCLUSIONS

For both disposal routes the technical feasibility and the reliability of the steps performed up to now have been demonstrated.

The cost savings resulting from the shift from reprocessing to interim storage and direct disposal have been assessed on the basis of the current costs and prices for reprocessing services.

In case of a positive decision by the competent authorities on the repository for radioactive material and on the commissioning of the conditioning plant at Gorleben the further disposal steps can be carried out under optimized technical and commercial aspects.

IAEA-SM-352/7

# SPENT FUEL DRY STORAGE IN HUNGARY



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**Abstract**

Paks Nuclear Power Plant is the only NPP in Hungary. It has four VVER-440 type reactor units. Since 1989, approximately 40-50% of the total annual electricity generation of the country has been supplied by this plant. The fresh fuel is imported from Russia. Most of the spent fuel assemblies have been shipped back to Russia. Difficulties with spent fuel transportation to Russia have begun in 1992. Since that time, some of the shipments were delayed, some of them were completely cancelled, thus creating a backlog of spent fuel filling all storage positions of the plant. To provide assurance of the continued operation, Paks NPP's management decided to implement an independent spent fuel storage facility and chose GEC-Althom's MVDS design. The construction of the facility started in February 1995 and the first spent fuel assembly was placed in the store in September 1997. The paper gives an overview of the situation, describing the conditions leading to the construction of the dry storage facility at Paks and its implementation. Finally, some information is given about the new Public Agency for Radioactive Waste Management established this year and responsible for managing the issues related to spent fuel management.

## 1. INTRODUCTION

Hungary has a population of 10.6 million and a land area of 93,000 km$^2$, with a population density of 114 inhabitants/km$^2$. Nuclear power generation in Hungary has an important role in the electricity supply. The country's only nuclear power station is located on the west bank of the Danube at Paks. The plant consists of four pressurised water reactors of the Soviet VVER-440/213 type, commissioned between 1981 and 1987. The rated electric performance of each unit is 460 MW. Since 1989, approximately 40-50 % of the total domestic electricity generation has been supplied by Paks NPP.

Concerning the fresh nuclear fuel supply and management the spent fuel assemblies arising from this activity, Hungary has so far relied on foreign services, first from the Soviet Union and later from Russia. Changes of the political and economical system in the Soviet Union and in the relations with countries operating VVERs, made inevitable some modification and diversification in fuel strategy.

## 2. FUEL CYCLE STRATEGY

### 2.1. Fresh fuel supply

The fresh fuel is being imported form Russia (previously from the Soviet Union). There was only one exception. Two years ago, 235 slightly used German fuel assemblies of Soviet origin were shipped from Greifswald NPP and then loaded into the reactors.

To diversify the fresh fuel procurement, Paks NPP together with the Finnish utility IVO signed a contract for developing alternative VVER-440 fuel, in 1995. The contract was awarded to BNFL (UK). The Lead Test Assemblies are being tested at Loviisa. If the results of the one-year test are satisfactory, the Hungarian Operator can start the domestic licensing procedure and will be in a position to choose from different suppliers.

## 2.2. Amounts of spent fuel unloaded from the reactor

There are 349 fuel assemblies in the reactor core. They have a hexagonal cross section, each enclosing 126 fuel rods. Yearly, 120 spent fuel assemblies are unloaded in average, which is equal to 14.4 t HM. According to this spent fuel production rate, more than 15,000 spent fuel assemblies will be discharged from the four reactors during their expected 30 years lifetime.

Since the early 1990s, Paks NNP has been working on introducing the "partly three four years" and "total four years" fuel cycles. Evaluation of the results and further extension of the programme is expected in the near future. This will result in higher burnup and a corresponding lower amount of discharged spent fuel. Taking into consideration the possible life extension of the reactors, however, the total amount of spent fuel generated could increase significantly.

## 2.3. Spent fuel management

### *2.3.1. Original ideas and their first modification*

According to the original fuel strategy, the Soviet Union undertook not only the supply of new fuel but also accepted the spent fuel for reprocessing. This arrangement included some unique elements. As it was foreseen by the original design, spent fuel was to be transported back after three years cooling in the at-reactor (AR) pool. Products of the reprocessing process (all radioactive waste, plutonium, uranium) were supposed to stay in the Soviet Union. Consequently, spent fuel racks in the AR pools were designed and constructed to hold spent fuel unloaded for 3 years (i.e. 349 assemblies).

Later, the original concept was changed, when spent fuel only after a five year decay cooling was accepted for shipment. To double the AR storage capacity, spent fuel pool racks were constructed and presently, the borated stainless steel containing compact storage racks provide for roughly 6 years AR storage.

### *2.3.2. Spent fuel shipments to Russia*

The first spent fuel shipment to the Soviet Union took place in 1989. The number of assemblies returned to the Soviet Union and later to Russia is listed in Table 1. All shipments were carried out using the standard Railway Transport Unit, including the TK-6 containers.

TABLE 1. SPENT FUEL ASSEMBLIES RETURNED TO RUSSIA

| Year | No. of Assemblies |
|---|---|
| 1989 | 116 |
| 1990 | 235 |
| 1991 | 210 |
| 1992 | 240 |
| 1993 | 180 |
| 1995 | 480 |
| 1996 | 240 |
| 1997 | 450 |
| 1998 | 180 |
| Total | 2,331 |

Until 1992, the shipments took place regularly, once or twice a year, providing the necessary at-reactor cooling pool capacity, but since that time the process had an occasional character. All transports were on an exceptional basis only and it is always possible that they will be discontinued, if Hungary does not take back the reprocessing wastes.

*2.3.3. Selection of the storage technology*

Taking into account the above described uncertainties and economic conditions, Paks NPP started to review the spent fuel strategy in the early 1990s, looking for new alternatives. Having reviewed other countries' solutions and in line with the "wait and see" policy, a decision was made to construct an AFR storage facility to provide interim storage at the Paks site. This solution keeps the way towards both options, once-through or reprocessing, open.

Information was collected about existing interim storage facilities and the requirements for the format and content of a Feasibility Study were drawn up. After a pre-selection process, 7 companies were invited to submit Feasibility Studies, mostly for dry storage technologies.

As a result of the wide range evaluation by Paks NPP with the contribution of the invited Hungarian organisations (designer, regulatory body), SKB of Sweden and the IAEA, the number of potential vendors was reduced from seven to three.

Having considered all aspects, the final decision by the management of the Paks Nuclear Power Plant was the selection of the GEC-ALSTHOM's (UK) Modular Vault Dry Storage (MVDS) system and a design contract was signed on 28 September 1992.

*2.3.4. Long-term considerations*

In case return of the spent fuel would no longer be possible to Russia, the AFR storage facility according to its modular nature, can be extended to accommodate all spent fuel assemblies discharged from the four reactors at Paks. This will provide the necessary time to select the site and technology for the disposal of high level waste and spent fuel. The final disposal facility for high level radioactive waste, or for conditioned spent fuel in case of direct disposal, is expected to be commissioned by 2040 and that defines the anticipated length of operation for the MVDS, i.e. 50 years.

Since the construction of a high level waste repository is required independently from the strategy for the back-end of the fuel cycle, as disposal of the operational and decommissioning waste originating from the nuclear power plant is necessary, a National Project was launched. In the first phase of the Project, a complex radioactive waste management strategy was developed.

To provide for the organisational framework of these activities, in the new Law on the Use of Atomic Energy, and related decrees published in 1997, a non-profit organisation was created to take care of all the relevant issues. According to this new Law, a so-called Nuclear Waste Fund was established in 1997.

## 3. IMPLEMENTATION OF THE DRY STORAGE FACILITY

### 3.1. Description of technology

The MVDS (Fig. 1) provides at least 50 years of interim storage for VVER-440 fuel assemblies in a contained and shielded system. The fuel assemblies are stored vertically in individual Fuel Storage Tubes, the matrix of Storage Tubes being housed within a concrete vault module that provides shielding. To prevent the development of eventual corrosion processes, the fuel assemblies are in an inert nitrogen environment inside the Storage Tubes. Decay heat is removed by a once-through, buoyancy driven, ambient air flow across the exterior of the Fuel Storage Tubes, through the vault and the outlet stack. There is no direct contact between the fuel assemblies and the air flow.

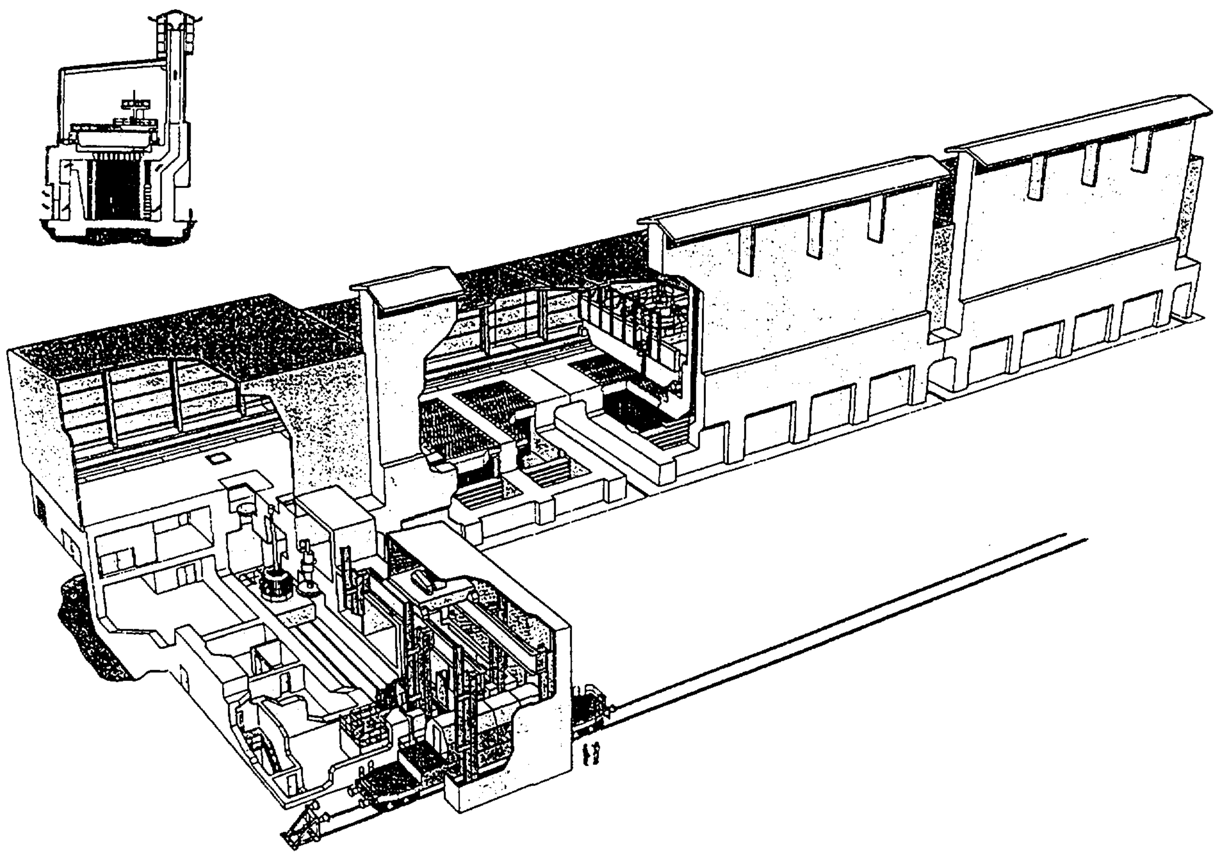

*FIG. 1. Paks modular vault dry storage facility*

The storage facility functionally can be divided into three major structural units. The first one is the vault module where the spent fuel assemblies are stored in vertical tubes. These vault modules include at least three or maximum five vaults, depending on the geometrical arrangement. Each vault is capable of accommodate 450 spent fuel assemblies (see Fig. 2).

The second major structural unit is known as the Charge Hall (Fig. 3) where the Fuel Handling Machine (Fig. 4) is located for the fuel handling operations. The hall is bordered by the reinforced concrete wall of the ventilation stack on one side and by a steel structure with steel plate sheeting on the other side. The basic function of the sheeting is to protect the Fuel Handling Machine against climatic stresses.

The third major unit is the so-called Transfer Cask Reception Building (Fig. 5) in which the reception, preparation, unloading and loading of the transfer cask takes place. The fuel handling system and other auxiliary systems are installed in this building.

The fuel assemblies are transported to the MVDS from the AR pool using the C-30 transfer cask and its railway wagon. The transfer cask is received in the transfer cask reception building where it is removed from the railway wagon and prepared for fuel assembly unloading. The fuel is raised into a drying tube directly above the cask where it is dried prior to being lifted into the Fuel Handling Machine. The fuel assemblies are transferred, within the Fuel Handling Machine, to the vertical Fuel Storage Tubes located in the vaults.

Once the Fuel Handling Machine has moved away from the Storage Tube the air is evacuated from the tube and replaced with nitrogen. After that operation the tube is connected to the built-in nitrogen system which monitors the storage environment of the spent fuel assemblies.

### 3.2. Licensing

The licensing process of the Independent Spent Fuel Storage Facility was very complicated and complex. A number of new, earlier not requested permits were required. Some authorities began to work out and to publish their general requirements during the implementation process. The main authorities issuing the permits involved several other authorities in the course of their licensing process. The situation was complicated with the fact that some regulatory processes were in a way connected with each other. The licensee had to address more than 20 different authorities during the licensing of the facility.

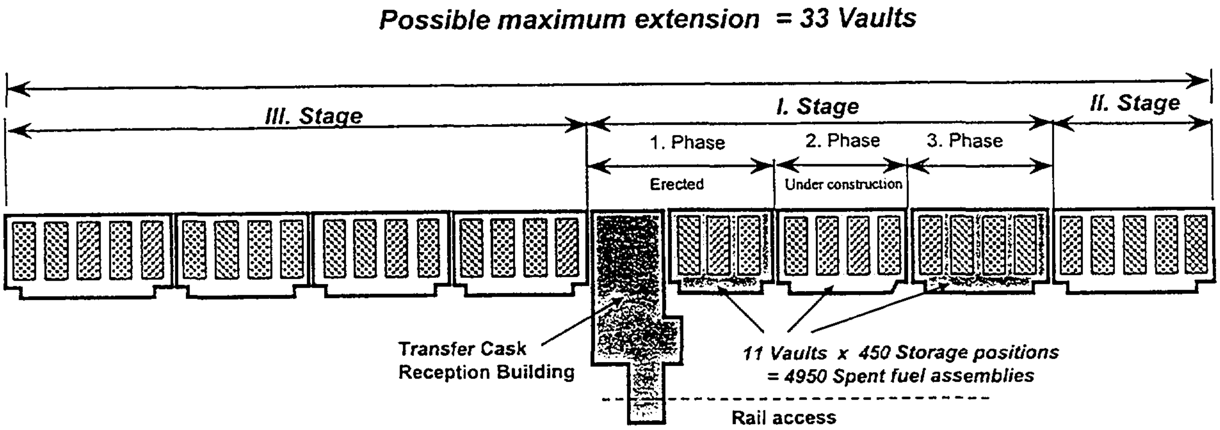


*FIG. 2. Construction phases of Paks MVDS*

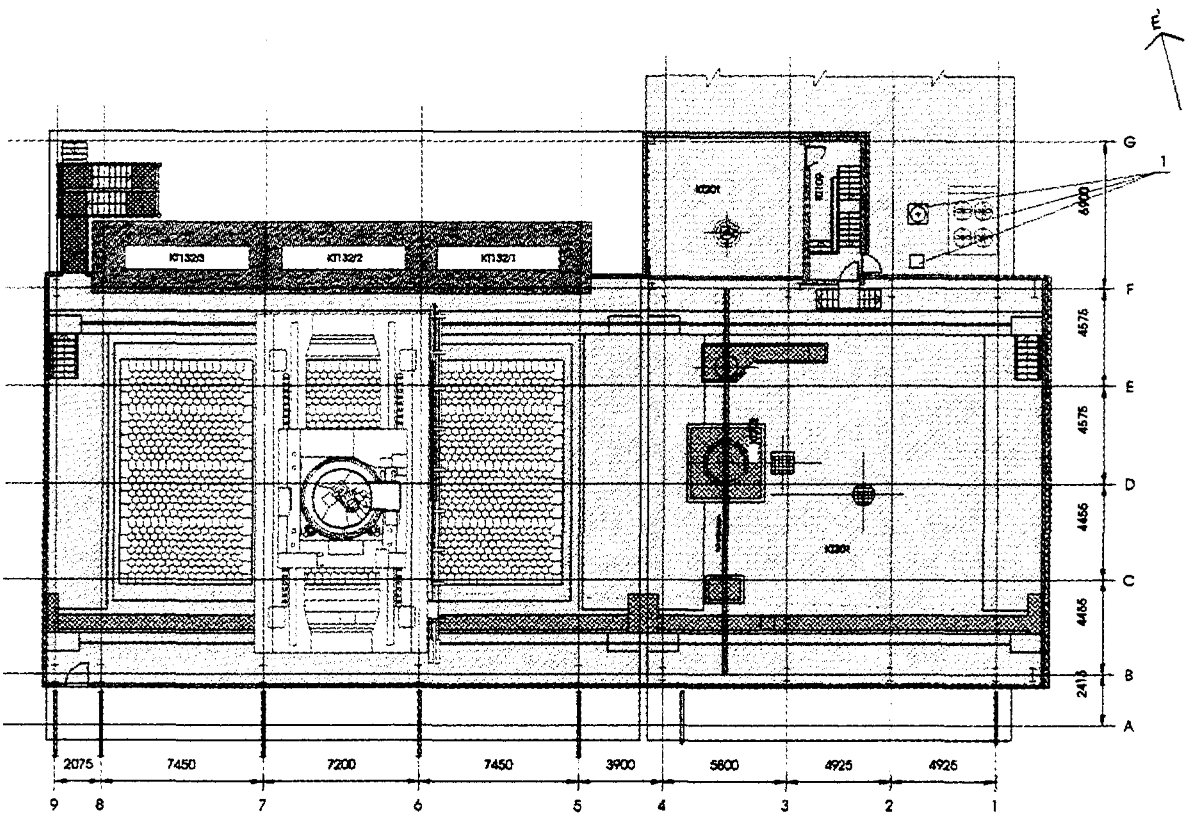


*FIG. 3. Paks MVDS - Charge hall*

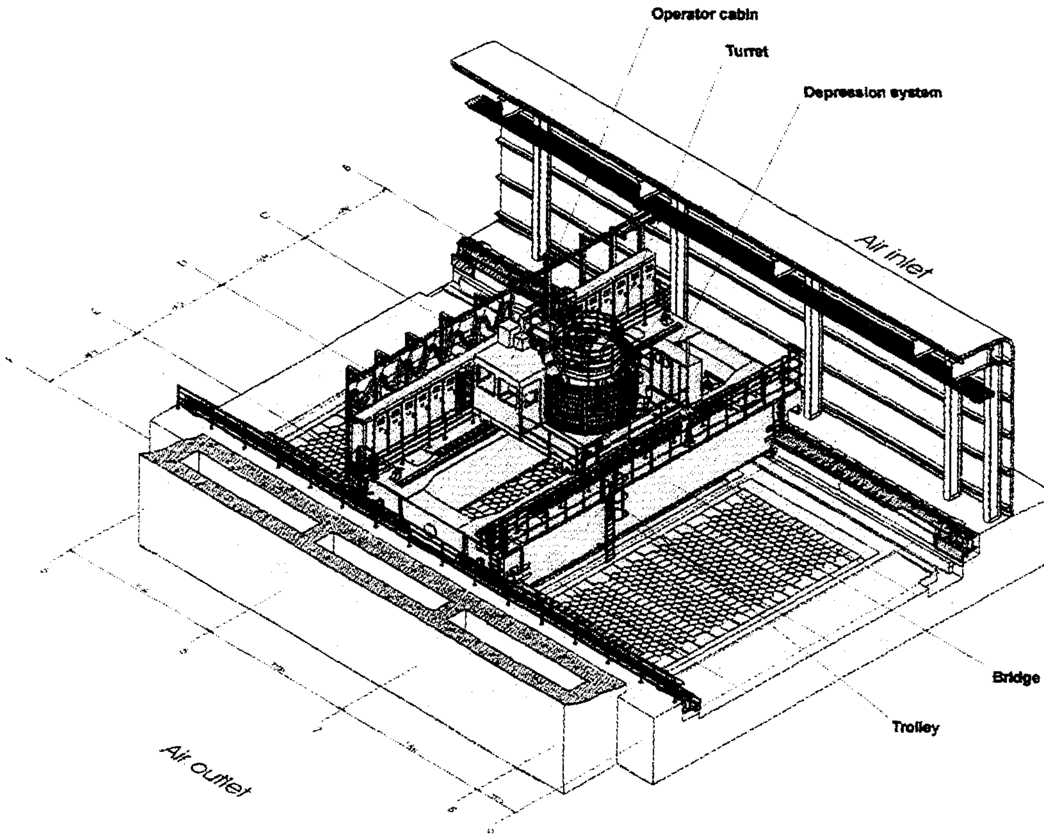


*FIG. 4. Paks MVDS - Fuel handling machine*

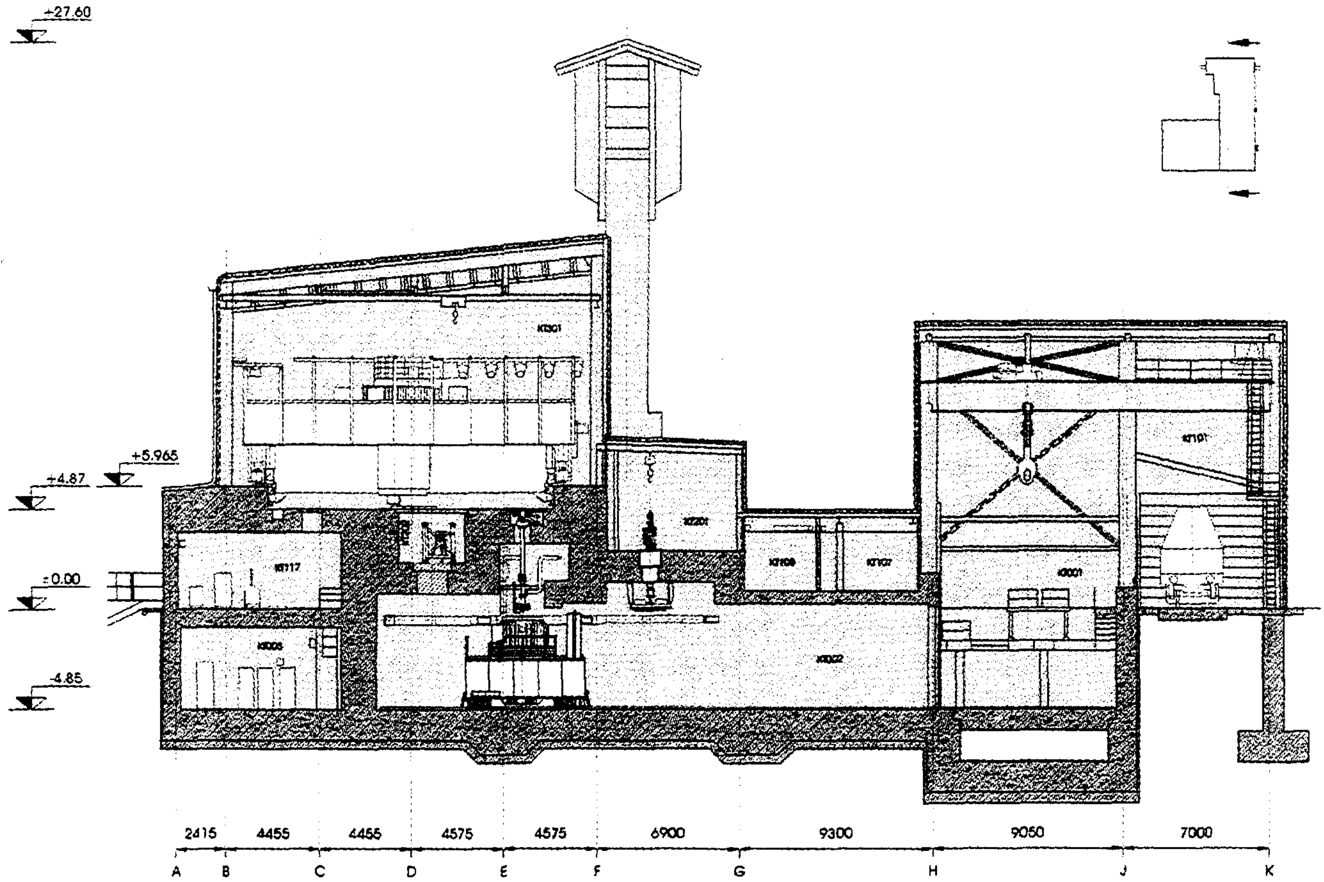


*FIG. 5. Paks MVDS - Transfer cask reception building*

In the beginning of the design, there were no Hungarian regulations regarding spent fuel storage facilities. Therefore, it was agreed by the competent authority (Hungarian Atomic Energy Authority, Nuclear Safety Directorate) and the operator of the NPP, to consider the relevant US regulations when compiling the PCSR. This essentially meant meeting the requirements of 10CFR72 and drawing up a Safety Assessment Report in compliance with the US NRC Regulatory Guide 3.48.

The licensing process preceding the construction lasted for 14 months. The main licensing milestones are shown in Table 2.

TABLE 2. MAIN MILESTONES OF THE LICENSING PROCEDURE

| Type of the license, or permit | Issued on |
|---|---|
| Environmental Permit | 20 July 1994 |
| Building Permit | 20 September 1994 |
| Water Permit | 28 September 1994 |
| Site Permit | 09 November 1994 |
| Nuclear Safety Permit | 23 December 1994 |

In possession of these permits the Hungarian Atomic Energy Commission issued the Construction Licence in February 1995, which allowed to start the construction work.

### 3.3. Construction and commissioning

Preceding the construction work, a number of activities relating to the preparation of the area were carried out. Parallel with these activities, contracts were concluded for the manufacturing of mechanical equipment requiring long delivery time (e.g. Fuel Handling Machine).

The actual construction work started in March 1995, and lasted for 14 months. In this phase of construction, the transfer cask reception building and a vault module including three vaults was erected.

The last task of the construction was the so-called comprehensive testing of the facility with fifteen dummy fuel assemblies, in December 1996.

The Commissioning License was issued by the Hungarian Atomic Energy Commission in February 1997. This stage was finished in December 1997 by filling the first vault with 450 spent fuel assemblies, which took almost 3 months, from 16 September 1997 to 8 December 1997.

### 3.4. Operational experience

Loading of the 450 fuel assemblies in the vault resulted in a collective dose of 2.4 man*mSv, which is considered to be a sufficiently low value for such operations. The highest individual dose was 0.028 mSv, while the average value was 1 μSv.

Effluents and airborne releases during the loading phase were in accordance with the design. According to the indications of the monitoring system, the environmental impact of the operation was insignificant.

During the passive storage period the operation went smoothly, there was no event to be mentioned.

Based on the evaluation of the commissioning phase and on the updated Safety Report, the Hungarian Atomic Energy Authority issued the operational licence on 26 August, 1998. Loading of the second vault to provide the necessary storage capacity in the AR pools began immediately.

### 3.5. Future plans

Due to the uncertainties of further spent fuel shipment to Russia, the management of the Paks NPP made a decision to continue the construction with Phase 2, to provide more storage space by 1999. Construction work began in February 1998 and was foreseen to be finished in December 1999. Decision about the construction of the Phase 3 will be made in near future, depending on the storage capacity demand. Thus, the total comes to 11 vaults accommodating 4,950 spent fuel assemblies.

Provision is made in the design to extend the storage facility to a total of 33 vaults, to 14,850 storage positions. This number would be sufficient to store all spent fuel generated during the rest of operation of the Paks NPP. The modular "in line" configuration used for the storage vaults allows the use of a single Fuel Handling Machine and a central transfer cask reception building throughout the planned extension phases.

## 4. THE PUBLIC AGENCY FOR RADIOACTIVE WASTE MANAGEMENT (PURAM)

### 4.1. Mission of PURAM

The new Atomic Energy Law prescribes: "As the solution of such matters is in the national interest, the performance of tasks related to the final disposal of radioactive waste, as well as to the interim storage and final disposal of spent fuel, and to the decommissioning of a nuclear facility shall be the responsibility of an organisation designated by the Government". According to that, the Government nominated the Public Agency for Radioactive Waste Management (PURAM) for carrying out the work defined above. The main fields of activities of PURAM can be defined as follows:

- *Low and intermediate level waste (L/ILW) management*
  - Operate the existing repository;
  - Prepare and construct a new L/ILW repository.
- *High level waste (HLW) and spent fuel management*
  - Elaboration and introduction of a new strategy for the back end of the fuel cycle;
  - Investigations to prepare for HLW disposal;
  - Operation and extension of the existing interim spent fuel storage facility.
- *Decommissioning of nuclear facilities*
  - Decommissioning of NPP Paks;
  - Decommissioning of the training and research reactors;
  - Decommissioning of the interim spent fuel storage facility.

### 4.2 Structure of PURAM

PURAM is a fully state owned non-profit agency established by the Director General of the Hungarian Atomic Energy Authority, acting on behalf of the Government. The managing director, to whom the following divisions report, heads the organisational structure of the Agency:

- Research and Development;
- Implementation and Ventures;
- Finance and Administration.

The operation of the existing L/ILW repository belongs to the "Implementation and Ventures" divisions. PURAM has two headquarters, one in Budaörs (suburb of Budapest) and the second in Paks, the existing L/ILW repository is located in Püspökszilágy. The staff of the Agency is about 70 persons.

IAEA-SM-352/8

# STORAGE OF SPENT FUEL FROM POWER REACTORS IN INDIA MANAGEMENT AND EXPERIENCE



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**Abstract**

The spent fuel management programme in India is based on closing the nuclear fuel cycle with reprocessing option. *This will enable the country to enhance energy security through maximizing utilization of available limited uranium* resources while pursuing its Three Stage Nuclear Power Programme. Storage of spent fuel in water pools remains as prevailing mode in the near term. In view of inventory build up of spent fuel, an Away-From-Reactor (AFR) On-Site (OS) spent fuel storage facility has been made operational at Tarapur. Dry storage casks also have been developed as 'add on' system for additional storage of spent fuels. The paper describes the status and experience pertaining to spent fuel storage practices in India.

## 1. INTRODUCTION

Nuclear technologists continue to ponder over the question 'What to do with the highly active spent fuel which still possesses an economic value'? In other words it calls for finding ways to close the nuclear fuel cycle. The long-term strategy for the nuclear power programme depends on the answer to this question.

A decision on this issue is not only a matter of paramount economic importance but will soon become an essential prerequisite for the operation of nuclear power plants, given the ever increasing restrictions imposed by regulatory authorities on environmental concerns and public acceptance.

### 1.1. Indian scene: Management of spent fuel

Considering the vast resources of thorium in the country and the limited resources of uranium, deciding the spent fuel management programme in India based on the reprocessing option is considered a prudent and a responsible choice.

Having chosen the option of reprocessing, development activities related to high level waste management, fabrication of MOX fuel and fast breeder technologies have gained importance. The scheme of management of spent fuel (SF) in India is given in Fig. 1.

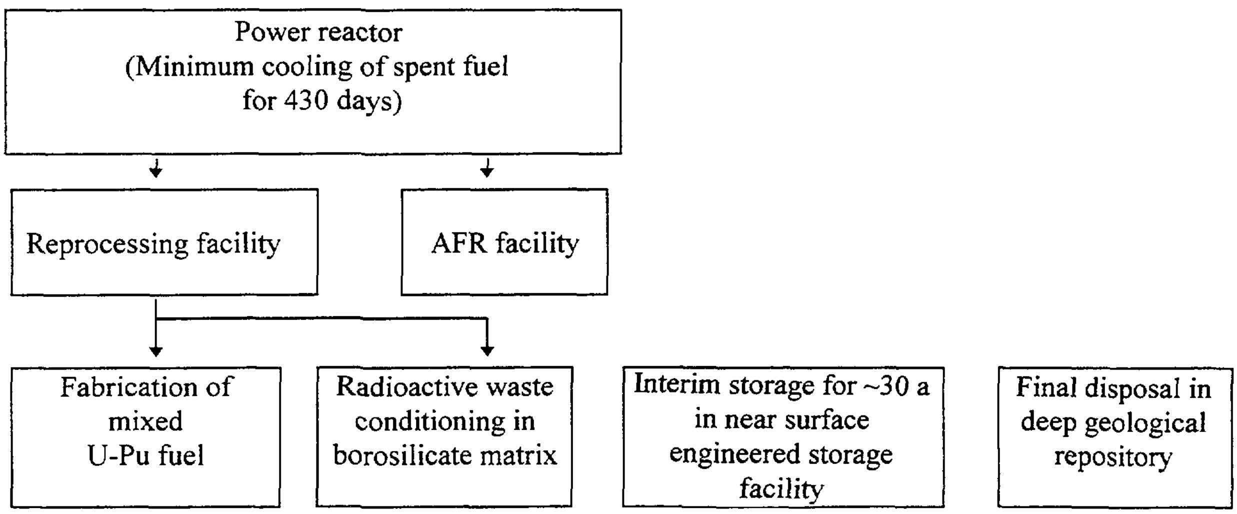


*FIG. 1. Spent fuel management in India*

## 2. SPENT FUEL ARISINGS

The reprocessing option adopted for spent fuel management would meet the fuel/fissile material demand for the country's Three Stage Nuclear Power Programme.

Most of the first generation Indian power reactors are using natural uranium dioxide as fuel. It is envisaged that the plutonium produced from the first generation reactors will be used, along with $U^{238}$ in the form of natural uranium and $Th^{232}$ in second generation fast breeder reactors. Fast breeder reactors will produce $U^{233}$ which in turn will fuel the third generation reactors.

The Indian Nuclear Power Programme began with the commissioning of two BWRs, TAPS 1 & 2 in 1969. These BWRs use low enriched uranium dioxide fuel, however, most of the reactors in our programme are essentially PHWRs using indigenous source of natural uranium. A number of 235 MW(e) and 500 MW(e) PHWRs are under construction or are being planned for execution to accomplish the nuclear power target of 3,200 MW(e) by the year 2004.

The commissioning of the 15 MW(e) Fast Breeder Test Reactor (FBTR) at Kalpakkam heralded the beginning of second stage of the nuclear power programme. Based on the experience gained with the FBTR, work on the 500 MW(e) Prototype Fast Breeder Reactor (PFBR) has begun. An Advanced Fuel Fabrication Facility (AFFF) has been built up at Tarapur to take up the MOX fuel fabrication on industrial scale.

India entered into the third stage of nuclear power programme by building the 30 kW KAMINI reactor at Kalpakkam using $U^{233}$ as fuel. The estimated spent fuel arisings are given in Table I.

TABLE I. OPERATING NPP AND ANNUAL SF DISCHARGES

| Name of Plant | Reactor type | Start of operation | Rated capacity MW(e) | Annual SF arisings t HM/a |
|---|---|---|---|---|
| TAPS 1 & 2 | BWR | 1969 | $2 \times 160$ | 21 |
| RAPS 1 & 2 | PHWR | 1973, 1981 | 100 and 200 | 45 |
| MAPS 1 & 2 | PHWR | 1983, 1985 | $2 \times 235$ | 60 |
| NAPS 1 & 2 | PHWR | 1989, 1991 | $2 \times 235$ | 60 |
| KAPS 1 & 2 | PHWR | 1992, 1995 | $2 \times 235$ | 60 |

## 3. SPENT FUEL TRANSPORTATION

Considerable experience has been acquired in spent fuel handling and transport. In India, SF transportation is carried out in specially fabricated and tested shielded cask by rail and road.

SF from TAPS is transported to the newly AFR-OS SF storage facility by road in SS clad and lead shielded cask of 70 t. Shipment of PHWR SF from RAPS/MAPS to the reprocessing facility or AFR is carried out by rail and road, with single shipment consisting of 3 casks each containing 220 spent fuel assemblies. Since the transportation is carried out in the public domain, the statutory requirements of transport safety, radiological safety, physical protection and nuclear material safeguards are fulfilled.

4. FUEL CHARACTERISTICS

CANDU type zirconium alloy-clad natural uranium dioxide fuel assemblies (Table II) are used to power Indian PHWR type reactors. These have proven well suitable for wet and dry storage with good mechanical durability. The durability is needed to meet the requirements of on-power reactor fueling as well as post-reactor handling for storage and transport operations. Fuel bundles are given a thin graphite layer (CANLUB) on the inside surface of the zircaloy cladding to prevent fuel-clad interaction. Fuel bundles are easy to handle and can be closely packed in simple storage containers and will not go critical in either wet or dry storage facilities regardless of the storage density. Zircaloy clad enriched $UO_2$ fuel is used to power the twin BWRs of TAPS 1 & 2.

TABLE II. BWR AND PHWR FUEL CHARACTERISTICS

| FUEL CHARACTERISTICS | BWR | PHWR |
|---|---|---|
| Number of elements per assembly/bundle | 36 | 19 |
| Cross section/diameter of fuel assembly/bundle (mm) | 110 ×110 Sq. | 81.69 |
| Length of fuel assembly/bundle (mm) | 3657.5 | 495 |
| Minimum thickness of zircaloy-2 cladding (mm) | 0.89 | 0.38 |
| Weight of uranium dioxide per assembly/bundle (kg) | 156.7 | 15.2 |
| Average discharge burn up MW·d/t HM | 20000 | 7000 |
| Weight of zircaloy per assembly/bundle (kg) | 21.7 | 1.51 |
| Type of fuel | Enriched $UO_2$ | Natural $UO_2$ |

5. SPENT FUEL STORAGE

**5.1. At-Reactor (AR) facility**

*5.1.1. BWR spent fuel assemblies*

The twin BWRs at TAPS are in operation since 1969 with an annual SF discharge capacity of 21 t HM/a at present rated capacity. The SF arisings from this station is presently kept under wet storage. To maintain the pool water chemistry, clarity and to remove the decay heat from SF to maintain the pool water below $42^0$ C, the pool water is recirculated through cleaning system comprising filter, ion exchange resin column and heat exchanger. The storage pool at the rector was initially designed to store 528 spent fuel assemblies (SFAs). Subsequently the capacity was increased to 1500 SFAs by reracking using high density racks. A separate wet storage facility, Away-From-Reactor (AFR) and On-site, has been constructed and made operational for storage of additional spent fuel assemblies from the reactor. During the interim period, when the AFR was under construction, spent fuel assemblies storage in Dry Casks (DC) was resorted to act as an interim measure.

*5.1.2. PHWR spent fuel assemblies*

RAPS 1 & 2 were the first two PHWRs to be commissioned in 1973 and 1981 in Rajasthan. The SFAs generated from these reactors are stored in SS trays in the spent fuel pool at the station. Pool water cooling and purification system operates continuously to maintain the temperature, water chemistry and activity within stipulated levels. The capacity of the fuel pool is adequate for about 10 years full power operation of the two reactor units. The storage capacity of the pool was increased by 30 % of its original capacity by reducing the spacing between each tray and by increasing the stack height. Concrete casks have been developed for dry storage of spent fuel assemblies with cooling period of more than 10 years. Presently, part of this fuel is stored in the AFR-OS facility at Tarapur, transporting in 70 t casks in dry condition.

All the twin PHWRs, at Kalpakkam (MAPS), Narora (NAPS), Kakrapar (KAPS) and Kaiga (Project) which are either in operation or under advance stage of completion are also provided with similar arrangement for spent fuel storage.

**5.2. At power reactor fuel reprocessing plant (PREFRE)**

*5.2.1. Facility description*

PREFRE plant at Tarapur is meant for reprocessing spent fuel from BWRs and PHWRs (see Fig. 2). The spent fuel is received in shielded casks on trailer-truck through air-lock room. There is a spent fuel pool at PREFRE plant for storage of spent fuel assemblies under demineralised water. pH of 6.8 - 7.2 and conductivity below 0.1 micro S/cm are maintained in the pool. Ion exchange beds are adequately shielded to reduce radiation level in the operating area. The pool capacity is about 63 t HM (21 cask loads) of PHWR fuel. Other details of the fuel pool are as follows :

| | | |
|---|---|---|
| (1) | Size | : 10 m(L) × 8 m(W) × 7.5 m(D)/6 m (D) |
| (2) | Clear water shielding | : 2.9 m |
| (3) | Total water volume | : 450 $m^3$ |
| (4) | Pool water turn over time | : 37.5 h |
| (5) | Fuel pool lining (Stainless Steel ) | : 3.2 mm. |

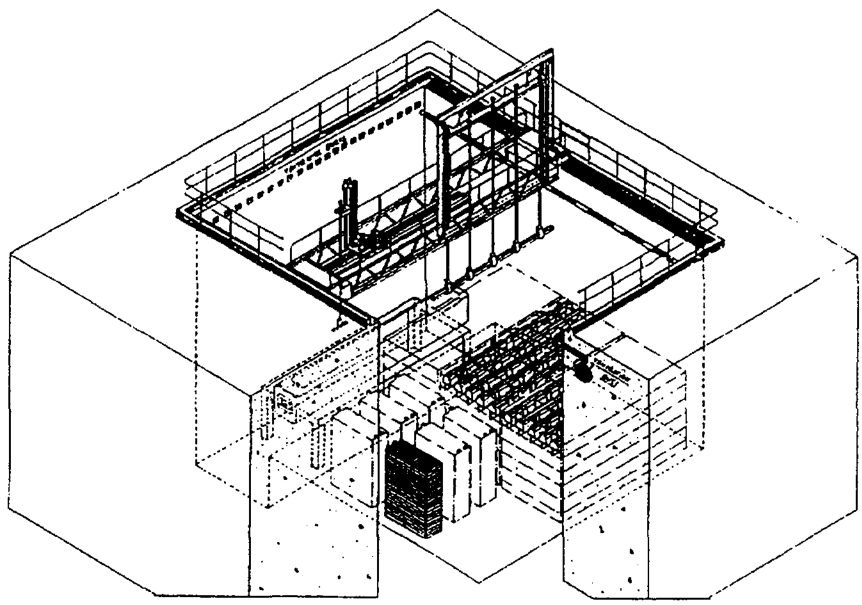

*FIG. 2. PREFRE spent fuel storage facility*

A movable bridge provided with five fixed and one sliding tong spans the top of storage pool. The tongs are meant for handling the fuel under water. 70 t Electric Overhead Travelling (EOT) crane is used to handle fuel casks in the area.

In order to sweep off traces of fission gases that may escape from defective spent fuel assemblies, a concept of surface ventilation system is used by locating supply and exhaust grills just above the fuel pool surface.

*5.2.2. Pool water clean-up improvements*

5.2.2.1. Candle type filter

The pool water clean-up system was improved by first replacing the conventional pressure sand filters with a glass wool cartridge type filters and then finally replacing with candle type screwed filters enclosed inside SS housing and placed under water. The system is working satisfactorily with water visibility maintained at very high level (turbidity below 0.2 ppm on silica scale). A high filtering efficiency of 99 % is obtained.

5.2.2.2. Fuel pool clean-up system

A pool clean-up unit consisting of a suction funnel, submersible pump set and four number of candle type filters was fabricated and is being used under water for removal of active loose fuel powder/dust from the pool floor. Remote under water operations have helped reducing background radiation and man-rem exposure. Cat-ion exchange resin bed was also introduced in pool water clean-up system. Table III gives experience on polishing fuel pool water.

TABLE III. POOL WATER ACTIVITY AND REGENERANT EFFLUENT DETAILS.

| Fuel stored | Average pool Water activity (Bq/ml) | Regenerant effluent details | | | Total activity per year (GBq) |
|---|---|---|---|---|---|
| | | No. of regenerations per year | Volume ($m^3$) | Average $\beta,\gamma$ activity (MBq/l) | |
| BWR/PHWR | 37 - 111 | 13-24 | 180-350 | 7.4 - 11.0 | 1,665 - 3,330 |
| PHWR | 16.7 - 27.8 | 3-12 | 50-160 | 7.4 - 14.8 | 740 - 2,220 |
| PHWR [a] | 3.7 - 18.5 | 1-3 | 15-45 | 7.4 - 14.8 | 185 - 555 |

[a] cat-ion exchange resin bed introduced

## 6. AFR-OS SPENT FUEL STORAGE FACILITY

The AFR-OS spent fuel storage facility commissioned at Tarapur for storing the spent fuel discharged form the BWRs for the entire life period of the station, is a wet storage facility with a capacity to store 2,000 SFAs with a provision to increase to 3,312 SFAs.

The fuel pool is 9 m (W) × 13 m (L) × 13 m (D). The depth accounts for 4.4 m fuel height, 5 m for cask height, 2.6 m for shielding and 1 m free board. Radiation shielding around the pool is provided by 1.5 m thick concrete wall. The pool is 5 m below the ground level and 8 m above ground and is lined with SS 304 L plate. The concrete structure has been tested for leak tightness. A leak detection system has been provided. SFAs are transported from TAPS to AFR in shipping/dry storage casks on trailer-truck. The dry storage cask is designed for 37 SFAs having a minimum burnup of 13,000 MW·d/t HM and a cooling period of minimum 10 years. The entire AFR facility comes under IAEA safeguards.

The decay heat of each spent fuel assembly is estimated as 0.1 kW. The fuel pool is provided with a cooling cum clean-up system to maintain pool water quality, clarity and temperature below 42$^0$C under normal operating conditions.

Further handling of the cask is done by a 80/10 t EOT crane. Then the fuel pool bridge with hoist of 1 t capacity will handle spent fuel assemblies one at a time for placing them on the storage rack of 12 × 12 array.

The maximum activity release in the pool with 2,000 SFAs is expected to be 25.9 GBq/d based on TAPS experience. For maintaining water quality, a water polishing plant of 30 m$^3$/h capacity is provided. Cartridge filters will remove particulate matter down to 10 μm size. A make up demineralised water plant of capacity 0.5 m$^3$/h is installed to cater the requirements such as evaporation loss of water, decontamination etc.

Air flow is maintained over the pool water surface (velocity of 2 m/min) by means of supply/exhaust ventilation system. This also causes surface evaporation and enhances the cooling effect. The supply air filter efficiency is 90% down to 10 μm size. Prefilters having an efficiency of 99% down to 5 μm size and absolute filters having an efficiency of 99.9% down to 0.3 μm are provided on the exhaust side. Fig. 3 shows the line diagram of the AFR facility.

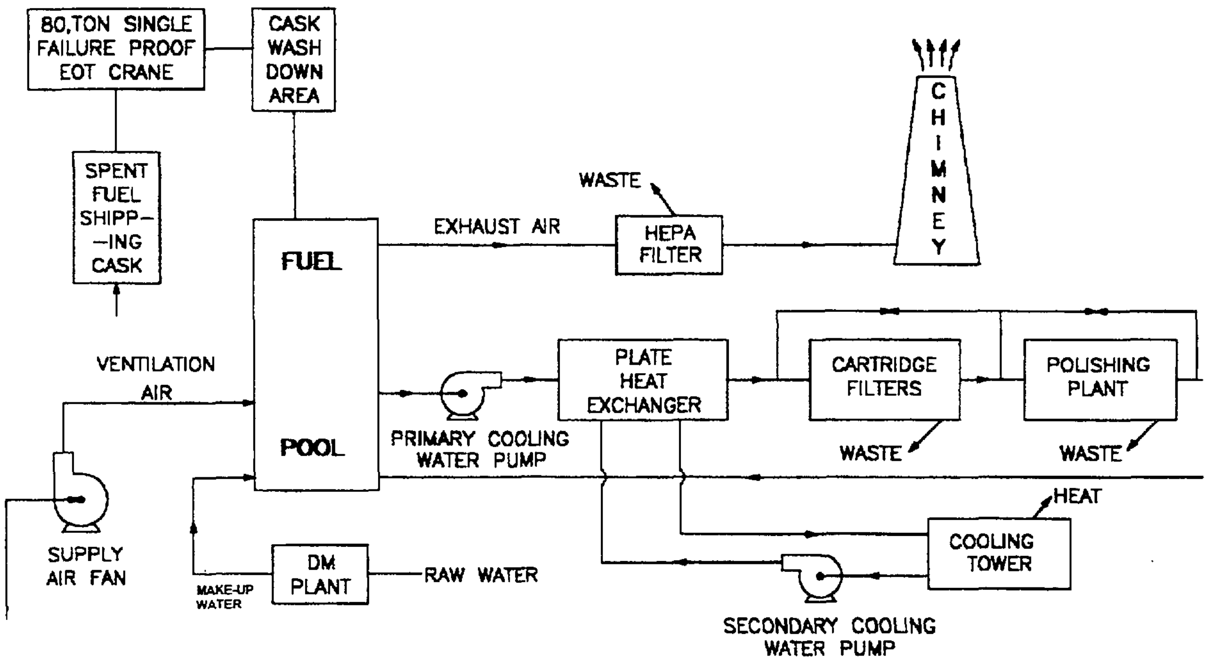


*FIG. 3. Line diagram of AFR spent fuel storage facility*

## 7. DRY STORAGE

Two types of dry storage casks (DSC) have been developed for storage as well as transportation of the spent fuel.

As the DSCs are modular, passive, easily constructible and comparatively of low cost, these have been adopted as 'add on' system for additional storage to the fuel storage pools. Dry storage of spent fuel assemblies from BWRs at TAPS was taken up as an interim measure when the available storage capacity in the AR facility was fully utilised. The DSC (Fig. 4) is designed to store 37 BWR spent fuel assemblies with a burnup greater than 13,000 MW·d/t HM and a minimum cooling period of 10 years. Strict administrative control is exercised to ensure the compliance of these parameters. Square boxes made of 3.25 mm thick SS 304 plates are used to support 37 spent fuel assemblies.

These plates also act as neutron poison and keep them in subcritical condition. Four such casks are in use at TAPS without any external cooling. These DSCs also have been used for transportation of the spent fuel from the reactor pool to AFR pool.

In view of high inventory build up of spent fuel, concrete casks have been developed for PHWR fuel from RAPS 1 & 2 and can accommodate 220 Nos. of 10 year cooled spent fuel assemblies with a maximum burnup of 10,000 MW·d/t HM. The cask has 750 mm thick reinforced concrete shielding on all sides and 850 mm at the bottom. The cask is lined with 6 mm thick steel plate both inside and outside and is designed to withstand mechanical stresses to control cracking of concrete. The total weight of the cask, when fully loaded with spent fuel assemblies, is 60 t.

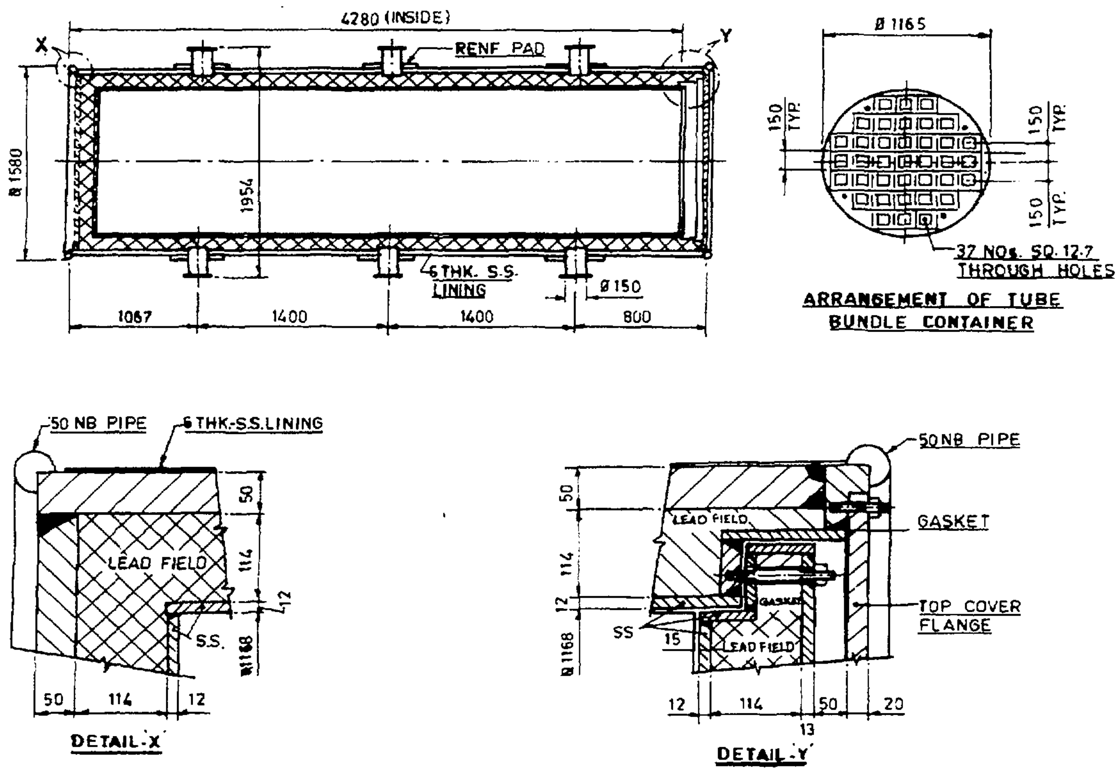


*FIG. 4. Dry Storage Cask for TAPS Spent Fuel*

## 8. CONCLUSION

Experience with wet storage systems in India is quite satisfactory wherein slowly all the problems like high pool water activity, frequent regenerations, personnel exposure etc. have been solved. The pool water activity can be maintained below 18.5 Bq/ml.

Storage of spent fuel assemblies in water pools remains as the prevailing mode for the near term. India has gained ample and successful experience from designing, constructing and operating various spent fuel storage facilities like AR, AFR and DSCs and transporting the spent fuel as per procedures laid down by regulatory bodies. With increase in spent fuel arisings due to planned growth of nuclear power generation, the reprocessing capacity is being increased to meet the future demand of fuel and to reduce spent fuel storage load. Co-location of adequate capacity reprocessing plants at reactor sites is planned to minimise the transportation through public domain.

# SPENT FUEL MANAGEMENT STRATEGY IN ITALY



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**Abstract**

As a consequence of a national referendum in 1987, the Italian Government decided to close definitively all operating NPPs in Italy. Plans for decommissioning of the NPPs and disposal of the spent fuel had to be reviewed and the strategies revisited. The majority of spent fuel was by large that generated by ENEL NPPs, which decided to proceed with the interim storage of the spent fuel (< 250 t/HM) not covered by reprocessing contracts. ENEL finally decided to follow the strategy of interim dry storage in metallic casks on the plant sites, which could ensure a timely removal of the fuel from the to be decommissioned plant pools, in compliance with decommissioning programmes, independently from the availability of a centralized interim storage site. Therefore, the casks will be stored provisionally on Trino and Caorso sites, then they will be transported to the centralized interim facility, as soon as it will be made available by the Government. Current planning foresees that the Trino spent fuel pool shall be emptied by the end of 2002 and the Caorso pool at the end of 2004. An international bidding phase is currently underway. A smaller residual quantity of spent fuel is also currently owned in Italy by ENEA, the National Agency, responsible also for the nuclear research. Also ENEA has a programme of storing its spent fuel in dry metallic casks with the aim of transporting them to the national storage site as soon as it will be available. ENEL's Technical Specifications for the casks are stringent, but in line with other European installations of the same type, taking into account also recent USNRC regulatory documents, in particular on protection against aircraft crash. Design margins to accommodate site characteristics not currently identified (the European Utility Requirements reference site parameters have been used), have been introduced. Some important issues are identified, such as: definition and identification of failed fuel elements and/or pins, specific requirements for MOX fuel, cask loading optimization, possibility of burnup credit in some circumstances, source term in all operating conditions, environmental impact and occupational doses in normal operation and accident conditions.

## 1. INTRODUCTION

As a consequence of a national referendum in 1987 the Italian Government decided to close definitively all NPPs operating in Italy. This decision affected directly and mainly ENEL, as the only Italian utility and owner of all NPPs, but evidently also the Italian industry as well as other nuclear structures and activities, such as ENEA, the National Agency with specific responsibilities in the nuclear R&D, including activities in the field of fuel cycle.

At that date, 4 NPPs were officially in operation, i.e. Caorso (a BWR of 870 MWe), Trino Vercellese (a PWR of 270 MWe), Latina (a Magnox of 160 MWe) and Garigliano (a BWR of 160 MWe, which, however, was not operating since 1978 because of plant modifications). Construction of two other units (Alto Lazio, BWRs of 1000 MWe) was almost complete (about 70%) and activities for the construction of 2 additional units at the Trino site (standard 1000 MWe PWRs) were started. All these plants and activities were immediately suspended and never restarted again.

One of the important consequences of this decision is related to the complete decommissioning of the NPPs, to waste and spent fuel management strategies, which had to be adapted to the early closure of the plants and to a prolonged phase-out of the use of nuclear energy in Italy for electricity production. Strategies were subject to reconsideration and changing. In particular, in this paper the consequences on spent fuel management are covered.

At that time, ENEL's strategy for spent fuel was reprocessing and contracts were in place with BNFL. At the beginning of 1990, ENEL decided on the basis of technical and economical evaluations, while honouring the reprocessing contracts already in place, to change the policy and to proceed with the interim dry storage of the remaining spent fuel. It was also recognized that, in the light of the Italian situation, reprocessing would not have brought relevant advantages in terms of final disposal, since vitrified HLW would have to be disposed anyway, together with other reprocessing generated wastes and their form would not have implied specific advantages in a final disposal situation, where only geological barriers could be credited. In addition the question of uranium and plutonium disposal would have remained open. ENEL does not believe at this point that even fuel rod consolidation will be necessary at the time of final disposal, but, also in this case, the questions of economics would remains in favour of interim storage.

After defining the new policy, ENEL started to search for sites where to store the spent fuel. Initially, ENEL's target was to collect all ENEL's spent fuel for interim storage in a single, privately owned site, with the potential of storing, if requested, also ENEA's fuel, and later on vitrified HLW. On that assumption and starting in 1995, a vault type storage facility was selected, conceived and engineered by the GENESI consortium, on the basis of the SGN technology in co-operation with ENEL. However, ENEL, in the absence of a definitive proof of private site licence availability and, on the other side, in the light of the commitment of the Government to provide a national interim storage facility by 2008, finally decided to stop this project and to follow an alternative strategy, which could assure a timely removal of the fuel from the plant pools, in compliance with the decommissioning programmes, independently from the immediate availability of a centralized interim storage site.

## 2. ITALIAN SPENT FUEL INVENTORY

As mentioned above by large the majority of the spent fuel has been generated by ENEL. In the past, the following spent fuel amounts have been sent to reprocessing:

TABLE 1. ENEL'S SPENT FUEL QUANTITIES SENT TO REPROCESSING

| Plant | t/HM |
|---|---|
| Latina | 1,425 |
| Garigliano | 44 |
| Trino | 107 |

All the Latina generated spent fuel has been sent to reprocessing, as required by Magnox fuel, and, therefore, no residual Latina fuel is present in Italy. No Caorso spent fuel has been sent to reprocessing. It shall also be noted that, according to current plans, additional 53 t/HM of the Garigliano fuel will be sent to reprocessing in the framework of completing current contracts.

The residual quantities considered for dry storage are indicated in Table 2. The locations indicated are the spent fuel pools in the Caorso and Trino stations and the independent spent fuel pool, located in the containment of the former "Avogadro" research reactor, which has been already dismantled. The "Avogadro" installation is located in the area of Saluggia town, a few kilometers away from the Trino site as indicated in Fig. 1.

TABLE 2. ENEL'S RESIDUAL SPENT FUEL QUANTITIES

| Plant | Current location | Reactor type | Fuel type | N° elements | t/HM |
|---|---|---|---|---|---|
| Garigliano | Avogadro | BWR | MOX | 63 | 12.9 |
| Trino | Trino | PWR | $UO_2$ | 39 | 12 |
| " | Avogadro | | " | 49 | 15.1 |
| " | Trino | | MOX | 8 | 2.5 |
| Caorso | Caorso | BWR | $UO_2$ | 1,032 | 190.4 |
| TOT | | | | 1,191 | 232.9 |

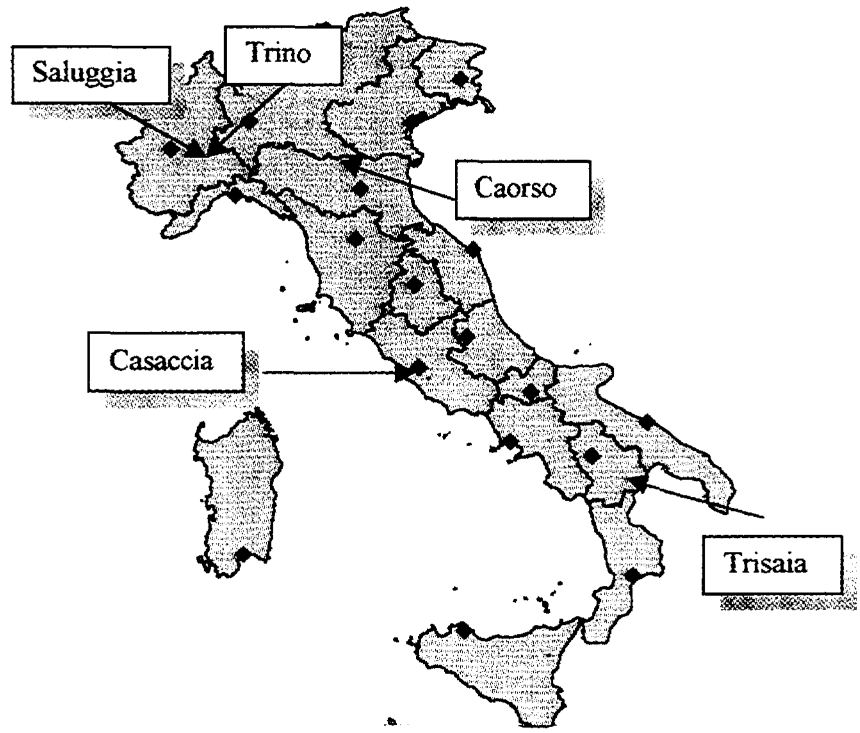


*FIG. 1. - Current spent fuel locations in Italy*

In addition to the above fuel quantities, which are currently stored in wet pools in Italy, there are to be considered also the fuel that will return from the Creys Malville (FBR) as part of the ENEL liabilities (about 1/3 of the spent and fresh fuel). And finally consideration shall also be given to the management of the vitrified HLW generated in the reprocessing activities and that will return to Italy.

ENEA had in the past an important R&D programme in the nuclear field, including research reactors, hot laboratories and facilities in the area of fuel cycle. In these activities ENEA owned directly some fuel for research reactor operations and received from ENEL some other spent fuel assemblies or fuel rods for research purposes. The current situation is described below.

The ENEA spent fuel is mainly stored in the EUREX plant (Saluggia) and ITREC plant (Trisaia), where fuel cycle R&D activities have been stopped. The total weight of the spent fuel is about 3.7 t and consists of:

- 52 cruciform, non-standard Trino PWR elements (EUREX);
- 64 Elk River (ERR) fuel elements (ITREC);
- 1 Garigliano fuel element (EUREX).

In addition, about 90 kg of various irradiated materials (fuel pins, fuel assembly sections, etc.) is stored at the Casaccia Research Center and 142 fuel elements of the, still operating, TRIGA reactor.

Finally, at the EUREX facilities 150 MTR elements were stored, but they have been transported to the US in the framework of the agreement with DOE for fuel of US origin. It is expected that also the TRIGA fuel will be shipped to the US in accordance with the current DOE policy. Discussions are underway to ship also the Elk River fuel to the US.

## 3. CURRENT STRATEGY FOR SPENT FUEL INTERIM STORAGE

The current ENEL strategy for spent fuel management is based on the following considerations:

- the residual spent fuel quantity is rather limited (< 250 t/HM);
- the Government is committed to make a national Centralized Interim Storage Facility available by the year 2008;
- removal of the spent fuel from the pools of the plants to be decommissioned is a vital precondition to proceed with the decommissioning activities and to reach a safe storage condition.

On the basis of the above considerations, ENEL defined its strategy of storing all the fuel (including the fuel in the "Avogadro" pool) on its sites of Trino and Caorso for the time needed to make the National Interim Storage Facility available. This would allow to decontaminate and possibly dismantle most of the current plant systems for handling the fuel (e.g. the spent fuel pool), in the process of the decommissioning activities for safe storage.

On these bases, it appeared from initial feasibility studies and cost evaluations that the solution of the metallic casks would have been the most suitable for the above conditions. The casks shall be "Dual Purpose", i.e. they have to be approved for storage and transport, allowing their transportation off site, as soon as the centralized storage facility will be available, without further direct manipulation of the spent fuel itself. Temporary installations on the plant sites would have no interference with the decommissioning activities and no further waste would be generated by their removal.

Since the storage on ENEL sites will be limited to a few years, consideration has been given not to build a new sheltering building, but to make use of an existing building with all the necessary adaptations. This will require civil works and mechanical components modifications and specific solutions for cask handling, so that a new building in each site would have been convenient in principle on strict economical bases. However, it is expected that a new building would have implied additional licensing delays and, on top of that, the impression to the public of a more permanent installation. Despite of the difficulties in these evaluations, consideration has also been given to maximize crane capacities, to reduce the number of casks and to minimize the required storage area.

ENEL has published a notice for prequalification of potential bidders for cask supply in the European Journal in April '98 and in July it has sent the Request for Bids to the qualified companies. Currently, ENEL is in the phase of technical bid evaluation. Planning foresees that the Trino spent fuel pool shall be emptied by the end of 2002 and the Caorso pool at the end of 2004. The time schedule is very tight, but discussions will go on with the potential suppliers on actions to maintain the target dates for emptying the fuel pools. A very schematic schedule is presented in Fig. 2.

Starting from 1990, ENEA has also investigated possible alternatives for the removal of the spent fuel from its facilities to be dismantled. Reprocessing was considered only for the Trino and Garigliano low enrichment fuel elements, but costs appeared overwhelming in comparison with other solutions. Therefore, a strategy of dry storage in dual purpose metallic casks, similar to the ENEL choice, has been adopted by ENEA. A special cask TN 24, supplied by Transnucleaire, has been ordered for the Elk River fuel, while discussions are underway to verify the possibility of sending also this fuel to US. A second cask is planned for all fuel elements of the EUREX facility. Casks will be of the same design and will differ only in the internal basket. It is expected, that this will facilitate both licensing and construction activities.

Currently, ENEA is planning to store its casks on the Saluggia and Trisaia sites, where the fuel are currently stored. However, ENEL, if required and approved by the Authorities, may accept the provisional storage of the ENEA casks on its sites, in order to reduce the number of spent fuel storage sites, as a contribution to an optimized solution for the spent fuel management in Italy.

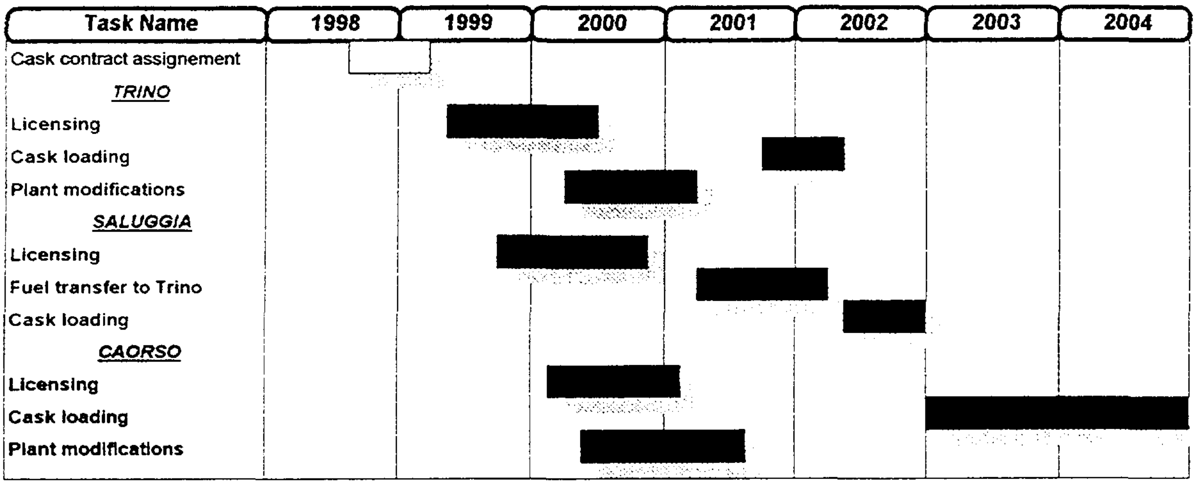


*FIG. 2. General schedule for ENEL spent fuel storage*

## 4. TECHNICAL REQUIREMENTS AND CURRENT ISSUES

In this section a few important technical issues are presented, which are to be considered in the supply of the storage casks. They have been defined by ENEL in the framework of the project for its spent fuel. They are briefly described below, because they could be, at least in part, non standard and, therefore, of interest to other utilities.

First of all we describe a few specific elements that imply some additional complexity to the ENEL project. A first element is that we are in the process of defining cask design parameters without knowing exactly the site characteristics of the centralized national Interim Storage facility. This implied that we chose envelope site parameters. Decision has been made that site parameters of the European Utility Requirements for standard European advanced plants were the reference, since representing an envelope of sites potentially suitable for a new NPP installation.. We are confident that these data will be representative or enveloping also those of possible sites for cask storage.

TABLE 3. REFERENCE DESIGN ENVIRONMENTAL CONDITIONS

| PARAMETER | VALUE | CORRESPONDING PERIOD |
|---|---|---|
| **External air temperatures & humidity conditions** | | |
| • Maximum air design temperatures | | |
| Long-term base | 32 °C | > 7 days |
| Short-term daily | 37 °C | 6 hours to 7 days |
| Instantaneous | 42 °C | 6 hours |
| • Minimum air design temperatures | | |
| Long-term base | - 15 °C | > 7 days |
| Short-term daily | - 30 °C | 6 hours to 7 days |
| Instantaneous | - 35 °C | 6 hours |
| • Design-basis maximum external humidity | | |
| Summer | 60 % at 37 °C (dry bulb) | |
| Winter | 100 % at - 15 °C | |
| **Wind and Wind-Generated Missiles :** | Maximum values | |
| • Basic wind speed | 43 m/s | |
| • Extreme wind speed | 60 m/s | |

A second element is that the type of fuel is rather heterogeneous. Different BWR and PWR fuel, both $UO_2$ and MOX, are included, so that at least 5 different fuel assemblies can be identified. It will be necessary to study a possible optimization of cask loading in order to minimize the number of casks. In addition, there are some known "leaking" elements that shall be loaded in the casks with special precautions, including the use of special canisters. For the reason explained above, they shall be loaded together with other elements and cannot be left in the pools as other utilities with reactors still operating can do. A programme for complete fuel integrity characterization is under review. Criteria for defining unacceptable cladding defects and criteria for storage of defected pins are currently under evaluation.

A third element is the choice of using existing buildings on plant sites. The best choice, also from an economical standpoint, would have been probably to build a new storage building as said above. Existing limitations include: height of the building, handling capacity, foundations and wall shielding capability .

A fourth element is the limitations with existing plant handling capacities. In Trino Vercellese, the current linear crane in the fuel building is limited to 68 t, while the Caorso reactor building polar crane is currently limited to 85 t as Maximum Critical Load (MCL). Both these figures are well below the current standard total cask weight of about 120 t. Studies and cost benefit analysis are currently underway on the possibility of increasing in both cases the handling capacities. On the basis of available information, it could be said, in general, that an increase in the total weight of the cask would greatly favour the net fuel "payload", since the weight of the cask itself would increase in a smaller percentage. This means, for example, that if one increases Caorso capacity to 100 t, the increase of the fuel content is expected to be greater than 15-20%.

A fifth element is the complexity of the licence process, involving two sites. In addition, a separate licensing process is considered for obtaining the approval to move to the Trino sites fuel elements that are currently in the "Avogadro" pool.

Finally, an additional element is that the acceptance criteria for the final disposal have not be defined yet. Therefore, all the specifications are based on the assumption of full fuel retrievability at the end of the cask life. This means essentially that fuel geometry and structural integrity degradation shall not occur in order to facilitate fuel retrieval in view of their final storage.

Other ENEL requirements included in the Technical Specifications for the casks are demanding, but, in general, they reflect both the Italian specificities and recent, European, US and international regulatory positions.

Despite the apparent simplicity of the a metallic cask, the assurance of a flawless and troubleless life for several decades and the considerations of extreme conditions both in storage and in transport conditions, as well as in extreme accidental conditions, brings some complexity in the design. However, the greatest attention shall be given to the fabrication process, which has been historically the source of several problems, at least with some vendors.

In the following, some important design criteria, included in ENEL's specifications, are presented. It shall be underlined, that these criteria are to be finally approved by the Italian Safety Authority (ANPA) and discussed with the bidders.

The design life of the casks shall be at least 50 years. The possibility to prolong the design life will be discussed. However, a limitation for that will be the demonstration of the fuel degradation in dry storage for such a duration, in the light of the current condition of assurance of retrievability at the end of the cask life.

Adequate specifications have been indicated in order to limit both the residual humidity content inside the casks and the maximum cladding temperature during normal conditions. Assurance of a low temperature of the cladding will be facilitated by the long decay time of all fuel elements, in some cases exceeding 20 years. Since the decay heat is very limited in most of the fuel assemblies, a packed pitch can be considered from a thermal standpoint.

There is the need to assure also the highest safety levels in terms of criticality. In principle, burnup credit in criticality calculations could be advantageous or even needed to support the tight pitch. This could require some detailed characterization of some spent fuel assemblies.

The consideration of MOX fuel might have important consequences in the thermal, shielding and criticality analysis and optimization of cask loading will be analyzed.

Aircraft crash as well as external explosions shall be considered among the severe design events. The same forcing function which is reported by EUR for future plants will be applied to casks (see Fig. 3). In Fig. 4 also the external explosion pressure loading is reported, even if it is expected that for the casks this will be a much less severe event.

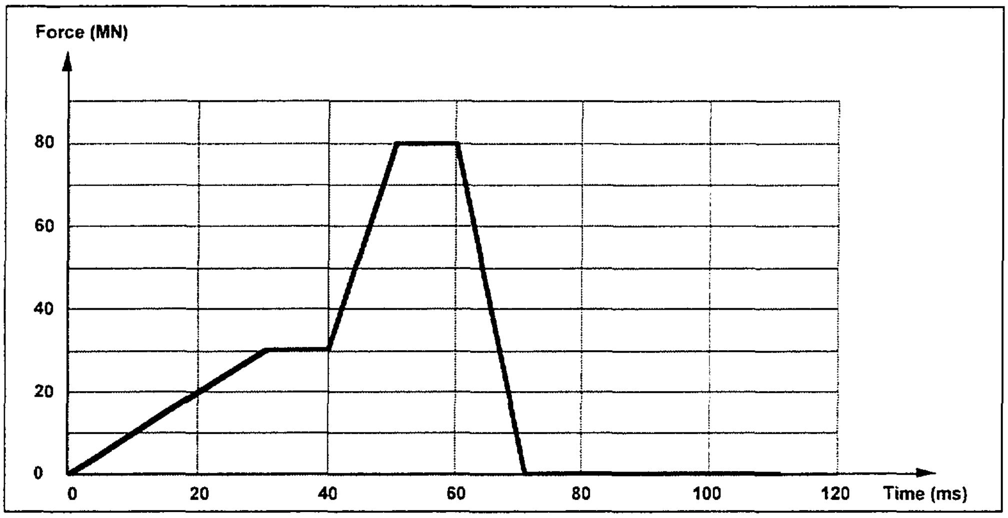


*FIG. 3. Aircraft crash loading function*

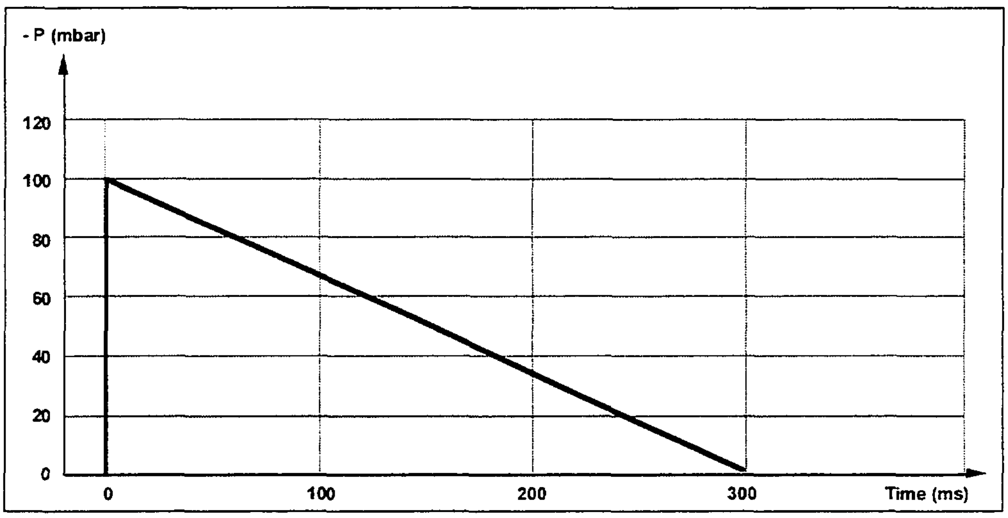


*FIG. 4. Design pressure wave*

Fuel shall be protected and environmental consequences controlled, also in case of collapse of the sheltering building, which may occur not only in case of the above mentioned events, but also in case of extreme seismic events. Therefore, the concept is base on the idea that essentially all the "technology" is within the casks and that the storage building is only a "sheltering" building with additional functions of shielding and improving the ventilation. This is more or less a standard approach in Europe and in the US, where casks may be stored even in an open space on a concrete pad.

Cask maintenance in normal and abnormal conditions shall be totally independent from the structure of the power plants. Plants will be preparing for safe storage and therefore, for example, pools will not be available any more for cask discharge. Solutions are available also in case of a seal leakage and a study of the procedures will be necessary. A cask double independent sealing barrier is required and their integrity shall be continuously monitored. IAEA standards have also been taken into account and referenced, and in particular the transportation requirements of the ST-1 standard [2] are required. Environmental impact and occupational doses in normal operation and accident conditions. ENEL's specifications are the following:

TABLE 4. RADIOLOGICAL TARGETS AND LIMITS

| | | **Normal conditions** | **Abnormal conditions** | **Accident conditions** |
|---|---|---|---|---|
| Contact dose (mSv/h)[a] | Target | 0.2 | - | - |
| | Limit | 2 | - | - |
| Dose rate @ 2m (mSv/h) | Target | 0.1 | - | - |
| | Limit | 0.2 | - | - |
| Max. dose rate @ 1m (mSv/h) | | | - | 10 |
| Max. dose rate @ building walls (μSv/h) | | 0.4 | - | - |
| Max. neutron flux from the building ($n/cm^2$ s) | | 1 | - | - |
| Worker annual individual doses (mSv/y) | Target | 5 | - | - |
| | Limit | ICRP 60 | - | - |
| Individual population dose[b] (mSv/y) | | 0.01 | 0.1 | 10[c] |

[a] On lateral surfaces.
[b] From all pathways to an individual standing @100m from the wall of the sheltering building without protection.
[c] Target is 5 mSv/y. If target cannot be met, justification shall be provided.

## 5. CONCLUSIONS

In conclusion, interim storage of spent fuel in metallic dual purpose casks is the current strategy for all the Italian spent fuel, owned both by ENEL and ENEA. This is the most flexible and also cost effective solution in the current specific Italian condition.

As a utility committed to solve the liabilities connected with the closure of its NPPs, ENEL is considering the interim storage of its spent fuel as a vital component of the entire strategy of plant decommissioning and a tight schedule associated with important resources have been assigned to this project. Design criteria both for the casks and for the storage facilities have been defined and are now in the process of being formally licensed by the Safety Authorities on the basis of the cask actual design. However, a preliminary consensus has been assured.

A final remark is that the question of the spent fuel final disposal remain open in Italy. Time for a decision is available, but it appears now that a national repository is not economically justified for the extremely reduced amount of fuel compared to other country inventories. Attention is given, therefore, to various initiatives currently underway for proposing the creation of regional repositories, which could bring both economical, safety and security advantages to the international community and to their single members.

## REFERENCES

[1] IAEA Safety Series N° 116, Design of spent fuel storage facilities, Vienna (1994).

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[3] USNRC - NUREG 1536, Standard Review Plan for Dry Cask Storage Systems, 1/97

[4] USNRC - NUREG 1567, Standard Review Plant for Spent Fuel Dry Storage Facilities, (draft for public comments), 12/96.

[5] USNRC - NUREG 1571, Information Handbook on Independent Spent Fuel Storage Installation, 12/96.

[6] ANSI/ANS 57.9, Design criteria for an independent spent fuel storage installation (dry type), (1992).

[7] ASME Boiler&Pressure Vessel Code, Section III, Division 3, Containment Systems and Transport Packagings for Spent Fuel and High Level Radioactive Waste, (1998).

# SPENT FUEL MANAGEMENT IN JAPAN



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**Abstract**

The numbers of current nuclear electric generation together with spent fuel arising are updated and their future predictions are made. Spent fuel discharged from LWRs is now stored at reactor sites, waiting for being reprocessed in the future. Meanwhile, it is pointed out that the interim spent fuel storage facility should be constructed and commissioned by around 2010, to accommodate superfluous spent fuel from nuclear power stations. Recovered plutonium is currently scheduled to be used in LWRs as MOX fuel and ultimately to be burned in FBRs in accordance with the *Long-Term Programme for Research, Development and Utilization of Nuclear Energy*.

## 1. INTRODUCTION

Because Japan has scarce energy resources, Japan has made efforts to utilize nuclear power since the mid 1950s. The Japanese basic nuclear policy is to reprocess the spent fuel and use recovered plutonium and uranium as fuel to achieve a stable supply of energy for the future and to minimize the environmental impact.

Currently, 52 nuclear power plants are being operated in Japan. The total nuclear electric generation capacity of these plants is 45,082 MWe, representing about one third of the total electricity generation. The nuclear electric generation capacity is expected to increase to 70,500 MWe in 2010. Fig. 1 shows the location and status of all nuclear power plants.

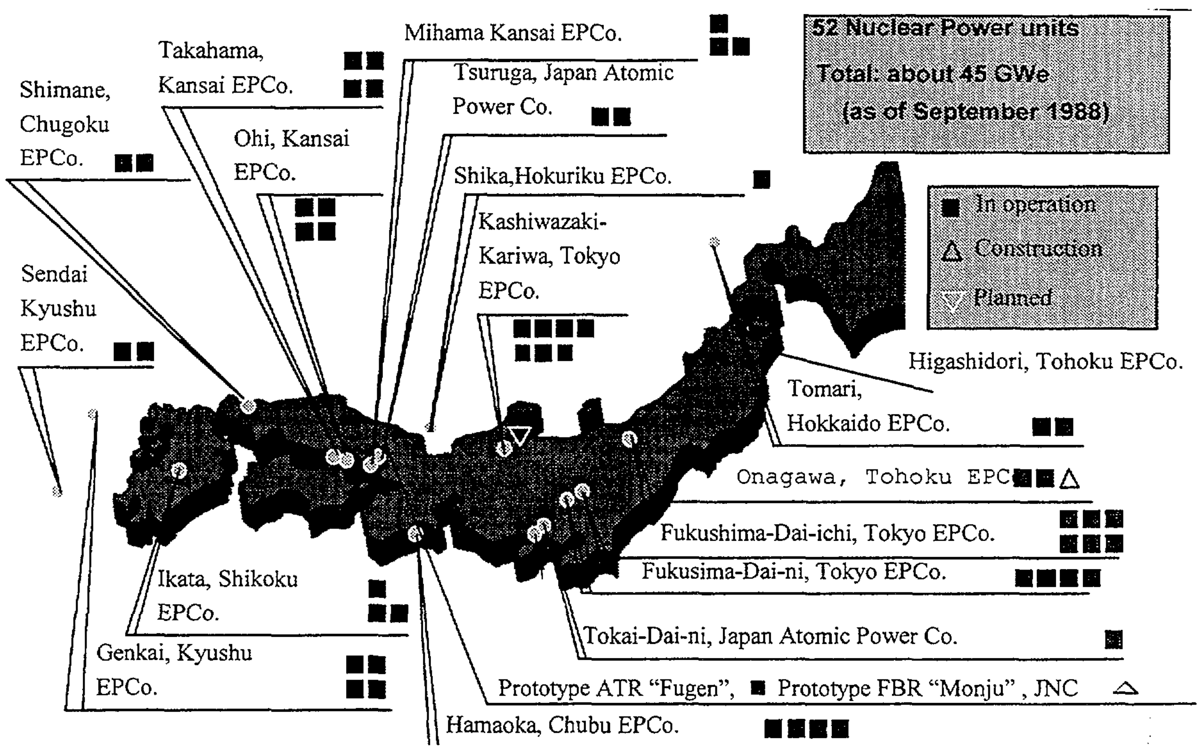


*FIG. 1. Current status of nuclear power generation in Japan*

## 2. POLICY STATUS ON SPENT FUEL MANAGEMENT

The *Long Term Programme for Research, Development and Utilization of Nuclear Energy* was revised by the Atomic Energy Commission of Japan (AEC) in June 1994. It describes "it is one of Japan's basic nuclear energy policies to reprocess spent fuel in order to be able to utilize the recovered plutonium and uranium". And it also describes "since spent fuel is considered to be a valuable quasi-domestic energy resource containing plutonium and uranium, the quantities of it in excess of domestic reprocessing capacity will be appropriately stored as an energy resource stockpile until such time as they can be reprocessed".

Nuclear fuel recycling is promoted on the principle of possessing no plutonium beyond the amount required for the programme, namely, the principle of no surplus plutonium as well as very strict management of nuclear materials, coupled with transparency so as to provide assurances regarding adherence to non proliferation of nuclear weapons.

After the sodium leakage of the secondary system of the prototype FBR MONJU in December 1995, the government had to make efforts to build a national consensus on the nuclear fuel cycle policy and to promote the disclosure of information and the participation of the general public in the process of deciding on policies. Standing on these efforts, the Cabinet issued a statement on "Policies to Promote the Nuclear Fuel Cycle" in February 1997. In this policy statement, commitment is given to steady promotion of the reprocessing programme for the plant under construction in Rokkasho, as well as to the promotion of nuclear fuel cycle through the following policy measures in the short term:

(1) **Plutonium utilization in LWRs**
- Start utilization with three or four reactors loading MOX fuel by 2000. Expand use of MOX fuel to ten odd reactors by around 2010.

(2) **Spent fuel management**
- Store spent fuel appropriately as an energy source until reprocessed. Immediate measures are necessary in some existing nuclear power plants to expand their storage capacities with the understanding of the local public.
- Initiate a study aiming at an early conclusion on the development of necessary environment to enable spent fuel to be stored at away-from-reactor sites by around 2010, in addition at reactor sites, with the increasing quantities of stored spent fuel expected in the long term prospect.

(3) **Back end measures**
- Present a total vision of disposal measures aiming towards the smooth implementation of final disposal of high-level radioactive waste, through a broad range of discussion, in the social and economic aspects.
- Put in place the institutional infrastructure necessary for decommissioning nuclear facilities.

(4) **Development of FBRs**
- Discuss future FBR development strategies, including treatment of MONJU by a Special Committee on FBRs established under the AEC.

## 3. CURRENT STATUS AND FUTURE PROSPECTS OF SPENT FUEL MANAGEMENT

As of the end of September 1998, 51 commercial nuclear power stations (26 BWRs, 2 ABWRs, 23 PWRs) and a Prototype ATR are in operation in Japan and the total electric generation capacity of these plants is 45,082 MWe. Nuclear electric generation capacity in 2010 is expected to grow to 70,500 MWe.

At present, the annual amount of spent fuel arising is about 900 tU/year. Part of spent fuel generated in Japan has been transported to overseas reprocessing operators, BNFL and COGEMA. The shipment of LWR spent fuel to the UK and French reprocessing plants started in 1973 and the final shipment was completed on 7th September 1998. The accumulated amount of shipment was about 5,630 tU. The accumulated amount of shipment of GCR spent fuels to the UK reprocessing plant was about 1,310 tU, and the remaining about 190 tU will be shipped hereafter (see Fig. 2). Plutonium recovered from Japanese spent fuel at the UK and French reprocessing plants will be fabricated to MOX fuel at the European nuclear fuel fabricators and returned to Japan to be utilized in LWRs starting from 1999. It is estimated from the projected power generation capacity that the annual rate of spent fuel arising will be 1,400 and 1,900 tU/year respectively, in the years around 2010 and 2030 (Fig. 3). As of March 1998, the cumulative amount of generated spent fuel was about 14,700 tU (excluding ATR fuel)

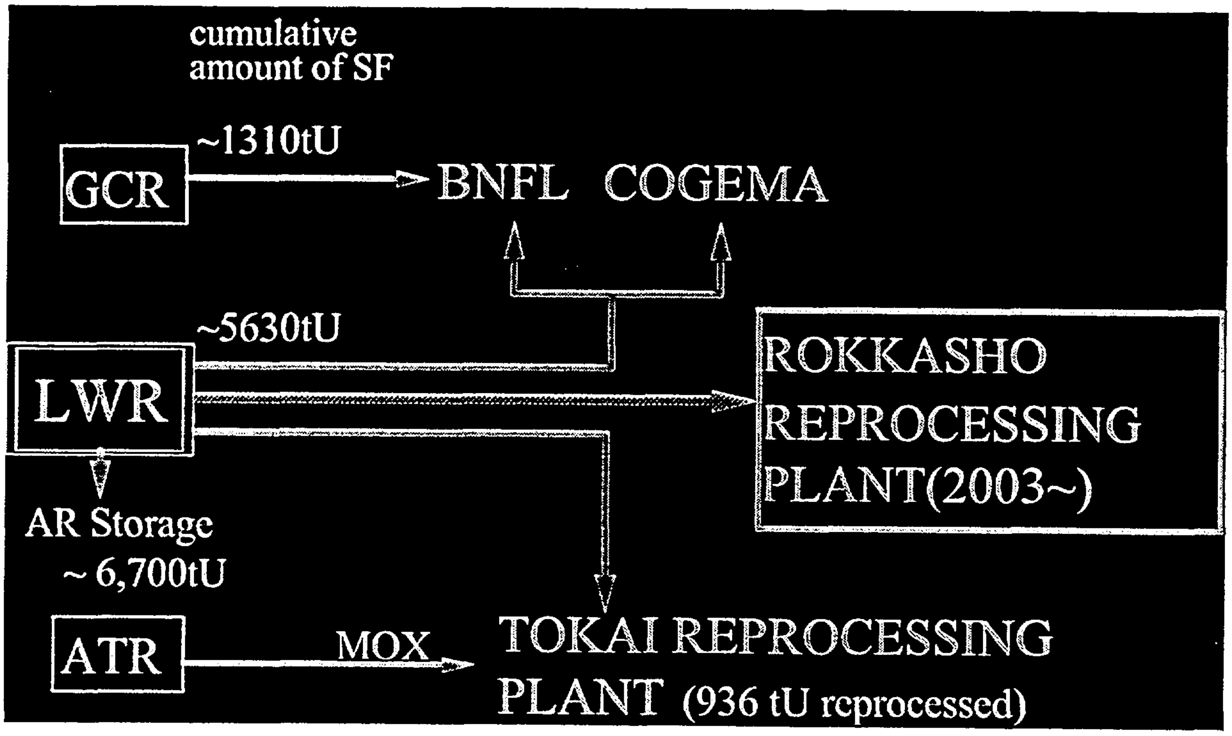


*FIG. 2. Flow of spent fuel (as of March 1998)*

Reprocessing service will be provided by the Tokai and the Rokkasho reprocessing plants and reprocessing has been contracted to BNFL and COGEMA. The Tokai reprocessing plant, which is operated by Japan Nuclear Cycle Development Institute (JNC) reorganized from PNC, has an annual reprocessing capacity of 100 tU. The Tokai reprocessing plant has reprocessed about 936 tU as of end of September 1998, since it started hot test in 1977 (Fig. 4). The Tokai reprocessing plant will shift its major role to research and development of future reprocessing technology.

The Rokkasho reprocessing plant, which is being constructed by Japan Nuclear Fuel Ltd. (JNFL), will be Japan's first commercial reprocessing plant. The plant will have an annual reprocessing capacity of 800 tU and is scheduled to start operation in January 2003. On 2nd October 1998, spent fuel for adjustment tests of the burnup monitor were received in the spent fuel storage pool of the Rokkasho reprocessing plant. The reprocessing capacity and technology of the second commercial reprocessing plant will be decided around 2010.

## 4. SPENT FUEL STORAGE IN JAPAN

Before reprocessing and recycling all the spent fuel arisings, spent fuel will be stored as an energy resource stockpile until the time it can be reprocessed, because the reprocessing capacity of Japan is now small compared with the amount of spent fuel arisings.

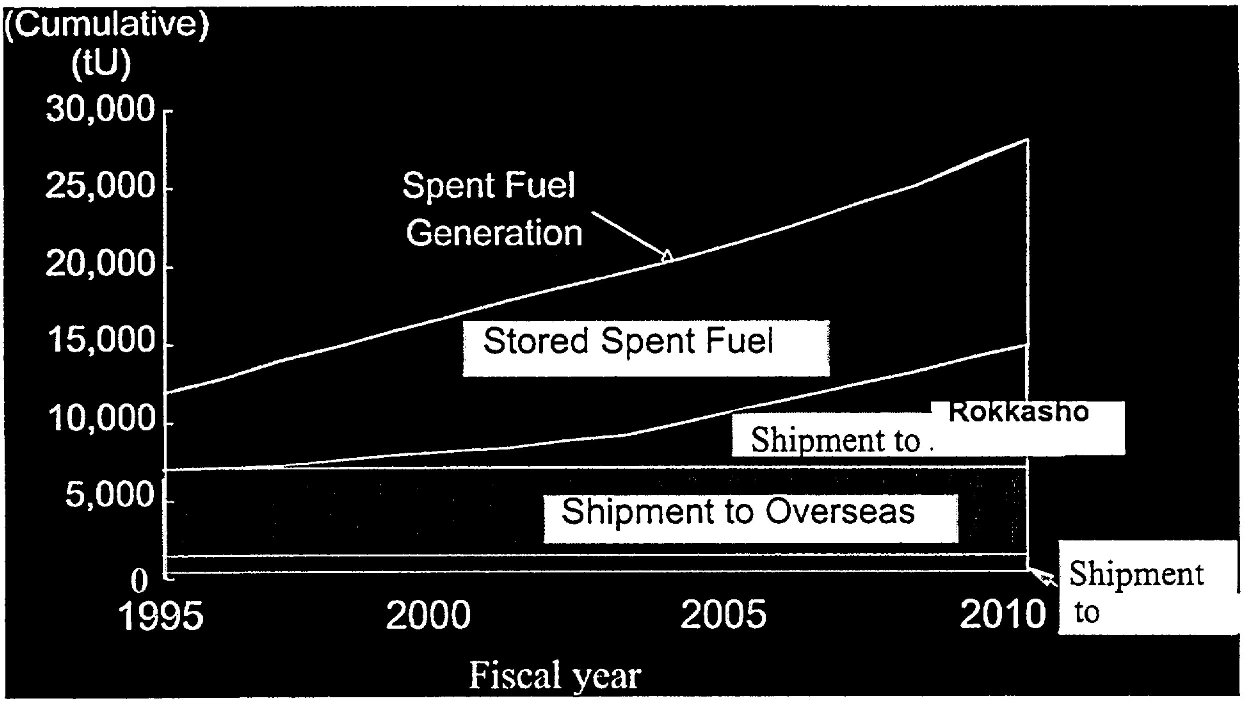


*FIG. 3. Prospect of LWR spent fuel management*

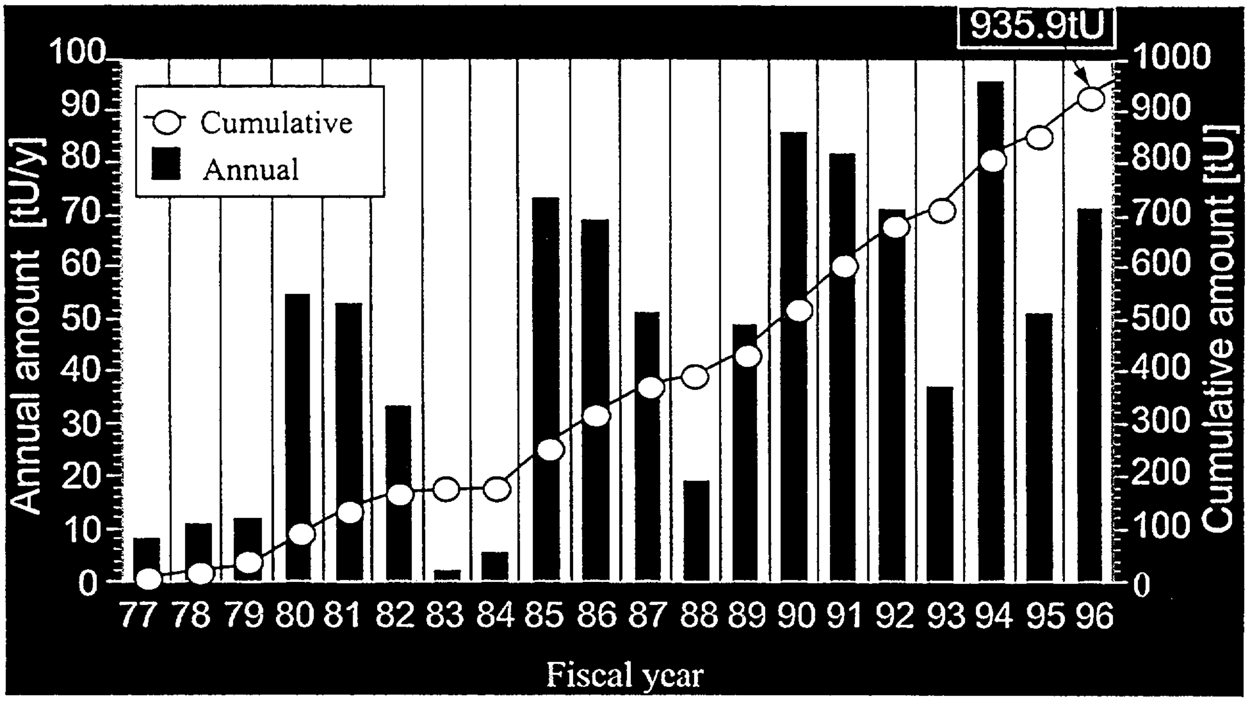


*FIG. 4. Tokai reprocessing plant - operating history*

As of the end of March 1998, a total of about 6,700 tU of spent fuel is being stored at each nuclear power station site. Total spent fuel storage capacity of Japanese commercial nuclear power stations is about 18,000 tU as seen in Table I. Some nuclear power stations intend to expand their capacity of spent fuel storage facilities.

TABLE I. STATUS OF SPENT FUEL STORAGE AT REACTOR SITES

(as of March 1998) (unit : tU)

| Utilities | Power Station | Stored Spent Fuel at-Reactor Site | Spent Fuel Storage Capacity | | |
|---|---|---|---|---|---|
| | | | Current | Under Construction | Total |
| Hokkaido Electric Power Co. | Tomari | 190 | 600 | 0 | 600 |
| Tohoku Electric Power Co. | Onagawa | 130 | 600 | 0 | 600 |
| Tokyo Electric Power Co. | Fukushima-Daiichi | 810 | 2,700 | 100 | 2,800 |
| | Fukushima-Daini | 1,030 | 1,900 | 0 | 1,900 |
| | KashiwazakiKariwa | 940 | 2,800 | 1,000 | 3,900 |
| Chubu Electric Power Co. | Hamaoka | 510 | 1,400 | 0 | 1,400 |
| Hokuriku Electric Power Co. | Shika | 30 | 200 | 0 | 200 |
| Kansai Electric Power Co. | Mihama | 190 | 500 | 0 | 500 |
| | Takahama | 650 | 1,500 | 0 | 1,500 |
| | Ohi | 440 | 1,200 | 0 | 1,200 |
| Chugoku Electric Power Co. | Shimane | 230 | 600 | 0 | 600 |
| Shikoku Electric Power Co. | Ikata | 240 | 700 | 0 | 700 |
| Kyusyu Electric Power Co. | Genkai | 240 | 1,500 | 0 | 1,500 |
| | Sendai | 490 | 800 | 0 | 800 |
| Japan Atomic Power Company* | Tsuruga | 360 | 600 | 0 | 600 |
| | Tokai-Daini | 200 | 400 | 0 | 400 |
| | Tokai | 30 | 200 | 0 | 200 |
| Total ** | | 6,700 | 18,000 | 1,100 | 19,100 |

* JAPCO's Tokai power station (GCR) has been shut down since 31 March 1998. About 190 tU of spent fuels were being generated and remained in its reactor core in addition to the figures of this Table.

** Rounded values.

In view of the increasing amount of spent fuel stored in the long term, an advisory committee was established by the Ministry of International Trade and Industry (MITI), the Science and Technology Agency (STA) and the electric power companies. It studied the introduction of off site storage of spent fuel and issued a report in March 1998, which can be summarized as follows;

- An off site spent fuel storage facility (a recycling fuel resource storage facility) is necessary to commence operation by around 2010. The needed capacity of such facility will be 6,000 tU by 2010 and 15,000 tU by 2020, see Tables II and III.
- The safety and the technologies of spent fuel storage were already established (e.g. through experiments in heat removal test rigs , see Fig. 5.
- In general, such facility has no restriction to geographical location.
- The operator of such facility is not necessarily limited to the electric power companies.
- Newly add the terms on Recycling Fuel Resource Storage Undertaking in the Law for the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors, and apply the established licensing procedure consisting of the government organization's examination followed by the examination by Nuclear Safety Committee (NSC) and AEC.
- To materialize, the government organizations will carry out institutional arrangement and the electric power companies will conduct location set-up and public acceptance.

| | Cask type | Vault type |
|---|---|---|
| Reduced scale | 1/5 | 1/5 |
| Dimension of a heater | 0.5m ϕX1mH | 0.1m ϕX1.3mH |
| Number of heater | 32 | 39 |

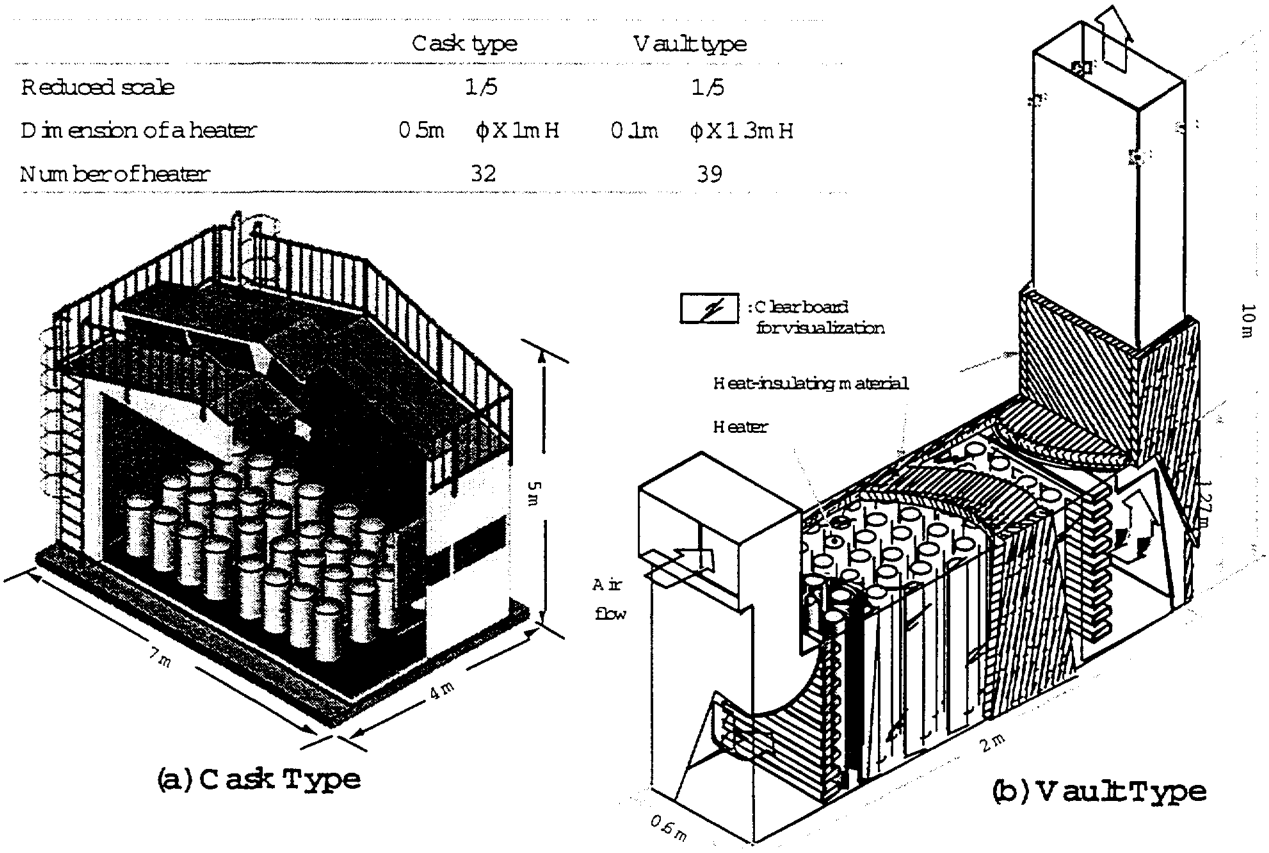


*FIG. 5. Heat removal test rigs*

TABLE II. PROSPECT FOR PRODUCED AMOUNT OF SPENT FUEL IN JAPAN

(ton U)

| | At Present | 2010 | 2030 |
|---|---|---|---|
| Annual Production of Spent Fuel | 900 | 1,400 | 1,900 |

TABLE III. AMOUNT NEEDED FOR SPENT FUEL INTERIM STORAGE

(ton U)

| Item | 1997-2010 | 2011-2020 | 2021-2030 |
|---|---|---|---|
| Produced Spent Fuel (a) | 15,200 | 16,000 | 19,100 |
| Dispatched to Rokkasho (b) | 7,600 | 8,000 | 8,000 |
| Dispatched Oversea (c) | 70 | - | - |
| Stored at Power Plant Site (d) | 5,200 | 4,200 | 2,500 |
| Needed Capacity for Interim Storage (a-b-c-d) | 2,300 | 3,800 | 8,600 |
| Needed Capacity for Accumulated Interim Storage | 2,300 | 6,000 | 14,600 |

Standing on this report, MITI's Subcommittee on Nuclear Energy under Advisory Committee for Energy studied the institutional arrangement for introduction of recycling fuel resource storage facility and published the report to recommend construction of interim storage facility for spent fuel as future recycling fuel resources. Government organizations are now preparing the necessary institutional arrangements for introducing such facilities.

## 5. RECYCLING PLUTONIUM AND URANIUM RECOVERED FROM SPENT FUEL

Since Japan promotes the policy of reprocessing all spent fuel, recovered plutonium must be utilized in a suitable programme in the spirit of the non proliferation policy. For a certain period of time, LWRs will continue to play a major role in Japan's nuclear power generation programme and some of them will use the recovered plutonium. FBRs will play a central role in the future nuclear fuel recycling system and will be the principal reactors to use the recovered plutonium in combination with LWRs.

Utilization of MOX fuel in LWR is important from the view point of utilizing the recovered plutonium before commercialization of FBRs. Corresponding to the Cabinet consent, the Federation of Electric Power Companies of Japan announced a programme for the MOX utilization in LWRs in February 1997 as shown in Table IV. It is necessary to construct a domestic commercial MOX fabrication plant for LWRs, in consideration of the operation plan of Rokkasho reprocessing plant. The domestic commercial LWR MOX fuel fabrication plant will have a capacity of around 100 t of MOX fuel per year.

TABLE IV. PROJECTION OF PLUTONIUM UTILIZATION IN LWRs

| | **1999** | **2000** | **> 2000** | **< 2010** |
|---|---|---|---|---|
| Tokyo Electric Power Co. | | | 1 | 0 - 1 |
| Fukushima 1-3 | 1 | | | |
| Kasiwazaki Kariwa 3 | | 1 | | |
| Kansai Electric Power Co. | | | | 1 - 2 |
| Takahama 4 | 1 | | | |
| Takahama 3 | | 1 | | |
| Chubu Electric Power Co. | | | 1 | |
| Kyusyu Electric Power Co. | | | 1 | |
| Japan Atomic Power Company | | | 2 | |
| Hokkaido Electric Power Co. | | | | 1 |
| Tohoku Electric Power Co. | | | | 1 |
| Hokuriku Electric Power Co. | | | | 1 |
| Chugoku Electric Power Co. | | | | 1 |
| Shikoku Electric Power Co. | | | | 1 |
| Electric Power Development Co. | | | | 1* |
| Total | 4 | | 5 | 7 - 9 |
| Total cumulative | | | | 16 - 18 |

* EPDC's Oma Power Station (eventual start in 2007) has announced the projection of plutonium utilization before the date of this table make-up

The prototype FBR MONJU, which is a loop type LMFBR of 280 MWe output, was first critical in April 1994 and started the generation of electricity in August 1995. Because of the sodium leak incident occurred in December 1995, the construction of MONJU was stopped and a total safety re-evaluation of the plant was performed in order to improve its safety.

As the Cabinet consent was shown in February 1997, further strategies for development of fast breeder reactors were discussed in the Special Committee on FBRs established under the AEC. The committee recommended the promotion of development of fast breeder reactors as one of the promising non-fossil energy sources for the future under a flexible programme while pursuing safety and economy.

Recovered uranium can be converted to uranium hexa-fluoride, followed by re-enrichment and re-conversion. Otherwise the recovered uranium can be mixed with enriched uranium, or can be mixed with plutonium to be recycled as MOX fuel. Re-enrichment process is considered to be the best method of recycling uranium from the economy and the amount of usable recovered uranium point of view. About 240 t of recovered uranium will be converted to uranium hexa-fluoride by JNC under a contract with Japanese electric power companies.

For the future nuclear fuel recycling system, it is important not only to strive for improvement of safety, reliability and economy but also to pursue the possibilities of reduction of environmental impact and the assurance of nuclear non-proliferation. Research and development will be conducted in the long term on advanced nuclear fuel recycling technology based on FBR, such as recycling of new types of fuel and recycling plutonium together with actinide elements. R&D programmes on the advanced nuclear fuel recycling technology are being discussed in the AEC's Advisory Committee on Nuclear Fuel Recycling Programme.

## 6. CONCLUSIONS

Japan intends to guarantee its future energy security by steadily carrying forward research and development efforts aimed at future commercial commissioning of nuclear fuel recycling, involving the reprocessing of spent fuel and the recovery of plutonium and uranium to allow the reuse of these materials as nuclear fuel in LWRs and FBRs.

Concerning spent fuel management, the policy measures include the expansion of storage capacity at reactor sites and the study on the storage option for the facilities at away-from-reactor sites in addition to the facilities at reactor sites.

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# THE NEW TECHNOLOGY OF INTERIM STORAGE OF FAILED SPENT FUEL FROM THE BN-350 FAST REACTOR

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**Abstract**

At present, intact spent fuel of the BN-350 reactor is stored in baskets of an open type. For storage of damaged and failed spent fuel, a cylindrical canister type T-1 made of austenitic stainless steel is used. The encapsulating procedure into such canisters is carried out in the water pool after discharge of such a fuel assembly from the reactor. The storage of such assemblies takes place with water inside the canister, which is separated from water of the pool by a hermetic lid. The transition to the "dry-wet" storage of failed spent fuel of the BN-350 fast reactor is considered an actual technical task. This paper describes possible practicable approaches for solving spent fuel storage problems taking into account utilization of existing standard and experimental equipment of the hot cell and water pool of the BN-350 reactor.

## 1. INTRODUCTION

Among the different spent fuel storage technologies, the at-reactor storage is the most practized technology. This technology is also put into practice at the BN-350 fast liquid metal reactor.

At present, intact spent fuel of the BN-350 reactor is stored in open type baskets. For storage of damaged and failed spent fuel, a cylindrical canister type T-1 made of austenitic stainless steel is used. The encapsulating procedure into such canisters is carried out in water pools after discharging the damaged fuel assembly from the reactor. The assemblies are stored of with water inside the canister, which is separated from the pool water by a hermetic lid. This lid seals the canister with the help of a rubber gasket. According operation experience, the life time for the gasket is not more than 15 years and this is obviously the time limit for safe storage of failed spent fuel using this technology. Further storage of failed assemblies having gas leakage and fuel dispersion would lead to contamination of the water pool by radioactive nuclides (e.g. fission products and plutonium).

## 2. DESCRIPTION OF FAILED FUEL

According to measurements of the failed cladding detection system, part of the fuel assemblies were classified as failed assemblies with gas leakage or fuel dispersion. All such fuel assemblies were sealed into canisters type T-1. The condition of failed fuel was investigated using the fuel assembly C-19 as an example.

The measurements showed alpha-activity in all water samples inside the canister. The analysis of alpha spectres revealed that more than 80% of alpha activity was caused by plutonium-239 and, in addition, there were alpha particles in the spectrum with an energy of more than 7 MeV.

The results of disassembling showed that the cladding of the pins had brittle damage because of the full exhaustion of the strength and plasticity of SS 16%Cr-15%Ni-3%Mo-Nb (in cold worked condition) over time. The fuel assembly showed corrosion damage of the duct and a large amount of precipitation of brown colour on the cladding due to the conditions in the storage pool.

Long-term storage of such fuel assemblies is unacceptable in water pools, because it can lead to the complete destruction of ducts and pins and considerably hampers work with fuel during its encapsulation and transition to the "dry" storage.

The fuel assemblies of the first and second type with a large bending (25 mm and more) are another type of failed fuel. Potentially it is possible to damage such fuel assemblies by handling during discharging and inserting, but also by hitting them accidentally when working nearby. The fuel assemblies overhang from the baskets is about 1 m and is not protected by the basket construction. Such fuel assemblies have to be subjected to geometry correction and subsequent encapsulation.

The third type of failed fuel are pins from fuel assemblies which were disassembled in the hot cell. Mostly they are stored in open canisters under water. The cladding of some of such pins is made of ferritic-martensitic steel 13%Cr-2%Mo-Nb-V-B, which is very corrosive in water.

So, there is a task to exclude degradation of failed fuel in water and to ensure a reliable and safe localization of the fission products. That is why it is necessary to transfer failed fuel to a different form of storage, where direct contact of the fuel assembly with water is excluded. Thus, it is a transition from a "wet" to a "dry-wet" and "dry" form of storage. Such transition is possible after approximately 5 years storage of the fuel assembly under water; the decay heat in this case is not more than 100 W and 12 W respectively, for a driver and a blanket fuel assembly.

## 3. DESCRIPTION OF THE "DRY-WET" STORAGE

Today, it is more effective to utilize a "dry-wet" storage for damaged fuel assemblies, by which the following is understood:

- the positioning of a fuel assembly in an inert atmosphere of a sealed stainless steel canister;
- the placing of a sealed canister under water into a basket with 19 seats.

The transition to the "dry-wet" storage is considered to be a preparatory period for transition of spent fuel to "dry" long-term storage. The "dry-wet" storage allows:

- to halt the degradation of spent fuel by corrosion of structural materials in water;
- to secure a safe working environment for handling and transportation of spent fuel;
- to guarantee safe storage of spent fuel in existing water pools during at least 50 years under high quality water condition.

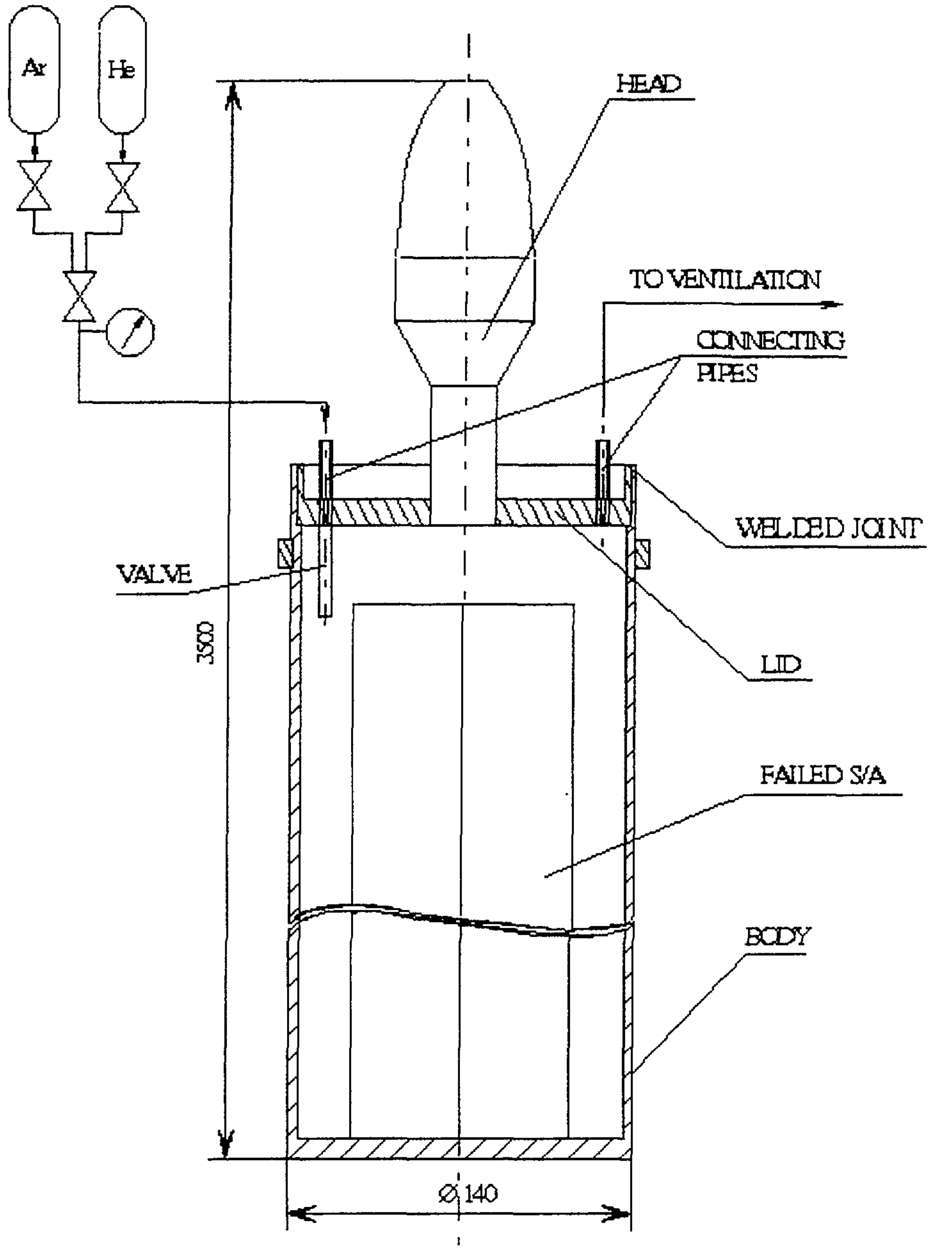
In light of this, the transition to the "dry-wet" storage of failed spent fuel of the BN-350 fast reactor is considered an actual technical task. This paper describes possible practical approaches for solving spent fuel storage problems taking into account real possibilities of existing standard and experimental equipment of the hot cell and water pools of the BN-350 reactor.

## 4. THE TECHNOLOGY DESCRIPTION

As a main approach in spent fuel encapsulation, a method of individual encapsulation for every fuel assembly is chosen. This approach takes the following into account:

- the duct and cladding material is brittle after irradiation;
- the equipment at the BN-350 reactor can handle only single fuel assemblies;
- equipment and remote technology for welding stainless steel canisters up to 150 mm in diameter is available;
- the technology and baskets for storage of such canisters is available;
- the encapsulated spent fuel can be safe handled under water;
- contamination of neighbour fuel assemblies, in case of destruction of a fuel assembly inside a canister, is excluded.

The canister (see Fig. 1) is a stainless steel tube 140x3 mm in diameter with a bottom and a lid made of steel 18%Cr-10%Ni-1%Ti. The lid is provided with a head, for handling the canister, and two small connecting pipes for filling the canister with argon. After welding the lid and filling the canister with dry argon and small quantity of helium, the unit is transported by the head using a standard grapple.




S/A stands for fuel assembly (subassembly)

*FIG. 1. Canister and its filling with Ar and He*

The technology of spent fuel encapsulation consists of:

- correction of a fuel assembly's geometry in the hot cell;
- drying of a fuel assembly by blowing through it hot air;
- placement of a fuel assembly into the canister and placement of a lid;
- transportation of a canister with a fuel assembly inside to the welding place;
- welding of a lid;
- controlled filling of the canister with argon and helium, and welding of connecting pipes:
- check-up of the tightness of the welding;
- a registration of the canister and transition to the "dry-wet" storage in water pools.

The encapsulation of spent fuel should start with fuel assemblies which are in an emergency state or are in unsatisfactory condition, such as:

- failed fuel and gas leakage;
- severe bending after irradiation;
- dismantled in the hot cell;
- with ducts having high pitting corrosion.

The geometry correction (Fig. 2) is an operation of cutting off some parts of the fuel assemblies with the goal of placement into the canister. For fuel assemblies with ducts made of ferritic-martensitic steel with small changes of the shape, only the heads and tails are being cut off. The length of the canister is in this case 3,000 mm. For fuel assemblies of the first and the second type with large bends, also a cut of the upper axial blanket will be done. The minimal canister length, which is obtained by cutting off heads and tails, accommodates the need for minimal dimensions for storage in casks and at dry storage facilities.

Drying of the fuel assemblies is necessary to avoid that water will get into the canister and will be done by blowing hot air through the fuel bundle in the hot cell. The scheme of the drying process is shown in the Fig. 3.

The most important operation during encapsulation is the welding of the upper lid to the body of the canister. This operation is performed by using an argon arc welding and remotely operated unit.

After checking the quality of the welding, the sealed canister is put into a 19-seater basket for "dry-wet" storage.

## 5. CONCLUSION

The approach expounded here is mainly intended for damaged spent fuel, but if needed, it can be extended to all spent fuel of the BN-350 reactor.

As in case of "wet" storage, the "dry-wet" one demands a continuation of a maintenance of water quality in pools according to standards:

- pH = 5.5-8.0;
- chlorides concentration [Cl] < 0.3 mg/l;
- electrical conductivity $\sigma$ = 8 mg/l.

It is necessary to note that the equilibrium concentrations of oxygen and hydrogen peroxide in the water pools of the BN-350 are equal to:

$[O_2]$ = 5 mg/l; $[H_2O_2]$ = 8 mg/l.

The high concentration of oxidants in the water led to the introduction of a strict restriction of the chlorides concentration. This restriction, will help to slow down significantly corrosion cracking of ducts, cladding and canisters made of stainless steels. For suppression of total and pitting corrosion of ducts made of ferritic-martensitic steel (13%Cr-2Mo-Nb-V-B and 12%Cr-2%Ni-Mo), a correction of the water quality in the pools has been suggested. The hydrogen coefficient has shifted to the alkaline side and is equal to 8.0. In case all spent fuel with ferritic-martensitic ducts will be converted to "dry-wet" storage, such correction is not needed.

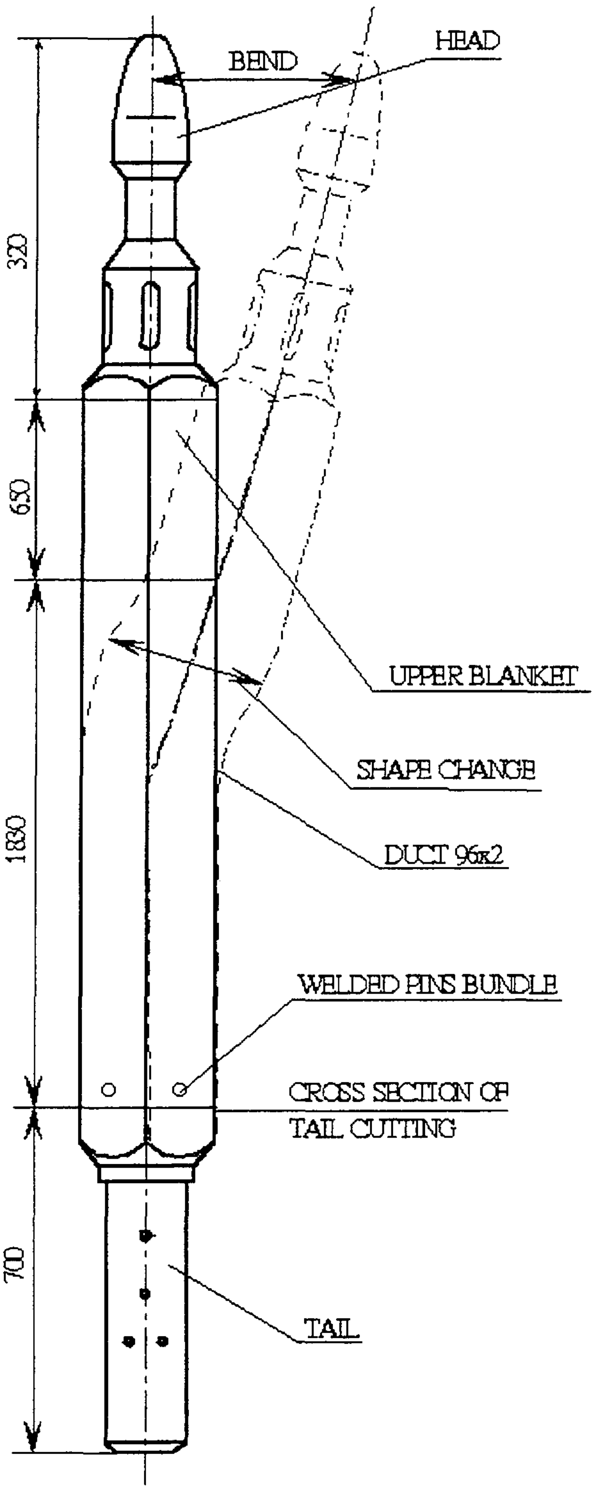
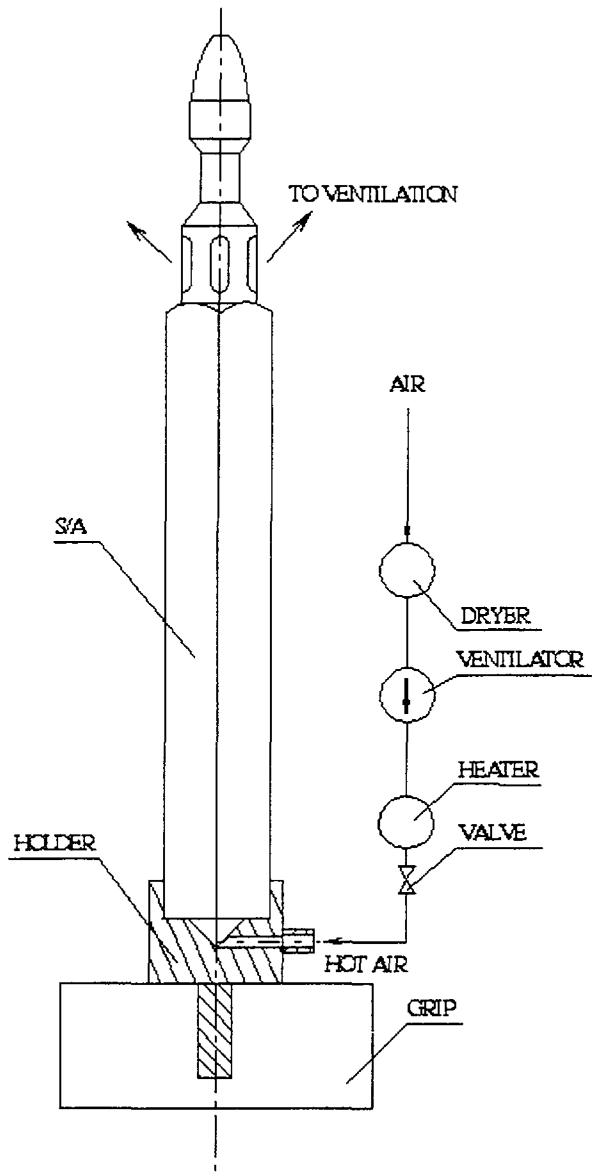



*FIG. 2. Geometry correction*




S/A stands for fuel assembly (subassembly)

*FIG. 3. Drying system*

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# CURRENT STATUS OF SPENT FUEL MANAGEMENT IN THE REPUBLIC OF KOREA



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**Abstract**

Due to the lack of indigenous energy sources in Korea, the government selected nuclear energy as one of the major sources of electricity generation. According to the Korean government programme of a nuclear power development, currently, 14 nuclear power plants (NPPs) are in operation and 4 NPPs are under construction. In addition, further 10 NPPs are planned to be in operation by the year 2015. The large amount of spent fuel discharged from the nuclear power plants is accumulated in at-reactor (AR) storage pools. Due to the limited capacity of these AR storage pools, the safe and economic management of spent fuel is to be resolved. The spent fuel management strategy in Korea, basically depends on the interim storage in wet and dry storage facilities, including expansion of storage capacity. This paper describes the current status and plans of the spent fuel management in Korea.

## 1. INTRODUCTION

Since the commercial operation of Kori unit-1 in April 1978, Korea has achieved a rapid growth in nuclear power and has now 14 NPPs, of which 11 pressurized water reactors (PWRs) and 3 pressurized heavy water reactors (CANDUs). At of the end of August 1998, the installed nuclear capacity was about 12 GWe representing 27.5 % of the total installed electrical capacity and 34.4 % of the total electricity production. There are 4 NPPs (3 PWRs and 1 CANDU) under construction.

According to the fourth long-term electricity development plan, recently updated by the Government (i.e. on 27 August 1998), there are plans for another 10 NPPs (about 11 GWe) to be in commercial operation by 2015, which will bring the total to 28 NPPs (see Fig. 1). It is expected that they will account for 34.2 % of the country's total installed electrical capacity and provide 46.3 % of the total electricity production [1].

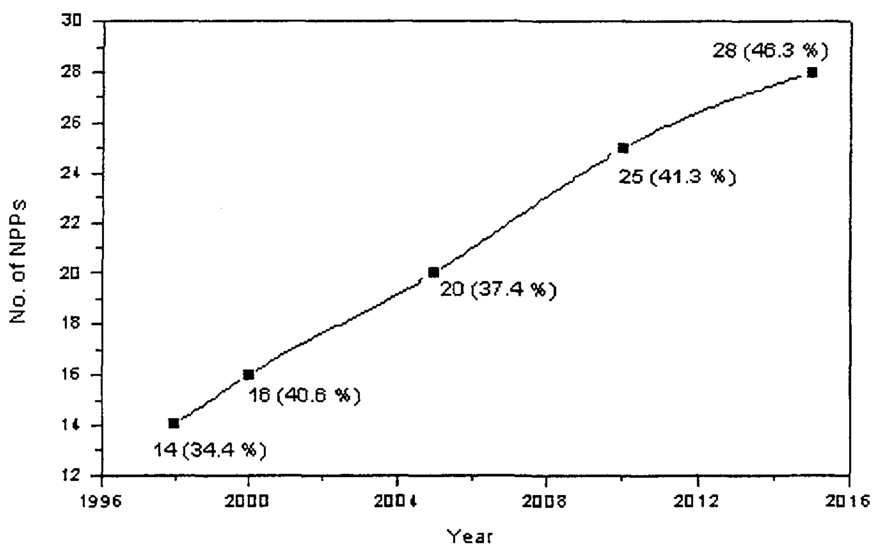


*FIG. 1. Nuclear power development plan*

This nuclear power programme entails the management of spent fuel discharged from the reactors. By the end of August 1998, the total amount of spent fuel discharged form the NPPs is about 3,397 tU and so far they have been stored at-reactor sites. It is estimated, that the onsite storage will be saturated around 2006~2007, judging from current storage capacities and annual discharge of spent fuel. Therefore, a variety of measures has been implemented to expand the storage capacity by reracking with high density rack, the installation of a dry storage facility, changing to longer reload fuel cycles and the transhipment among neighbouring units.

## 2. SPENT FUEL MANAGEMENT POLICY

The Atomic Energy Act (AEA) specifies the legal and technical aspects related to the spent fuel management such as the storage, treatment and disposal of spent fuel. In pursuance of the Act, the spent fuel management policy was reviewed and decided by the Korean Atomic Energy Commission (AEC). In July 1988, the AEC has set forth a resolution that an away-from-reactor (AFR) storage facility be built by the end of 1997 as an Interim Storage Facility (ISF), to accommodate the mid-term spent fuel arisings. According to the spent fuel management plan, recently updated and currently reviewed by the AEC, construction of the ISF should start in 2008 and it should be completed in 2016. Up to the year of 2016, the spent fuel discharged from the NPPs should be stored in using one of the measures mentioned earlier.

In December 1994, it was announced that a small island, Gurop, off the west coast of the Kyunggi province, had been selected as a suitable site for an interim spent fuel storage facility. However, a year later the plan was cancelled after geological studies by the site investigation team found active faults in the seabed.

## 3. SPENT FUEL ARISING AND STORAGE

The amount of spent fuel discharged per year from the nuclear power reactors differs from the type, size and fuel loading pattern used. The 2-loops PWRs of such as Kori units 1 and 2 discharge approximately 15 to 17 tU of spent fuel per year, while the 3-loops PWRs (the remaining 9 PWRs) produce approximately 19 to 22 tU per year. Approximately 95 tU of spent fuel is discharged per year from the Wolsong units 1-3 (CANDUs), each. At the end of August 1998, a total amount of 3,397 tU spent fuel was discharged from the 14 operating reactors (11 PWRs and 3 CANDUs) and is stored at the reactor sites. Table I shows the status of the reactor site storage of spent fuel. It is anticipated, that the reactor site storage will be full around 2006, taking into account the annual spent fuel arising together with the storage capacities.

The annual spent fuel arising from the 16 units, which are expected to be in operation by 2000, will be about 550 tU/year. After the year of 2015, when a total of 28 nuclear units will be in operation, the annual discharge rate of spent fuel will about 940 tU/year. The projected cumulative amount of spent fuel will reach about 15,000 tU by the year 2015, and 19,000 tU by the year 2020 (see Fig. 2).

TABLE I. STATUS OF SPENT FUEL ARISING AND STORAGE IN KOREA

(unit : tU)

| Site | Reactor Type | No. of Units | Storage Capacity | Annual Discharge | Accumulated Spent Fuel | Storage Saturation Year (estimated) |
|---|---|---|---|---|---|---|
| Kori | PWR | 4 | 1,533 | 65 | 1,003 | 2006 |
| Yonggwang | PWR | 4 | 1,271 | 80 | 550 | 2006 |
| Ulchin | PWR | 3 | 1,059 | 57 | 394 | 2007 |
| Wolsong | CANDU | 3 | 4,576 | 294 | 1,450 | 2006 |
| Total | | 14 | 8,439 | 496 | 3,397 | |

Note: as of the end of August 1998.

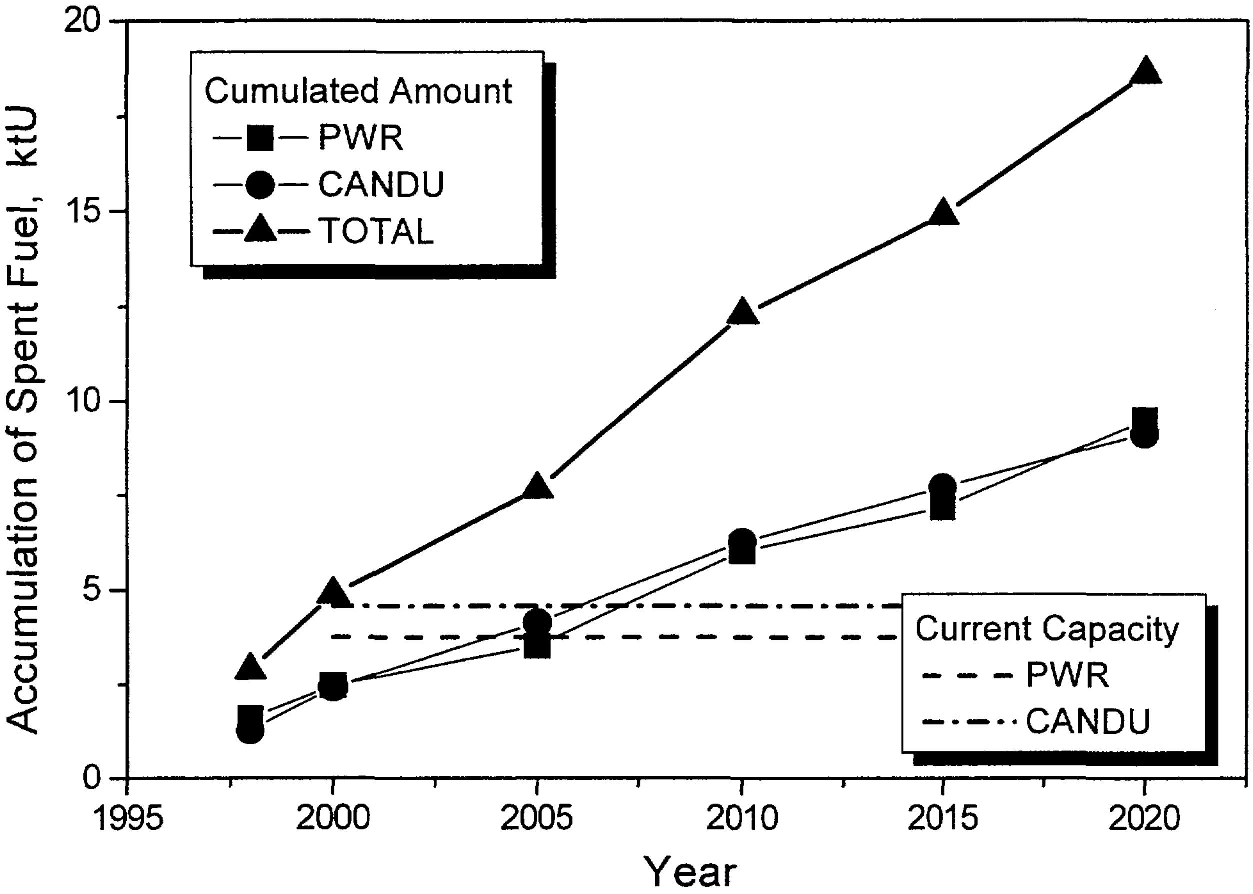


*FIG. 2. Projection of cumulative spent fuel arisigs*

4. EXTENSION OF AR STORAGE CAPACITY

In 1988, the AEC announced proposal for building a pool-type interim storage facility for spent fuel by the end of 1997. However, a suitable site to build the facility has not been selected because of strong local opposition even though a lot of efforts have been paid to look for the site. Therefore, no progress has been made on this project at all. Nevertheless, the spent fuel management strategy based on the interim storage has been unchanged.

The present strategy for spent fuel management in Korea is to store it safely at-reactor sites for the time being, and transported to the away-from-reactor (AFR) interim storage facility and stored for several decades.

Current storage capacities at reactor sites are not enough to store spent fuel to be discharged during the whole lifetime. Therefore, a wide range of measures for the storage capacity extension has been and being implemented. The measures are transhipment among neighbouring units, addition of or re-racking with high-density racks and the installation of dry storage silos (see Table II). In addition, longer fuel cycles, which results in reduction of the annual discharge of spent fuel, were implemented in PWR reactors.

**4.1. Transhipment among neighbouring PWRs**

Transhipment of PWR spent fuel among neighbouring units has been performed at the Kori site. A total of 312 spent fuel assemblies has been transferred from unit 1 to units 3 & 4. Two KSC-4 casks, which were developed by Korea Atomic Energy Research Institute (KAERI), had been

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employed for this transhipment. The B(U) type cask weighs 37 tons and is capable of loading four PWR spent fuel assemblies with a burnup of 38 GW·d/tU and a cooling time of 3 years. It can be used in both dry and wet conditions.

TABLE II. MEASURE FOR EXPANSION OF AR STORAGE CAPACITY OF SPENT FUEL

| Site | Programme |
|---|---|
| Kori | • Addition of high density racks into pools of units 3&4.<br>• Transhipment from unit 1 to units 3 & 4.<br>• Installation of dry storage facility (for 400 fuel assemblies).<br>• Full re-racking with high density racks in pools of units 3&4 |
| Yonggwang | • Addition of high density racks into pools of units 1 & 2 (for 400 fuel assemblies each).<br>• Full re-racking with high density racks unit 3&4 pools |
| Ulchin | • Full re-racking in unit 1&2 pools.<br>• Installation of dry storage facility |
| Wolsong | • Addition of dry storage silos (80 silos for 43,200 fuel bundles). |

### 4.2. PWR high density racks

The design basis of spent fuel racks was 35.56 cm (or 14 in.) in center-to-center distance when the reactors were constructed. It was thus intended to reduce the center-to-center distance as much as possible by adopting neutron poisonous materials. As a general rule for PWR, high-density racks feature the so-called discrete zone two region (DZTR) storage arrangement to reduce the center-to-center distance. Region 1 racks are used to store 5 wt% enriched-uranium fuel while maintaining $k_{eff}$ less than 0.95 without any burnup credit. On the other hand, region 2 racks have an enrichment/burnup credit on it. For the DZTR storage scheme, neutron absorbing materials such as boron SS, boraflex or boral have been used. Thus, a center-to-center distance of spent fuel racks was designed with about 24.89 cm on region 1 racks, and about 23.11 to 24.89 cm on region 2 racks.

### 4.3. Wolfing (CANDU) Dry Storage

Wolsong spent fuel dry storage (SFDS) is a storage concept in which irradiated fuel bundles are put in stainless steel baskets and in turn stored in dry concrete silos. A silo can contain as much as nine fuel baskets with each basket having 60 fuel bundles. The silo is 6.5 m high with 3.1 m in outside diameter. Spent fuel is usually cooled down in the spent fuel bay for at least 6 years before it is stored in dry concrete silos.

60 silos with a capacity of 32,400 spent fuel bundles were constructed in 1992, and 80 more silos with a capacity of 43,200 bundles were constructed recently. In total 140 silos with a capacity of 75,600 bundles are employed to store CANDU spent fuel in dry condition.

### 4.4. Longer fuel cycle

As an effort to increase the burnup and to reduce the annual discharge rate of spent fuel, the 15/18 months longer fuel cycle has been adopted for PWRs since 1987. Currently, all the PWRs in Korea adopted the 18 months fuel cycle. According to the operation experiences, the longer fuel cycle (18 months) operation for three loops PWRs reduced the annual discharge rate of spent fuel by 16 %.

## 5. RESEARCH AND DEVELOPMENT ACTIVITIES

Various research activities have been conducted to support the spent fuel management programme in Korea. The research activities are being carried out in following areas:

- Long-term dry storage of spent fuel;
- Transportation system;
- Burnup measurement of spent fuel assembly;
- Remote handling of spent fuel;
- Long-term storage behaviors of intentionally defected spent fuel in pool;
- DUPIC (direct use of spent PWR fuel in CANDU) fuel cycle.

The reactor mix strategy of PWR with CANDU in Korea allows a unique way for spent fuel management, i.e. the DUPIC fuel cycle to re-fabricate spent PWR spent fuel directly into CANDU-type DUPIC fuel without any separation process. [2]

## 6. CONCLUSIONS

Summing up the reviewed spent fuel management programme, the following conclusions can be drawn:

Various measures, such as re-racking of high density rack, installation of the dry storage facility, transhipment between neighbouring units and longer reload fuel cycles, have been implemented to extend the AR storage capacity of spent fuel. No significant problems have been encountered so far.

In the current situation, in which the best solution for long-term spent fuel management has not been established, many efforts for the extension of on-site storage capacity, such as the installation of on-site dry storage facilities and re-racking, etc., should be made to overcome the shortage of AR storage capacity until the completion of the ISF storage facility.

The related R&D activities should be continued for the technological support of the national spent fuel management strategies.

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# THE DRY SPENT RBMK FUEL CASK STORAGE SITE AT THE IGNALINA NPP IN LITHUANIA



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**Abstract**

At present, there are about 15,000 spent RBMK fuel assemblies stored in the water pools near the reactors at the Ignalina Nuclear Power Plant (INPP). Part of them are cut in two bundles and stored in standardized baskets in the pools. Each basket is loaded with 102 bundles. For long-term interim storage of this fuel, it was decided to use dry storage in casks. For this reason, the total activity to be stored is split into individual units (casks). Each cask represents a closed and independent safety system, fulfilling all safety-relevant requirements for both normal operational and hypothetical accidental conditions. The main safety relevant features of the storage cask system are:

(1) Inherent safety system;
(2) Double barrier system;
(3) Passive cooling by natural convection;
(4) Safety against accidents.

The cask dry storage system is a cost effective and multi-functional system for storage, transport after the operation time and final disposal under consideration of additional protective elements. From an economical point of view, cask storage has a number of advantages. Two cask types have been intended for the INPP storage site:

(1) The CASTOR RBMK cask made of ductile cast iron;
(2) The CONSTOR RBMK sandwich cask made of an inner and outer steel shell and reinforced heavy concrete.

The CASTOR RBMK and the CONSTOR RBMK casks are designed to withstand severe storage site accidents and – with help of impact limiters – to fulfil the IAEA test criteria for type B(U)F packages. The INPP spent RBMK fuel storage site is designed as an open air storage for an operational time of 50 years. The casks are arranged on the concrete storage pad. The site is equipped with a crane for cask handling and technological buildings and security systems. The safety analyses for fuel and cask handling and for cask handling and for cask technology at the site have been made and accepted by the Lithuanian Competent Authority.

## 1. BACKGROUND INFORMATION

The Ignalina nuclear power plant is situated in the north-east of Lithuania near the borders of Latvia and Belarus, on the shore of the Drukshiai lake, the largest lake in Lithuanian. The nearest cities to the plant are Vilnius at 130 km with over 600,000 inhabitants and Daugavpils in Latvia at 30 km with 126,000 inhabitants. The residence of the INPP's personnel is Visaginas, situated 6 km from the plant.

The INPP consists of two similar units of RBMK-1500 reactors. "RBMK" is a Russian acronym for "Channel-type Large Power Reactor" and is a water-cooled graphite-moderated channel-type power reactor. The RBMK-1500 reactor is the largest power reactor in the world and has a thermal power output of 4,800 MW and an electrical capacity of 1,500 MW. The first unit went into service at the end of 1983, the second unit in August 1987. Their design lifetime is projected to 2010 and 2015, respectively. A total of four units were originally planned on this site. Construction of the third unit was terminated in 1989.

The INPP has a direct cycle configuration - saturated steam, formed in the reactor by passing light water through the reactor core, is fed to the turbine at a pressure of 6.5 MPa. The light water circulates in a closed circuit. Each unit contains two K-750-65/3000 turbines with 800 MW generators. Each unit is provided with a fuel handling system and unit control room. The turbine room, waste gas purification and water conditioning rooms are common for the units.

## 2. HISTORICAL OVERVIEW OF SPENT FUEL MANAGEMENT

The problem of storing spent fuel at INPP arose several years ago. Spent fuel was originally intended to be shipped back to Russia for reprocessing and disposal. But, as a consequence of the disintegration of the former Soviet Union, this became impossible. Therefore, it was decided to build an interim spent nuclear fuel storage facility on the INPP site with a lifetime of about 50 years. It was supposed that the problem of spent nuclear fuel reprocessing and disposal will be resolved in this period.

Originally, a spent fuel storage facility was designed and proposed to the INPP by the All-Union Research and Development Institute of Power Engineering (Russian abbreviation - VNIPIEhT) in 1992. The design provided for two storage phases, i.e. a wet storage and dry storage, including an intermediate preparation of spent fuel for dry storage by encasing spent fuel in leak-tight casks to be filled with inert gas. The wet storage facility was supposed to be a separate on-site building measuring 220x60x40 m and equipped with all auxiliary systems required for its safe operation. It was a huge project which required a large material, financial and time expenditure.

Research on safe ways to store spent fuel done in countries operating nuclear power plants has shown that "dry" storage in inert environment is recognized as the safest long-term fuel storage after a certain period of wet storage. This method allows to store the fuel safely for 50 and more years. It can be done in two ways: the spent fuel can be stored in metal casks or in steel canisters placed inside concrete vaults.

In order to be able to make a decision which technology would be the best for INPP, an international tender was announced by the INPP and the Ministry of Energy of Lithuania. Nine companies which are world leaders in marketing of dry spent nuclear fuel facilities took part in the tender.

- Framatom/PNS, France/USA
- GEC Alsthom, UK
- GNB, Germany
- Ontario Hydro/AECL, Canada
- Siemens, Germany
- SGN, France
- VNIPIEhT, Russia
- Westinghouse, USA.

After the evaluation, the proposal of GNB was accepted to store spent nuclear fuel outdoors in sealed metal casks of the CASTOR type filled with inert gas. GNB's casks may be further licensed as transport casks for transportation of spent fuel outside Lithuania to a reprocessing facility or to a final repository.

An agreement was signed in December 1993, for the supply of 60 metal casks of the CASTOR type including technical documentation and handling equipment. Twenty CASTOR casks have been delivered up to now. In order to reduce the storage costs within the framework of the signed agreement, a decision was made to move to the CONSTOR type cask, which is made of a steel-concrete sandwich design.

In parallel with the active contract, the Lithuania Ministry of Energy has initiated a new tender for a dry spent fuel storage facility. Two Canadian companies: AECL and OH, and GNB from Germany were invited. In October 1996, the first round of the evaluation was performed. Two companies, GNB and AECL, were on the short list. AECL has proposed a concept of steel canisters loaded into concrete vaults, and GNB offered a steel-concrete sandwich cask of the CONSTOR type.

For some reasons, it was decided to conclude a contract with AECL. The licensing procedure for the CONSTOR type cask was started in January 1998. In October 1998, the Lithuanian Competent Authority VATESI has issued the permission for manufacturing of the CONSTOR cask. This type of cask has passed the licensing procedure in Russia as type B(U) package.

## 3. THE SPENT FUEL STORAGE AND HANDLING SYSTEM

The system for handling and storing the spent fuel assemblies is designed to perform the following main functions:

- cooling of the spent fuel assemblies (SFA);
- cutting of the fuel assemblies (FA) into fuel bundles (FB);
- placing them into transport baskets designed to accomodate 102 fuel bundles;
- storing of the loaded transport baskets in the pools;
- loading of the transport baskets with 102 spent fuel bundles in casks; and
- moving of the casks to the on-site spent fuel storage facility (SFSF) for long-term storage.

In addition, the system considers the treatment and disposal of process wastes generated by various fuel handling systems. The system also provides transport services for maintenance activities which involve the use of special technologies.

The main safety goals are to ensure radiation safety under normal operational conditions and to keep radiation exposure of the personnel, general public and environment in case of an accident within the limits set by radiation standards and health regulations. The major objectives of *radiation safety are:*

- prevention of occasional accidental criticality;
- provision of the required radiation protection;
- prevention of unacceptable radioactive releases.

The spent fuel storage and handling system is represented by several independent subsystems, each performing its own function in a certain sequence. The system includes the following subsystems:

- handling of spent fuel in the unit;
- storage of the spent fuel assemblies extracted from the reactor prior to cutting;
- cutting of the spent fuel assemblies into fuel bundles and placing them into the transport basket;
- storage of the loaded transport baskets with cut fuel assembly in the storage pools;
- storing of spent fuel bundles in the storage casks (CASTOR and/or CONSTOR)
- transporting of spent fuel outside the unit to a separate on-site storage facility.

The components of the spent fuel storage and handling system are located in the reactor building.

Spent fuel assemblies discharged from the reactor and spent fuel bundles in casks are stored in storage pools of which the ceilings come out in the storage pool hall (SPH). All process operations related to handling of the spent fuel are performed in the SPH.

After placement in the pool, the spent fuel assemblies remain for at least a year, after which *they may be removed for cutting. The cutting bay is located in the reactor compartment between the* storage pool hall and the reactor hall. The bay, which includes a hot cell (He), control room and maintenance area, is designed for:

- cutting of the spent fuel assembly into two fuel bundles;
- placing them into a transport baskets;
- cutting of the structural parts central rod, load bearing tube of the spent fuel assembly;
- containerizing the pieces and taking containers away for disposal.

After spent fuel bundles have been stored for a certain time, the loaded transport baskets with spent fuel are placed into the casks and are taken away for a long-term storage (up to 50 years) to the on-site spent fuel storage facility.

The storage time before loading into the "CASTOR" type metal container and "CONSTOR" type steel-concrete sandwich cask is at least 5 years after discharge from the reactor. The on-site spent fuel storage facility for spent fuel is planned to be constructed at the Ignalina site approximately 1 km away from the units. This will be a dry storage facility in which spent fuel will be stored in the CASTOR and CONSTOR casks.

At the initial stage, 20 metal "CASTOR RBMK" casks delivered by GNB (Germany) will be used for storing and transporting spent fuel. At a later stage, the steel-concrete sandwich CONSTOR RBMK cask (also made by GNB), will be used.

The treatment of process wastes generated in the course of fuel handling, consists in cutting of long pieces of the fuel assemblies (LPCF), containerizing the cutted pieces and transferring containers to the solid radwaste storage facility (SRWSF).

Transport and process areas are equipped with hoisting-and- conveying machinery such as cranes, electric overhead-track hoists, electric hoists and hand pulleys, self-propelled transport trolleys. The rooms are linked together by a system of transport and railway corridors which allow to transfer material and equipment via corresponding doors.

## 4. CASTOR AND CONSTOR STORAGE SYSTEMS

For long-term interim storage of the spent fuel bundles, it was decided to use dry storage in casks. For this reason, the total activity to be stored is split into individual units (casks). Each cask represents a closed and independent safety system, fulfilling all safety-relevant requirements for both normal operational and hypothetical accidental conditions.

The main safety relevant features of the storage cask system are

- Inherent safety system;
- Double barrier system;
- Passive cooling by natural convection.

The cask dry storage system is a multi-functional system for storage, transport after the operation time and final disposal under consideration of additional protective elements. From an economical point of view, the cask storage has the following advantages:

- High flexibility and easy expandability;
- Easily adaptable fabrication technology for local manufacturing conditions in case of the CONSTOR steel concrete sandwich cask;
- Low costs for operation, maintenance and decommissioning only.

Two cask types have been intended for the INPP storage site:

- The CASTOR RBMK cask (see Fig. 1) made of ductile cast iron and closed with a screwed double barrier lid system;
- The CONSTOR RBMK sandwich cask (see Fig. 2) made of inner and outer steel shells steel on and of reinforced heavy concrete. The lid system consists of a bolted lid and two welded lids.

The CASTOR RBMK and the CONSTOR RBMK casks are designed to withstand severe storage site accidents, such as:

- drop;
- earthquake;
- fire;
- gas cloud explosion;
- airplane crash.

With the help of impact limiters the casks fulfils the IAEA test criteria for type B(U)F transport packages.

The nuclear safety of both cask concepts has been analysed by internationally accepted codes [1]. The safety analyses results were confirmed by experimental programmes in which the mechanical and thermal cask behaviour under accident conditions has been tested [2, 3].

The CASTOR RBMK cask has been licensed for storage by the Lithuanian Competent Authority VATESI. The CONSTOR RBMK cask was certified as a type B(U)F package by GOSATOMNADZOR of Russia.

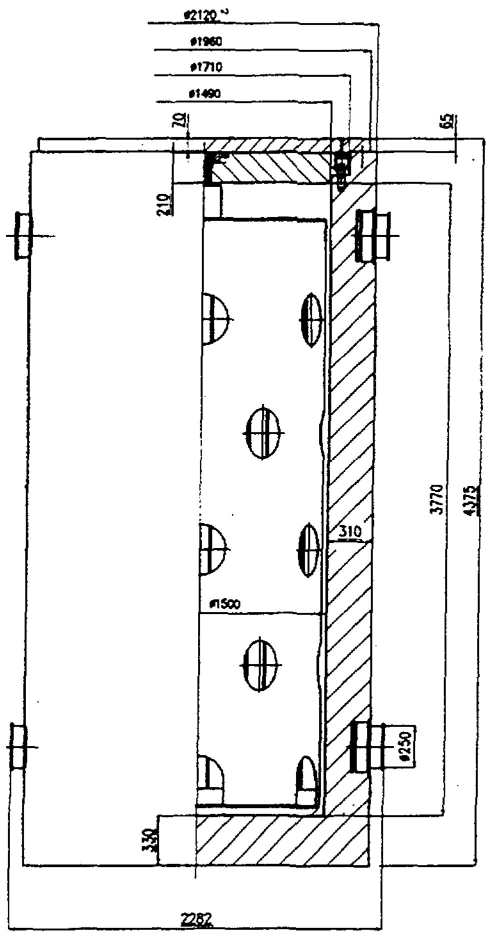


*FIG. 1. CASTOR RBMK storage cask*

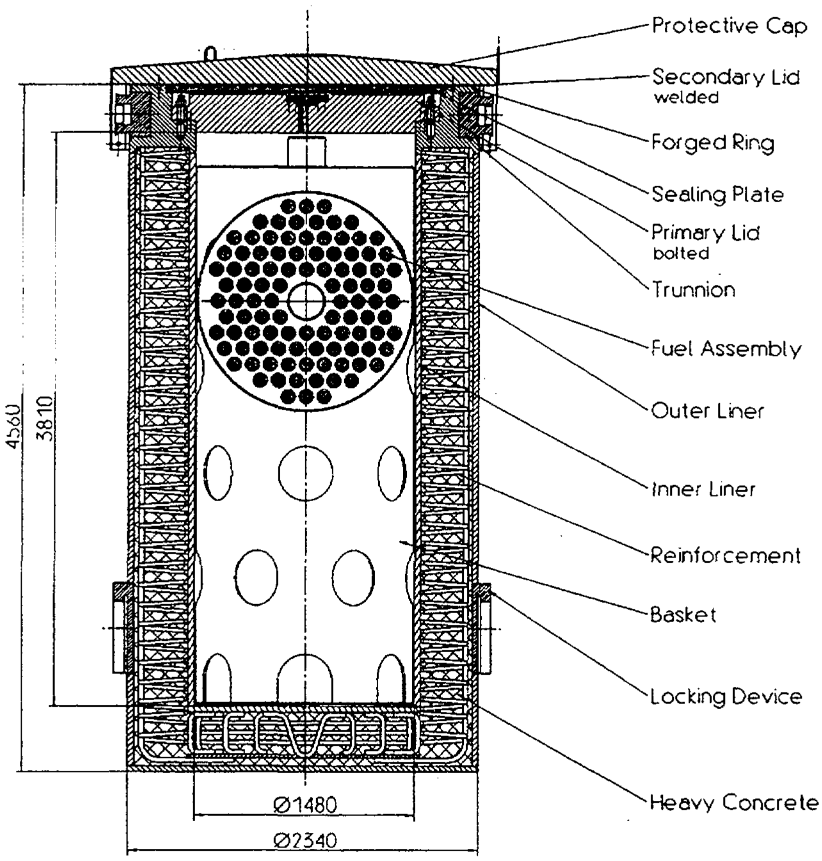


*FIG. 2. CONSTOR RBMK storage cask*

5.STORAGE SITE AT IGNALINA NPP

The INPP spent RBMK fuel storage site (see Fig. 3) is designed as an open air storage for an operational time of 50 years. The spent fuel storage facility is located at the INPP site at a 700 m distance from the INPP Unit 2 and at a 400 m distance from the Drukshai lake.

The main parts of the spent fuel storage are:

- the concrete storage pad for loaded and empty casks;
- production building;
- transformer substation;
- rain-water drainage system;
- observation wells;
- engineering service lines;
- checkpoint;
- radiation and dose rate control systems;
- roads and railways.

The spent fuel storage site is surrounded by a protective steel concrete wall and 3 rows of safeguard fence equipped with an alarm system. For safe handling of the storage casks a gantry crane will be used.

The casks are set in groups on the concrete pad at the site; the distance between the cask centers within each group is 3 m. The distance between the casks' groups is 4.1 m. Such a disposition allows the return of each cask if necessary. Along the whole perimeter of the storage there is a system providing a permanent dose rate control. The measuring signal's output will be transmitted to the INPP radiation control board.

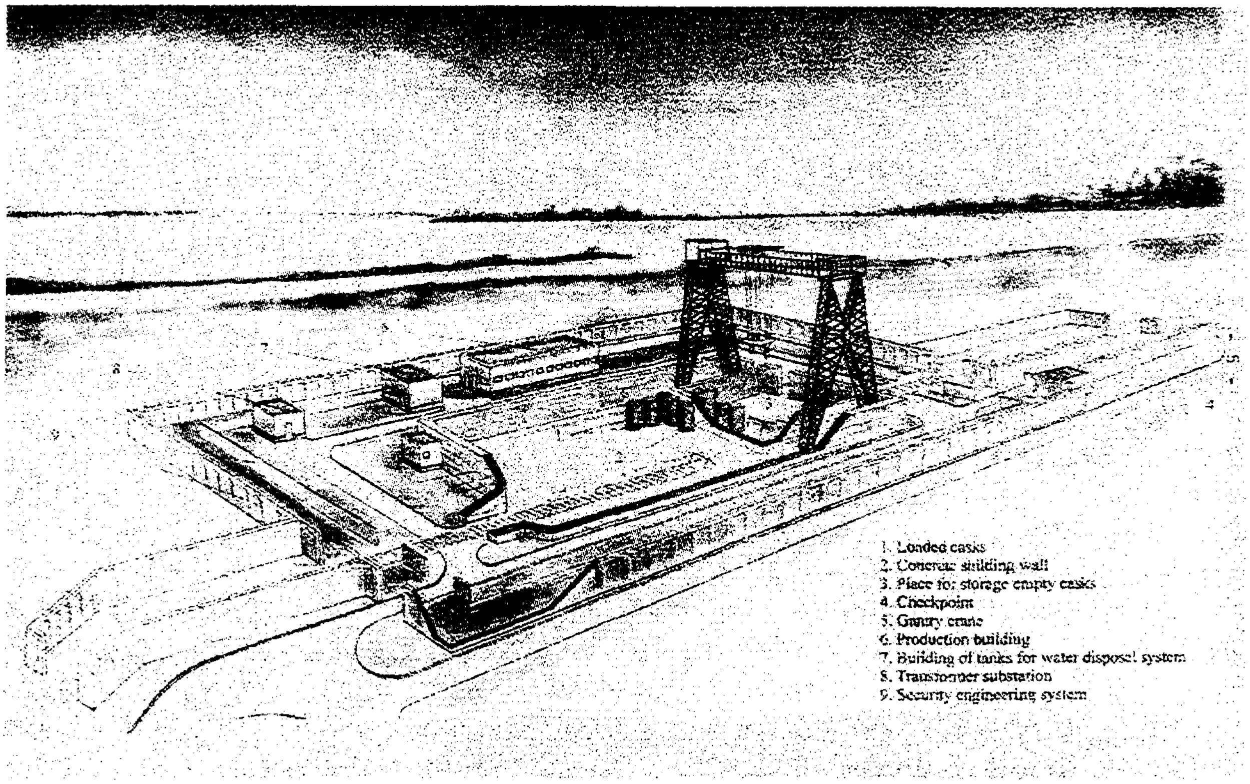


*FIG. 3. Ignalina spent RBMK fuel cask storage site*

The safety analyses for fuel handling and for cask handling at the site have been made and accepted by VATESI the Lithuanian Competent Authority. The site is planned to be put into operation in 1999, with the first loading of a CASTOR cask at the beginning of 1999. The delivery of the first CONSTOR RBMK cask will take also place at the beginning of 1999. Recently the spent fuel storage facility is in the stage of commissioning, with completion at the end of this year.

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# STATUS OF THE SPENT FUEL DRY STORAGE PROGRAMME FOR CERNAVODA NPP



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**Abstract**

The Cernavoda NPP Unit 1 (600 MWe Standard type) is in operation since December 1996. Within the framework of the R&D Radioactive Waste and Spent Fuel Management Programme, investigations, studies and research are carried out on site identification and conceptual designs for both a Spent Fuel Interim Storage Facility and a Spent Fuel Disposal Facility. The status of the work performed in the framework of this programme as well as the situation of the spent fuel resulting from the Research Institutes will be presented in the paper.

## 1. INTRODUCTION

In 1996, Romania became the $30^{th}$ country in the world operating a nuclear power plant. The Cernavoda Unit 1 equipped with a CANDU type reactor will produce about 9% of the Romanian electricity need. There are other 4 units erected on the Cernavoda NPP site. However, only Unit 2 is being prepared for completion. Measures for reduction of the radiological impacts on the environment have been taken according to the CANDU project, but it is necessary to ensure the improvement of the radwaste management in accordance with IAEA recommendations and the ALARA principle. According to the law, the handling of radioactive waste, including spent fuel from research and power reactors, must be safely managed and disposed off in such a way that the protection of human and environment, now and in the future, is ensured.

## 2. ROMANIAN SPENT FUEL AND RADIOACTIVE WASTE MANAGEMENT POLICY

The main objective in the management of radioactive waste is to protect current and future generations from unacceptable exposures to radiation from man-made radioactive materials. The sources of radioactive waste producers in Romania are users of radiation and radioactive materials in industry (including nuclear electricity generation), medicine, agriculture and research and the processing of materials that are naturally radioactive, such as uranium ores, thorium associated ores and phosphate fertilisers. The different types of radioactive waste are classified into four categories of waste: exempted waste, low level waste, medium level waste and high level waste. This classification is related to the concentration of the radioactivity in the waste and hence to the intensity of the emitted radiation.

In Romania, spent fuel is produced both in the power reactor and in the research reactors. The former RENEL (the Romanian Electricity Company) was responsible for the waste management of the Cernavoda NPP. The spent fuel which contains 99% of total radioactivity will be stored for 7 to 10 years in the so called Spent Fuel Storage Bay. RENEL was also responsible for the waste management of the Fuel Fabrication Plant and, through the Institute for Nuclear Research (ICN), for the waste management of the 14 MW TRIGA reactor (used for material testing) and the associated irradiation examination facility. The Institute of Physics and Nuclear Engineering "Horia Hulubei" (IFIN-HH) is responsible for the waste management of its VVR-S research reactor. The spent fuel assemblies from the research reactor are kept in a wet storage pool.

Based on a governmental decision, the Romanian power sector has been restructured this summer. The first step was the creation of a National Power Grid Company (CONEL SA) and the separation of the nuclear activities from RENEL into a National Nuclear Company "NuclearElectrica" (SNN SA). The National Power Grid Company is responsible for power transmission, dispatch and the thermal and hydro power branches. The National Nuclear Company "NuclearElectrica" comprises three entities, i.e. CNE PROD for nuclear generation (unit 1), CNE INVEST for nuclear development (units 2 to 5) and FCN for fuel fabrication (the nuclear fuel plant). The responsibility for the management of the spent fuel from NPP Unit 1 falls under SNN SA/CNE PROD. During the second step, the national power regulatory body will be formed and in the third step, the thermal and hydro power branches will be separated and privatized.

## 3. OVERVIEW OF THE SPENT FUEL PRODUCERS IN ROMANIA.

### 3.1. Research facilities

#### *3.1.1. Institute for Nuclear Research(ICN), Pitesti*

Since 1978, a TRIGA reactor is in use at Pitesti and it is expected to stay in operation for other 15 years. The facility contains a pool, in which the TRIGA reactor with a nominal power of 14 MW and another pulsed reactor are placed. Both reactors are using fuel.

The reactor research facilities contain a hot cell in which radioactive sources up to 1 MCi (Cs-137) can be handled. In the hot cell, research work can be done with irradiated fuel bundles, for example examination of irradiation damage and burnup determination. In the temporary storage pool next to the reactor pond are now 4 spent fuel clusters. The spent fuel arising is low. The return of the spent fuel to US for reprocessing is provided under the contract.

#### *3.1.2. The Institute of Physics and Nuclear Engineering "Horia Hulubei" (IFIN-HH), Magurele*

IFIN - HH is the owner of the Russian VVR-S research reactor type. The VVR-S is a tank-type research reactor of 2 MW nominal thermal power and a maximum neutron flux of $2 \times 10^{13}$ n/cm$^2$/s. The reactor designed and manufactured in Russia, has been in operation since 27 July 1957, without any recorded events and without significant refurbishment. The reactor is currently shutdown, awaiting a safety audit to be made by the regulator (CNCAN). Depending on the results of the audit, a decision on the decommissioning activities will be taken.

Basically, it is planned to store the spent fuel for 40 to 60 years followed by direct disposal. Currently, the spent fuel is being cooled in the AR pond for one year and in an adjacent spent fuel storage building for longer term storage. The storage system is being assessed by the plant operator, whilst the AFR at the Magurele site and the medium and long term storage options are also evaluated.

### 3.2. Cernavoda Nuclear Power Plant, Unit 1

The Cernavoda Nuclear Power Plant was originally planned as a four unit plant equipped with CANDU 600 reactors. Within three years of construction start, this was increased to five units. The CANDU 600 reactor uses natural uranium, is moderated and cooled with heavy water under pressure and has a thermal power of 2,180 MW. The original design is developed by AECL, Canada. Each unit has a General Electric turbine (1,500 r.p.m.), which operates with saturated steam. The unit produces an electrical power of 706 MW with a cooling water temperature of 15°C.

Unit 1 is operating since December 1996. At present, efforts are concentrated on Unit 2. The delay in work has caused a slippage of financial sources and not because of missing the governmental warranties. The project is going on continuously, however, at a slower pace.

The spent fuel is currently stored in the main storage bay located in the service building. The bay has a capacity of 49,000 fuel bundles. Approximately 5,000 fuel bundles are stored annually (80% reactor capacity factor) in the spent fuel bay. Under these conditions, the quantity of spent fuel discharged from the 5 units in 30 years is estimated to be about 14,400 tU.

## 4. ROMANIAN REGULATORY FRAMEWORK

### 4.1. Licensing authorities, structures and responsabilities

In Romania, the Law on the the safe deployment of nuclear activities, Law no.111/1996 published in 1998 empowered the National Commission for Nuclear Activities Control (CNCAN) as the national authorithy competent in the nuclear field exercising the regulations, authorizations and control powers.

CNCAN is a governmental organization responsible for the development of the regulatory framework, the control of its implementation and for the licensing of nuclear facilities. CNCAN issued a set of standards and regulations, based on IAEA recommendations, Canadian and US nuclear regulations. Some of these regulations have been updated and all are going to be revised and aligned to the current practice in the European Community.

The Ministry of Water, Forests and Environmental Protection is responsible for issuing environmental protection licences and permits through its local agencies for Environmental Protection. According to the Law No. 137/1995 on environmental protection, any industrial utility needs to apply to the local agency for an environmental licence or permit and these will be issued by the Ministry or its local agencies for Environmental Protection. For licensing a spent fuel facility, a special procedure for environmental impact assessment is required.

According to Law No. 3/1978 and Law No. 100/1998, the Ministry of Health is responsible to issue the sanitary licences.

A public hearing is also required for licensing.

### 4.2. Legislative framework

In Romania, as already is mentioned, the legal framework that co-ordinates the achievement of nuclear objectives consist of:

- Law 111/1996 republished in 1998 regarding the safe operation of nuclear activities;
- Law 137/1995 regarding environmental protection;
- Norms, requirements, guidelines, orders issued by CNCAN or ministries;
- International Multilateral Treaties and Conventions.

Since December 1996, a new Atomic Act, the Law No. 111/1996 on the safe deployment of nuclear activities, has come into force. The Law provides for a comprehensive legal framework for regulation, authorisation and control of activities concerning the peaceful use of nuclear energy including nuclear safety conditions, protection of workers, the public, the environment and property. The development of nuclear energy should be achieved with a minimum risk foreseen by the regulations and, in addition, by observing the obligations pursuant to the agreements and the conventions of which Romania is a part.

Recently the Law No. 111/1996 was modified and the amendaments entered into force in February 1998. Therefore, it respects now exactly the provisions of the Safety Convention and is armonized with the EU legislation.

According to this Law the handling of radioactive waste resulting from users of radioactive materials in medicine, agriculture, industry, research and from nuclear electricity generation must be safely managed and disposed off in ways that ensure the protection of humans and the environment, now and in the future.

The Law provides the legal requirements for radioactive waste management. According to this Law, the waste producer bears the responsibility for the management of his radioactive waste and also for the financial and material arrangement for covering the collection, transport, treatment, conditioning and disposal of the waste arising from the deployment of his activities and also for decommissioning of his facilities. The licensee shall pay a legal mandatory tax for financial contribution to the Radioactive Waste Management and Decommissioning Fund.

On 30 March 1997, CNCAN issued the draft of the Law on setting up of the Radioactive Waste Management and Decommissioning Fund. The draft of the law is still under review.

Dose limits and risks are regulated by the Republican Standards (Norms) of Radioprotection/1976, which have been updated according methods and values in [1].

According to this, the occupational exposure of any worker shall be controlled in such a way, that the following limits are not exceeded:

- an effective dose of 20 mSv per year averaged over five consecutive years;
- an effective dose of 50 mSv in any single year;
- an equivalent dose to the eye lens of 150 mSv in a year; and an equivalent dose to the extremities (hands and feet) or the skin of 500 mSv in a year.

As regards to the population, exposure of members of the public attributable to the practices shall not exceed the following limits that shall apply to estimated average doses to the relevant critical group:

- an effective dose of 1 mSv in a year;
- an equivalent dose to the eye lens of 15 mSv in a year.

### 4.3. Licensing requirements / Procedures for spent fuel storage and disposal

The practices and experience gained during the licensing process for the Cernavoda 1 NPP will be used and adapted for the licensing requirements/procedure needs for the interim storage and disposal facilities. Also, the experiences and practices used in the Western countries will be considered by CNCAN. The IAEA documents and recommendations will be used to develop the specific CNCAN requirement.

Licences are being issued in a staged manner against safety submissions initially in a preliminary format then resubmitted in a more elaborated format. The licence is limited in time, non-transferable and may be suspended or withdrawn whenever violations of the law or unanticipated situation, which pose unacceptable risks, do occur.

Basically, the facility compatibility with the site is being evaluated in an initial safety analysis, to prove fulfilment of the environmental and safety criteria, with the aim to issue a **siting licence**.

The **construction licence** is being issued on a system-by-system basis and the preliminary safety report shall demonstrate that the facility design comply with the safety requirements as enacted. When adequate information is available, the revised preliminary safety report is aimed to support the application for an overall construction permit to replace the partial construction permits for each system or structure, or as the case may be, identifying the design changes needed to comply with the whole set of licensing requirements.

Then, the as-built facility is checked against licensing requirements in the first issue of the final safety reports, with the aim to issue a **commissioning licence**, with various holding points. At the end, a revised final safety report will incorporate facility behaviour during the tests for set up the basis for an initial operation. Then, a reassessment of the plant behaviour will allow issuing an **operational licence**. The renewal of the licence will be based again on facility's safety reassessment. Similarly, facility modification, upgrading, retrofitting or decommissioning does require specific licence.

## 5. STATUS OF ROMANIAN SPENT FUEL MANAGEMENT

The status of the spent fuel management from the research area was presented in chapter 1. The most important quantity of the nuclear spent fuel is and will be produced by the units from the Cernavoda NPP.

In 1992, RENEL, the former owner of the Cernavoda NPP, had initiated a Research & Development Spent Fuel Management Subprogramme, as part of a Radioactive Waste Programme, with the main following directions:

- The establishment of the conception for the Spent Fuel Interim Storage Facility .
- The selection and the substantiation of the Spent Fuel Final Disposal Site.
- The development of a concept for this facility.

The main targets of the Programme are presented in Table I.

TABLE I. RADIOACTIVE WASTE PROGRAMME TARGETS

| FACILITY/ACTIVITY | COMMISSIONING (year) |
|---|---|
| Spent Fuel Interim Storage Facility (SFISF) | 2005 |
| Opening the Spent Fuel Final Disposal Laboratory | 2035 |
| Opening the Spent Fuel Final Disposal Facility (SFDF). | 2050 |
| Closing of the Spent Fuel Final Disposal Facility | 2075 |

### 5.1. Spent fuel interim storage facility

The N.P.P. Cernavoda project includes a Spent Fuel Storage Bay (SFB) with a storage capacity for ten years operation of one unit at a 80% load factor. Based on this load factor, the quantity of spent fuel discharged from the 5 units in 30 years is estimated to be about 14,400 tU, with a quantity of about 94 tU discharged year from one unit, that means about 5,000 bundles per year.

Taking into account that the capacity factor of the Cernavoda Unit 1 obtained in 1997 was 87.3 % and that the capacity factor increased much in 1998 (for example to 92.2% in February), it is expected that the quantity of spent fuel stored in the SFB will increase. Under these conditions, it is expected that the deadline for commissioning of the Spent Fuel Dry Storage Facility has to be advanced.

The particularities of the Cernavoda NPP offer two possibilities for using existing areas and systems for loading and preparation of the spent fuel for dry interim storage. One possibility is to use the Reception Bay and the other one is to use the Spent Fuel Bay. If the SFB will be used for loading and preparation of the spent fuel for dry storage, the storage capacity of the bay will be reduced with about 2 years. In this case, the deadline will be before year 2004.

One important parameter for the SFISF is the spent fuel storage capacity. Taking into account the above facts and the uncertainty of how many of the planned units will be finished and commissioned and at which pace, it was decided that the spent fuel would result only from 2 units. The analysis related to the amount of spent fuel to be stored in the SFISF has proposed a modular concept which permits an investment according to the real spent fuel quantity discharged from the reactors.

The characteristics of CANDU fuel present advantages and disadvantages in the selection and design of the solution for the facility and constitute to an other design input data. Related to this data, we considered the experimental results the from the Canadian experience[2].

The CANDU fuel contains natural uranium, has a low weight and small dimensions and is free from a criticality hazard in water. Also, the low burnup (90 MW·h/kgU) determines both the low released thermal power and the low specific activity. These characteristics indicate that the spent fuel is easy to be managed. On the other hand, taking into account the large quantity of fuel (94 t/year) and the large number of fuel bundles (approximately 5,000/year) to be transferred and prepared for storage require a special attention in the design process.

First of all, the possibility to place this Interim Storage Facility on the Cernavoda NPP site or in its vicinity has been studied. At a preliminary stage, 6 possible locations within or outside the Cernavoda NPP area have been analysed for interim spent fuel storage. As a conclusion of these studies, the optimum location seems to be at the NPP boundary. This selection has as a direct advantage the avoidance of the transport on public roads and the simplification of the transport system.

A location close to Unit 5 of the Cernavoda NPP was selected. This position on natural strata cannot make a significant perturbation to other activities in this area and has the benefit of existing roads, utilities, other services, etc.

The geotechnical characteristics of the foundation soil (muddy clay or silty clay) are not convenient (bearing capacity 2.0 daN/cm$^2$). Foundation on a concasated stone pad (with 2.5 m in thickness) should be recommended, leading to an increase in bearing capacity to 3.4 daN/cm$^2$.

Regarding the storage system, already different types have been analysed:

a) Ponds
b) Dual purpose casks
   b1) One piece flask (DSC flasks/Ontario-Hydro, CASTOR/GNS, TN/Transnucleaire)
   b2) Canister in a concrete flask (TranStore/BNFL
c) Vaults
   c1) concrete monolithic module (CANSTOR/AECL)
   c2) modular concrete vault (MVDS/GEC-Alsthom, CASCAD/SGN).

Taking in view the limitations imposed to the area necessary for the extending of the existing pool and the disadvantages related to spent fuel storage in water, we renounced to develop furthermore the ponds solution.

The Canadian solutions were deeply analysed first of all. Also, other European solutions were considered, taking in view the particularities of our site, a crowded site, and the fact Romania is a European country.

As a result of the preliminary analyses and evaluation, three ISF solutions were selected to be compared and more deeply studied. For the comparison, a methodology with a low degree of subjectivity was requested. IAEA documentation [3, 4] and a method for technical product evaluation developed by the Politechnical University from Bucharest were consulted. Based on these documents and the characteristics of the SFISF, the comparative analysis had two evaluation levels with a different degree of complexity:

- level 1: analysis based on the general evaluation of the solutions;
- level 2: analysis based on main systems evaluation (including main components).

Four criteria for comparing the solutions (costs, safety aspects, technical aspects, interface with nuclear power plant) were selected and their weight factors established. The nuclear safety criterion refers to the capability to fulfil certain fundamental requirements: adequate containment, shielding for gamma and neutron radiation, heat dissipation and criticality. Also, it must be sufficiently strong against external hazards. The technical criterion refers to the technical aspects of the objective and takes into account the performances in operation, materials, tests and inspection during operation, systems redundancies, necessary auxiliary systems, decommissioning, spent fuel recovery.

All the storage solutions require, more or less, modifications of the NPP Cernavoda project, including new systems, modification or linking with the existing systems, modification of procedures. The interface with the nuclear power plant criterion takes into consideration these aspects.

The criteria were analysed for each solution, taking into consideration the weight of each criterion. The result was an order among the three solutions. As a conclusion of this analysis, 2 solutions were selected for further analysis in a technical/economical study for recommending the optimal solution for the facility, i.e.: the Concrete Monolithic Module and the TranStore ™ solution.

At the same time, the documentation required by the regulatory body for obtaining a site licence for SFISF is being prepared.

**5.2. Spent fuel disposal facility**

Within the framework of the Spent Fuel Management Subprogramme, some documentation studies concerning the identification of the proposed site, natural performances and geological assessment of some deep geological formations, have been included.

The following geological formations were analysed:

a. Platform green schists from Central Dobrogea Worst. Their qualities are: massively, dimensions (thickness ~ 3,000m, surface > 1,000 km$^2$), homogeneity (about 300 m thickness in upper level with fine granulation), the lowest permeability. In addition, a low density of the population and short distance from the Cernavoda NPP. Five areas were selected in the West of Central Dobrogea.

b. Salt formations from Transylvanian Depression. Their qualities are: the plasticity and the low permeability in a large massive body (about 500 to1,000 m thickness at 300 m depth, in a big structure = 16,000 km$^2$ in all Transylvania area with little exceptions). In addition, 80 to 99% NaCl and the lowest seismicity in Romania. Four bodies were selected.

c. Granite rocks in big bodies like batolit and lacolit with 200 to 700 km$^2$ surface, the thickness much more them 2,000 m, in zones with lowest seismicity (exception Macin site from Dobrogea) and their homogeneity, offer good conditions to the host facility. Unpredictable water circulation in zones with large hydro-energetical, a criteria which eliminated many sites from the seventeen bodies analysed. Four granite bodies were selected.

d. Volcanic tuft of Dej from Transylvanian Depression is the last Romanian geological formation in this hierarchy. The alterability, the permeability, the possibility of water and gas accumulations, eliminated many analysed sites. Only the north-east of Transyvania Lunca Bradului area offers good conditions (10 km$^2$ surface, 300 to 400 m thickness, at 600 to 1,000 m depth). In a second stage, minimum field investigations are necessary and two deep geological formations could be taken into account, i.e. platform green schists from the west part of Central Dobrogea and salt formations from Transylvania.

## 6. MILESTONES OF ROMANIAN STRATEGY

The first phase for a medium term storage period for spent fuel under safe conditions, so that the producers may provide protection of personnel, population and environment according to the responsibilities is stated by Law No. 111/96.

The deadline for commissioning the ISF is 2005. The achievement of this objective is on its way. The results from the studies performed in the last 5 years will permit to prepare a feasibility study. The documentation for obtaining the site licence and a technical-economical study for selection of the optimal solution of the ISF are under preparation.

A second phase is represented by the assessment of a strategy for long-period to use and adopt a proved disposal technology for the spent fuel, in direct correlation with a selected site. At this time, the year 2035 is considered to be the deadline to have a selected site for the Spent Fuel Disposal Facility. It is intend to open a laboratory for developing supplementary investigations on this site (about 10-15 years) before the commissioning of the Disposal Facility.

It is expected that the experience gained during the design, construction and commissioning of the first Unit of the Cernavoda NPP could be an important factor for the cost reduction in the area of the design and project management for these facilities.

## 7. CONCLUSIONS

The present spent fuel management strategy in Romania is to store the spent fuel in the AR storage pool for a minimum of 7 years and then to transfer the spent fuel in an AFR interim storage facility for several decades. In one word, the selection of "Wait and See" option in the spent fuel management strategy has been the best strategy for Romania until an internationally acceptable spent fuel management and back-end fuel cycle technology will be established.

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# CURRENT STATE AND PERSPECTIVES OF SPENT FUEL STORAGE IN RUSSIA

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**Abstract**

Twenty-nine power units at nine nuclear power plants, having a total installed capacity of 22 GW(e), are now in operation in the Russian Federation. They produce approximately 12% of the generated electricity in the country. The annual spent fuel arising is approximately 790 tU. The concept of the closed fuel cycle was adopted as the basis for nuclear power development in the Russian Federation, but until now this concept is only implemented for the fuel cycles of WWER-440 and BN-600 reactors. The WWER-1000 spent fuel is planned to be reprocessed at the reprocessing plant RT-2 which is under construction near Krasnoyarsk. The RBMK-1000 spent fuel is not reprocessed. It is meant to be stored in intermediate storage facilities at the NPP sites. The status of the spent fuel (SF) stored in the storage facilities is given in the paper. The principal characteristics of the fuel cycles of the Russian NPPs in the period up to 2015 is also given in the report. The key variant of the current spent fuel management at RBMK-1000 NPPs is storage in at-reactor and in away-from-reactor wet storage facilities at the power plant site with a capacity of 2,000 tU. The storage capacity at the operating RBMKs (including the increase due to denser fuel assembly arrangement) will provide SF reception from the NPPs only up to 2005. For RBMK spent fuel, intermediate dry storage is foreseen at power plant sites in metallic concrete casks and thereafter transportation to the central storage facility at the RT-2 plant for long-term storage. The SF will be reprocessing after completion of the reprocessing plant at RT-2. In the Programme of Nuclear Power Development in the Russian Federation for the period 1998 to 2005 and for the period until 2010 year, provisions are made for the construction of a central dry storage facility before 2010. The facility will have a design capacity of 30,000 tU for WWER-1000 and RBMK-1000 spent fuel and is part of the reprocessing plant RT-2. The paper considers the status and prospects of spent fuel management at shut down AMB reactors. For the solution of the spent fuel problem of the Bilibino NPP, the concept of the final stage of spent fuel management is developed taking into account the low density of the population and the long distance from main communication ways. Reliable data about the behaviour of construction materials of assemblies and storage facilities under these conditions are necessary. Recommendations on the duration of wet storage are needed.

## 1. STATUS OF SPENT FUEL STORAGE

Since 1964, 9 NPPs with 33 nuclear reactors of different types were put into operation in Russia with a total installed capacity of more than 21.2 GW(e). At present, there are 29 power units in operation, 11 RBMK- 1000 units, 6 WWER-440 units, 7 WWER-1000 units, 4 EGP-6 units and one fast reactor, the BN-600 unit. Four units are disconnected from the grid, 2 WWER units at Novovoronezh NPP and 2 AMB units (i.e. channel type reactors) at Beloyarsk NPP and their decommissioning is being carried out.

The concept of the closed fuel cycle was adopted as the basis of nuclear power development in Russia, but until now this concept is only for spent fuel (SF) of WWER-440 and BN-600 reactors fully implemented. Spent fuel is transported to the reprocessing plant RT-1 (PO "Mayak") and recovered uranium is used for the fabrication of, RBMK fuel. WWER-1000 spent fuel is transported to the central storage facility at Krasnoyarsk Mining - Chemical Plant (KMCP). After the completion of the reprocessing plant RT-2 construction at the KMCP site WWER-1,000 SF will be reprocessed too. For SF of RBMK and AMB reactors only the initial stage of the fuel cycle is realized: storage at NPP sites in away-from-reactor (AFR) and at-reactor water pools. The status of the spent fuel inventory stored at power plants facilities as of 1 July 1998, is shown in Tables I and II.

One can see from Table II, that the situation at with RBMK and EGP plants is critical, since the amount of spent fuel arisings comes close to the capacities of the spent fuel storage facilities and can maintain the operation of the power plants only until 2005. For other power plants this problem does not exist for the time being. The principal characteristics of the fuel cycle of the Russian NPPs in period to 2015 are shown in Table III.

TABLE I. THE SPENT FUEL INVENTORY AT NPPs WITH CHANNEL TYPE REACTORS

(status 01.07.1998)

| NPP number of units | Capacity of AR and AFR storage facilities (FAs) | Number of fuel assemblies (FAs)/amount of tU | Remarks |
|---|---|---|---|
| LENINGRAD | AR -1 - 1,700 | 940/108 | 1,784 gas leaking FAs |
| 4 RBMKs | AR -2 - 1,700 | 1,232/142 | Unit No 3 is repaired |
| the annual discharge | AR -3 - 1,700 | 0 | |
| from the reactors is | AR -4 - 1,700 | 1584/182 | |
| ~1,800 FAs | AFR - 17,600 | 29671/3,412 | |
| | (35,200*) | 31,843/3,662 | |
| | Total - 42,000 | | |
| KURSK | AR -1 - 1,700 | 324/37.3 | 663 gas leaking FAs |
| 4 RBMKs | AR -2 - 1,700 | 1,384/159.2 | |
| the annual discharge | AR -3 - 1,700 | 1462/161 | |
| from the reactors is | AR -4 - 1,700 | 1,592/183 | |
| ~1,800 FAs | AFR - 17,600 | 19,8481/2,283 | |
| | (29,600*) | 29,934/2,867 | |
| | Total - 36,400 | | |
| SMOLENSK | AR -1 - 3,800 | 2,310/265 | 258 gas leaking FAs |
| 3 RBMKs | AR -2 - 3,800 | 3,336/384 | |
| the annual discharge | AR -3 - 3,800 | 3,170/364 | |
| from the reactors is | AFR 13,600 | 3,780/436 | |
| ~1,350 FAs | Total 25,000 | 12,596/1,449 | |
| BELOYARSK | AR -1 - 2,500 | 2,500/95 | The number of |
| 3 units | AR -2 - 2,500 | 2,500/95 | defective FAs is |
| - 2 AMBs under | AR -3 - 2,438 | 1,228/66 | unknown |
| decommissioning | Total - 7,438 | 6,228/256 | |
| - 1 BN-600 | | | |
| BILIBINO | AR -1 - 2,000 | 1,967/49.4 | Defective FAs - 0 |
| 4 EGP-6 | AR -2 - 2,250 | 2,244/57.2 | |
| | AR -3 - 2,000 | 113/2.8 | |
| | Total - 6,250 | 4,324/108.4 | |

* The capacity of the AFR after reracking

The data in Table III indicates the dynamic of variation of spent fuel inventory at power plants since the first unit was put in operation up to 2005 year and principal tasks on the maintenance of operation of the power plants. Presented data and measures are defined in compliance with the governmental “Programme of Nuclear Power Development in Russian Federation for the period 1998-2005 and for the period until 2010” and the "Federal Programme on Radioactive Waste and Spent Nuclear Materials Management, their .Utilization and Disposal for the period 1996-2005". Data in Table III shows, that at WWER-440 and BN-600 power plants the rate of spent fuel accumulation corresponds to the rate of its removal from plant sites for subsequent reprocessing.

The reprocessing of WWER-440 and BN-600 spent fuel is carried out at the RT-1 plant (PO "Mayak"). The RT-1 plant was commissioned in 1977 and has a capacity of 400 t spent fuel/year. Up to now, the plant has reprocessed 2,500 t of WWER-440 and BN-600 spent fuel from Russian power plants. It is expected, that it will reprocess an additional 5,000 t SF.

TABLE II. THE SPENT FUEL INVENTORY AT NPPs WITH WWER - TYPE REACTORS

(status 01.07.1998)

| NPP number of units | Project capacity of AR and AFR storage facilities (FAs) | Number of fuel assemblies (FAs)/amount of tU | Remarks |
|---|---|---|---|
| BALAKOVO | AR - 1- 376 | 173/67.5 | 17 gas leaking FAs |
| 4 WWERs-1000 | AR -2 - 359 | 107/41.3 | |
| | AR -3 - 373 | 1.62/62.4 | |
| | AR -4 - 596 | 116/44.8 | |
| | Total - 1,704 | 558/216 | |
| KALININ | AR -1 - 413 | 239/96.3 | 28 gas leaking FAs |
| 2 WWERs-1000 | AR -2 - 413 | 245/98.7 | |
| | Total - 628 | 4841195 | |
| KOLA | AR -1 - 616 | 173/67.5 | 38 gas leaking FAs |
| 4 WWERs-440 | AR -2 - 637 | 188/21.8 | |
| | AR -3 - 662 | 483/56.0 | |
| | AR -4 - 664 | 116/44.8 | |
| | Total - 2,576 | 558/156.1 | |
| NOVOVORONEZH | AR -1 - 1,050 | 296/36 | 52 gas leaking FAs |
| WWER-210 & 365 | AR -2 - 300 | 0 | |
| under decommissioning | AR -3 - 652 | 293/34 | |
| 2 WWERs-440 | AR -4 - 665 | 196/23.2 | |
| 1 WWER-1000 | AR -5 - 233 | 66/26.9 | |
| | AFR - 916 | 171/69.8 | |
| | Total - 3,816 | 1,022/189.9 | |

WWER-1000 spent fuel is unloaded from the core and stored in an at-reactor (AR) storage facility for a period to enable the decay heat to reduce to a level, which permits the transportation of the spent fuel from the NPP site (about 3 years). The rates of accumulation and removal correspond to each other, but this fuel is not being reprocessing for the time being, because the construction of the reprocessing part of the plant is not completed yet. The capacity of the storage facility in the Complex of the RT-2 plant is 6,000 t and has at present accumulated 2,120 t SF from Russian WWERs-1000.

At NPPs with RBMK-1000 the key variant of spent fuel management now is storage in at-reactor and in away-from-reactor wet storage at the plant site with a capacity of 2,000 t. Unloading of spent fuel cassettes (two assemblies per cassette) is conducted with the reactor on-load and the number of these cassettes is 8 to 10 per week. After 1.5 - 3 years cooling in an AR storage facility, the spent fuel is transported to an away-from-reactor storage facility for long-term storage. The residual decay heat after 3 years of cooling is 0.22 kW/cassette and it permits to transfer the fuel to an away-from-reactor wet storage facility without water in a special on site cask. Each cassette is placed into a canister filled with water. The detailed description of spent fuel management technology at power plants in Russia is presented in [1]. To maintain the operation of power plants with RBMK-1000, the refurbishment of all storage facilities (AR and AFR) will be accomplished to increase their capacity by a factor 1.8, however, at the cost of the modification of the hanger's construction. This modification permits to decrease the distance between fuel cassettes in ponds with maintaining the established levels of safety.

For RBMK spent fuel, intermediate dry storage at the power plant sites is planned in metallic-concrete dual purpose casks (i.e. for storage and transportation), followed by the transportation to the central storage facility at the RT-2 plant for long-term storage and ultimate reprocessing. The full scale tests of the dual purpose casks were started this year and will be continued in 1999. At the present time, the cask storage facility project, which includes a hot cell for cassette disassembling, a system of radwaste treatment and space for storage of the casks, is under development too. In the "Programme of Nuclear Power Development in the Russian Federation for the period 1998-2005 and for the period until 2010" provisions for the construction of a central dry storage facility with a capacity of 30,000 t WWER-1000 and RBMK-1000 SF by 2010 are foreseen, as part of the reprocessing plant RT-2. On the base of a feasibility study, it was recommended to construct a vault type storage facility. Prior to the vault's operation, spent fuel will be stored in dual purpose casks at the power plants and at KMCP.

TABLE III. PRINCIPAL CHARACTERISTICS OF FUEL CYCLE OF RUSSIAN NPPs UP TO 2015

| Type of reactor, stage of fuel cycle | Characteristics of fuel cycle | 1998 | 2005 | 2015 | Remarks |
|---|---|---|---|---|---|
| WWER-210, 365, 440 BN-660, 800 | Number of units under operation/shut down | 9/2 | 9/2 | 11/4 | |
| SF is transported to RT-1 for reprocessing | Operation time, in reactor-year | 182 | 231 | 287 | |
| Annually transported ~ 140 t | SF discharged since 1964, FAs/tU | 24,720/ 2,554 | 30,600/ 3,158 | 39,380/ 3,425 | |
| First unit was commissioned in 1964 | SF stored at NPPs, FAs/tU | 3,359/ 393 | 3,752/ 439 | 1,821/ 209 | |
| WWER-1000, 640 | Number of units under operation / shut down | 7/0 | 8/0 | 14/0 | Dry storage facility at KMCP |
| SF is transported to KMCP | Operation time, in reactor-year | 182 | 231 | 287 | for WWER and RBMK SF will be |
| annually transported ~ 125 t | SF discharged since 1980, FAs / tU | 3,428/ 1,382 | 6,125/ 2,470 | 12,080/ 5,260 | commissioned in 2010 |
| First unit was commissioned in1980 | SF at NPPs, FAs/tU | 1,279/ 515 | 1,414/ 571 | 4,069/ 1,670 | |
| RBMK-1000, EGP-6, AMB | Number of units under operation / shut down | 17/2 | 18/2 | 18/7 | 2,000 FAs of AMB are stored at RT-1 |
| SF is stored at NPPs | Operation time, reactor-year | 308 | 416 | 489 | It is planned the transportation |
| First unit was commissioned in1964 | SF discharged since 1964, FAs/tU | 80,697/ 8,352 | 113,400/ 12,066 | 175,770/ 18,878 | RBMK SF to KMCP since2006 |
| | SF at NPPs, FAs/tU | 78,697/ 8,276 | 111,400 11,715 | 111,400/ 11,715 | |

For the Beloyarsk NPP, the main problem is ensuring the safety of spent fuel management at the shut down AMB reactors. During the operational period of these reactors, more than 7,000 cassettes (361 t) were unloaded, of which 2,000 cassettes were shipped to the storage facility at the RT-1 plant and 5,000 cassettes were packed into dry canisters accommodating 17 or 35 cassettes. The storage process is complicated by corrosion of 17 sections of carbon steel made canisters with damaged fuel. The paramount task is the creation of additional barriers for the damaged cassettes to prevent the release of radionuclides to cooling ponds and the development of an effective water cleaning system. It is planned to reload the spent fuel from the cooling ponds to steel casks for long-term dry storage and transportation. At the present time, design plans and specifications for these casks are under development and a test model is being fabricated.

Spent fuel of the 4 EGP-6 reactors at the Bilibino NPP is stored in three at-reactor storage facilities. During the operational period, ab[illegible]09 t of spent fuel was unloaded. As AR storage was filled, the spent fuel was converted to a dr[illegible]rage regime after one year of cooling. At the present time, the AR storage facilities are filled and converted to a dry storage regime. For the solution of the spent fuel problem of Bilibino NPP, the concept of the final stage of spent fuel management is being developed taking into account the low density of the population and the long distance from communication lines. The principal idea of the concept is the removal of the spent fuel from the NPP site, but the final stage is not yet determined. Thus, at the present time the main direction in solving the spent fuel management problem for RBMK, AMB and EGP reactors is the organization of long-term storage at the NPP sites and in a central storage facility for WWER-1000.

During the operational period of Russian power plants, 12,286 t of SF was discharged, of which 9,184 t is stored at power plants and 2,164 t was reprocessed at the RT-1 plant (PO "Mayak"). This situation, where the main portion of the spent fuel will be stored at the power plants, will continue over the next twenty years and consequently, the inventory of the spent fuel will increase.

In this respect, it will be necessary to solve the arising problems to ensure the long-term safety of spent fuel storage:

- Redesign of existing AR and AFR storage facilities, to increase their capacity up to the utmost permissible limits. The adjustment of storage projects in compliance with standing safety requirements and application for licences from regulating bodies for the operation of spent fuel storage facilities. This work has been done for the Leningrad and Kursk power plants and is currently under way for the Beloyarsk, Bilibino and Smolensk NPPs.

- Justification of the duration of spent fuel storage. For this purpose, investigations on the zirconium alloy cladding from the upper part of the RBMK fuel assemblies were conducted after 13 years of cooling in water pools. The results of the investigations showed, that dense oxide deposits were formed on the cladding and their thickness increased from the top to the lower part of the examined fuel rods. The change of the cladding thickness along the fuel rod was within allowable limits. The minimal thickness was 680 μm under a layer of deposits, that corresponds to 83% and doesn't cause apprehensions from the point of view of storage reliability. Insignificant hydrogenation of the cladding metal did not cause the deterioration of the mechanical characteristics. On the contrary, the strength characteristics were increased thanks to radiation hardening, but the ductility stayed rather high (16%). Thus, 13 years of RBMK fuel cooling in water pool caused no serious change in mechanical properties.

Full scale tests of zirconium and stainless steel specimens showed that the corrosion had a uniform character on the whole. The corrosion rate for zirconium alloys was 0.7 - 1.0 μm/year and for stainless steel 0.5 - 1.0 μm/year. These rates of corrosion guarantee the safe storage of spent fuel for at least 40 years for the cladding material and a much longer period for canisters and pool lining. The corrosion under the deposit layer causes some apprehension, but in this case it will be possible to guarantee not less than 40 years of safe storage without of special measures.

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# STATUS ON SPENT FUEL MANAGEMENT IN SPAIN



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**Abstract**

To confront the lack of spent fuel storage locations at the pools of the nuclear power plants, different actions have been undertaken by Enresa in conjunction with the Plant utilities. Basically, these measures have consisted in expanding the capacities of the spent fuel storage pools to their maximum capacity by exchanging their racks and in those cases where reracking is no further possible, dry storage will be provided, initially by means of dual purpose metallic casks.

## 1. INTRODUCTION

The Spanish Nuclear Programme consists of nine nuclear power plants with an overall capacity of 7.4 GWe, representing around 38% of the electricity production. In 1982, the National Energy Plan established the open cycle as the strategy to be followed, thereby halting the reprocessing option. As a consequence of this Plan, a state-owned company, ENRESA, responsible to the Ministry of Industry and Energy, was created in 1984 with the responsibility to manage all kinds of radioactive waste.

To comply with its responsibilities for the interim storage of spent fuel, ENRESA, together with the plant owners, designed a strategy based on expanding as much as possible the capacity of the reactor pools and providing dry storage facilities to those units where further reracking was no longer possible. Fig. 1 shows the location of the nuclear power plants and nuclear fuel cycle facilities.

## 2. SPENT FUEL PRODUCTION AND POOL STORAGE CAPACITY

Around 2,000 tU are presently stored in the reactor pools. being The overall forecasted spent fuel production over 40 years of plant operation is estimated at 6,800 tU.

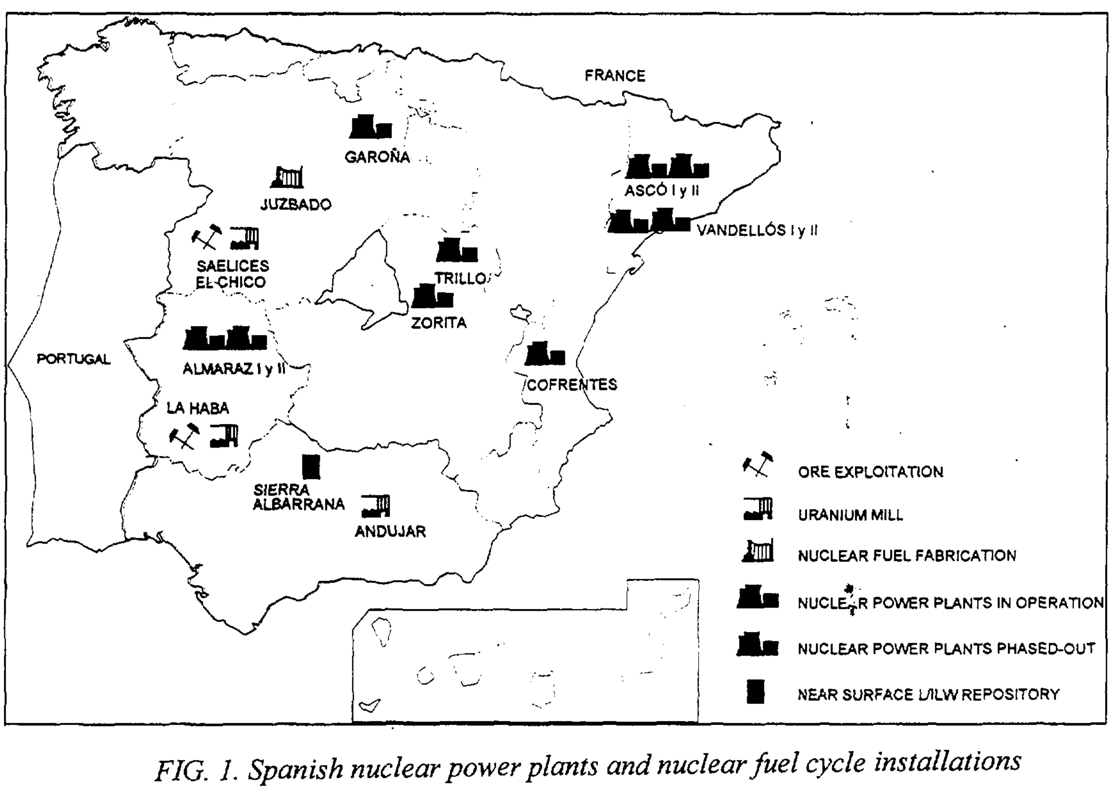


*FIG. 1. Spanish nuclear power plants and nuclear fuel cycle installations*

Along the last 7 years, the pools of all the plants have been reracked, increasing the overall spent fuel storage capacity from 1,901 tU (6,684 positions) up to 4,836 tU (14,487 positions), with no credit in these figures given to the full core reserves. Individual figures for the increase in pool capacities are provided in Table I.

The main characteristics of the new racks installed at the pools are the following ones:

- All of them are made of borated stainless steel;
- Racks are of the "free-standing" and "free-sliding" type;
- The pools are divided in two different regions: one -with a capacity for the Full Core Reserve (FCR) plus some additional locations - able to store unirradiated fuel, and the second one - comprising most of the pool area - where burnup credit is accounted for;
- They can accept consolidated fuel.

Spent fuel cooling systems have been able to accommodate the increase in thermal load with minor modifications except for the old plants, where some changes have been made. The overall forecast for spent fuel cumulative production, pool capacities and additional storage needs throughout the years is presented in Figure 2.

The first plant demanding out of pool spent fuel additional storage capacity will be Trillo by 2002, at a rate of 21 tU/year, followed around 2010 by the necessity for massive storage linked to the needs of 3 other plants and to the decommissioning phase of the older ones. This type of demand suggested the adoption of an At-Reactor (AR) solution followed by a centralized Away-From-Reactor (AFR) one to be implemented around 2012. This strategy assures the independence between the final disposal programme and the operation and future dismantling of the nuclear power plants.

## 3. AT REACTOR DRY STORAGE CONCEPT

The spent fuel storage design concepts adopted for each of the AR and AFR needs are briefly outlined below. Initial studies for dry storage systems were started back in 1988. At that time, a decision was made to select dual purpose casks among other options as these units would serve for the storage of spent fuel at-reactor and to transport it to the AFR Storage Facility whenever available. Once there they could be used as:

TABLE I. SPENT FUEL POOL CAPACITIES BEFORE AND AFTER RERACKING (RR) OF THE SPANISH NUCLEAR POWER PLANTS

| NPP | Total capacity - FCR[a] | | Stored fuel (12-97) | Year of saturation | |
|---|---|---|---|---|---|
| | before RR | after RR | | before RR | after RR |
| José Cabrera | 241 | 479 | 192 | 1999 | 2015 |
| Garoña | 1,327 | 2,209 | 1,284 | 1998 | 2014 |
| Almaraz 1 | 455 | 1,647 | 624 | 1992 | 2020 |
| Almaraz 2 | 455 | 1,647 | 620 | 1993 | 2022 |
| Ascó 1 | 431 | 1,264 | 516 | 1993 | 2013 |
| Ascó 2 | 431 | 1,264 | 432 | 1995 | 2016 |
| Cofrentes | 2,414 | 3,912 | 1,956 | 2000 | 2016 |
| Vandellós 2 | 415 | 1,437 | 452 | 1997 | 2014 |
| Trillo | 415 | 628 | 428 | 1996 | 2002 |

[a] Net - not accounting for Full Core Reserve (FCR) - capacity in fuel positions

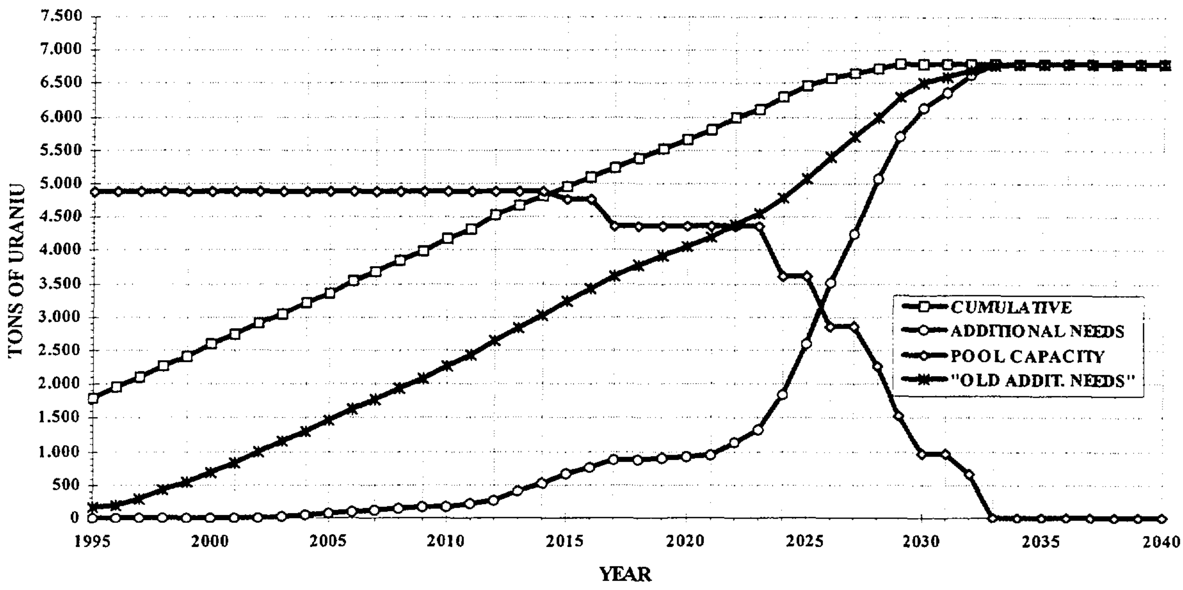


*FIG. 2. Spent fuel cumulative production, pool capacities and additional storage needs*

- storage units without the need to further handling of the spent fuel at the AFR installation; or
- transport units (after unloading the spent fuel thereby stored) to ship to the AFR facility the rest of spent fuel from the nuclear power plants.

A contract was awarded to NAC International under a co-operative agreement between VEPCO, EPRI and ENRESA to design this type of cask, known as the STC, which was licensed by the US. Nuclear Regulatory Commission (NRC) for both purposes in 1994 (COC for Transport) and 1995 (Storage).

The design of a similar cask to the STC, able to accommodate the spent fuel discharged from the German design Trillo nuclear Power plant, was awarded to the Spanish company Equipos Nucleares, S.A. (ENSA). This cask, known as the DPT, was licensed in 1996 by the Spanish Nuclear Regulatory Body, the Consejo de Seguridad Nuclear (CSN), also for both storage and transport modes.

The main characteristics of these two casks are briefly described in Table II. Figure 3 shows a sketch of the DPT cask. Manufacturing of the first DPT units has started in January 1998. The casks will be stored at the Trillo Plant inside a building which has been designed within the controlled area boundary for these purposes. The first loading of the casks and start-up of the installation are scheduled to take place early in 2000, thereby providing enough margin before the saturation of the spent fuel pool would take place.

## 4. AWAY FROM REACTOR MASSIVE STORAGE CONCEPTS

Former studies undertaken by Enresa in the past concluded in the preliminary design of an AFR installation which was a combination of pool and cask technologies. Along the last year a revaluation of that study has taken place which envisages a facility based mostly on dry technologies.

## 5. CONCLUSIONS

The strategy for providing the utilities with additional spent fuel storage capacity has been explained under the Spanish perspective. It is based on the expansion of the capacity of the pools and the selection of a modular and flexible solution for the needs of one of the plants until an AFR installation would be available before 2012.

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TABLE II. MAIN CHARACTERISTICS OF THE NAC-STC AND ENSA-DPT CASKS

| | STC | DPT |
|---|---|---|
| Capacity | 26 PWR | 21 PWR |
| Fuel parameters | | |
| Burnup (MWD/MTU) | 40,000 / 45,000 | 40,000 |
| Cooling time (years) | 6.5 / 10 | 5 |
| Initial enrichment (% $^{235}U$) | 4.1 / 4.2 | 4 |
| Heat load per assembly (watts) | 850 | 1,160 |
| Main dimensions | | |
| Cavity diameter (mm) | 1,803 | 1,676 |
| Cavity length (mm) | 4,191 | 4,331 |
| Cask body outer diameter (mm) | 2,210 | 2,108 |
| Neutron shield outer diameter (mm) | 2,514 | 2,368 |
| Impact limiter diameter (mm) | 3,150 | 3,150 |
| Overall length with impact limiters (mm) | 6,528 | 6,629 |
| Maximum weight of the package (kg) | 114,000 | 118,000 |

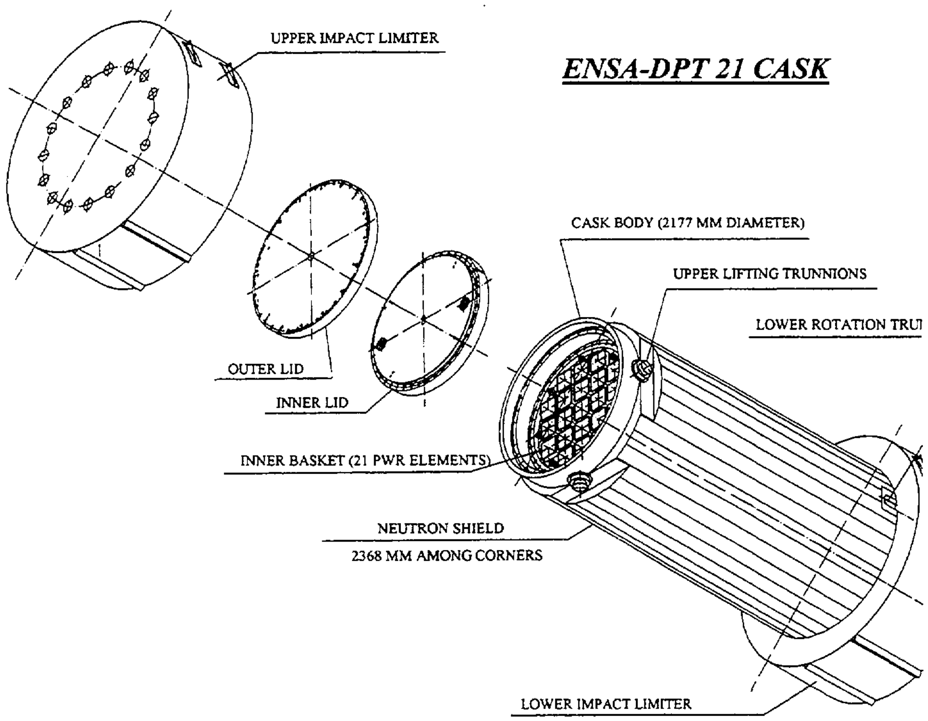


*FIG. 3. Sketch of the DPT cask*

# RADIOACTIVE WASTE MANAGEMENT IN SWEDEN EXPERIENCES AND PLANS



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**Abstract**

Since some years, the necessary facilities are in operation in Sweden for the safe transport and storage of radioactive waste and spent fuel from nuclear power production. These include a final repository, SFR, for short-lived low and intermediate level waste, a central interim storage facility, CLAB, for spent fuel and a sea-based transport system. The experiences from the operation of these facilities have generally been very good. The next step is the development of an encapsulation facility and a deep repository for the spent nuclear fuel. R&D-work on direct disposal have been conducted in Sweden for more than 20 years. In the preferred method the spent fuel will be encapsulated in a copper canister with a steel internal structure, and the canister will then be disposed of at about 500 metres depth in the Swedish bedrock. The siting and design of the encapsulation facility and the deep repository is now in progress.

## 1. INTRODUCTION

About 50 % of the electricity in Sweden is generated by means of nuclear power from 12 reactors located at four sites (see Fig. 1) and with a total capacity of 10,000 MW. Nine of the reactors are BWRs and three PWRs. The first commercial reactor was put in operation in 1972 and the latest in 1985. Discussions about closing the first reactor for political reasons is going on in Sweden. If all reactors are operated until 2010, about 8,000 tonnes of fuel will have been generated and will have to be taken care of as spent nuclear fuel. Today, about 3,700 tonnes have been generated.

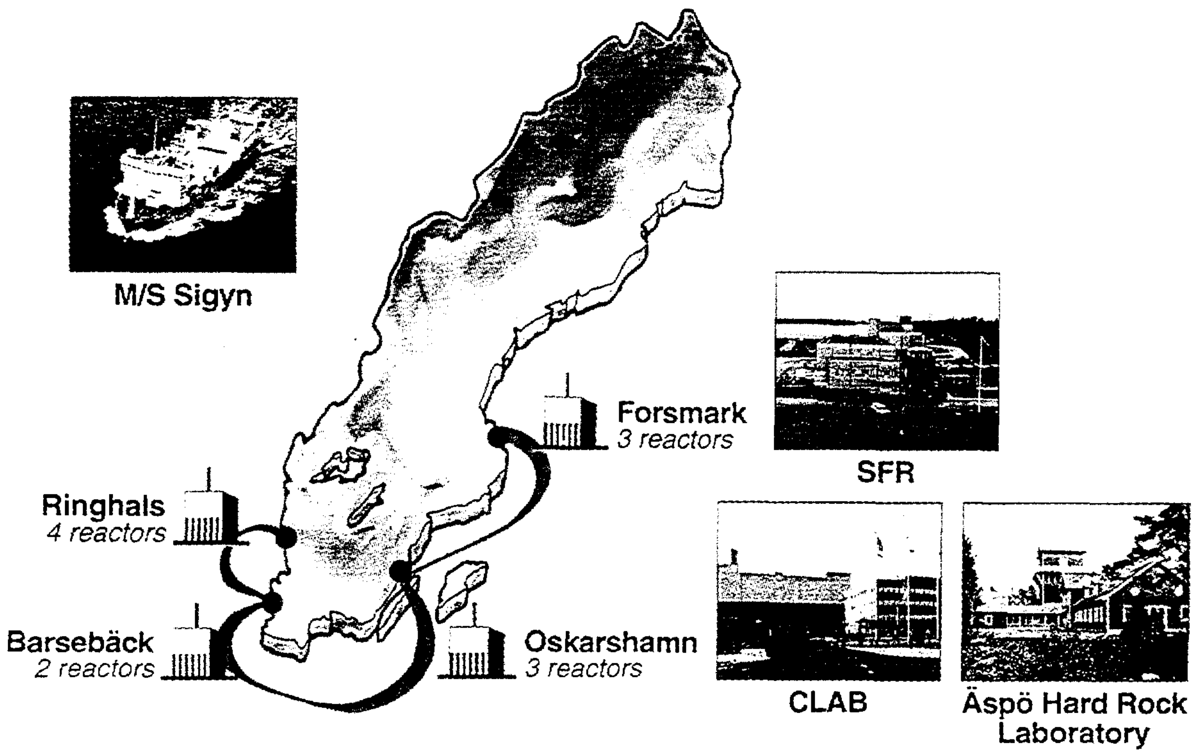


*FIG. 1. Location of the Swedish nuclear power plants and nuclear facilities*

The responsibility for the management of the spent nuclear fuel, as well as for other radioactive residues from nuclear power production, lies with the operators of the nuclear power plants, i.e. the four nuclear utilities. The utilities have jointly created SKB, the Swedish Nuclear Fuel and Waste Management Company, to safely manage the spent fuel and radioactive waste from the reactors to final disposal. The task of SKB is thus to plan, construct, own and operate the systems and facilities necessary for transportation, interim storage and final disposal.

SKB has developed a system that ensures the safe handling of all kinds of radioactive waste from the Swedish nuclear power plants for a long time period ahead. The keystones of this system are (Fig. 2):

- A transport system which has been in operation since 1983;
- A central interim storage facility for spent nuclear fuel, CLAB, in operation since 1985;
- A final repository for short-lived, low and intermediate level waste, SFR, in operation since 1988.

The remaining components of the system that are now being planned are:

- An encapsulation facility for spent nuclear fuel; and
- A deep disposal facility for encapsulated spent fuel and other long-lived radioactive wastes.

The costs for the management of all radioactive residues from the nuclear power plants, as well as for the decommissioning of the plants is borne by the operators of the reactors. To cover all future costs and ensure that means will be available, a fee is levied on nuclear electricity. The fee is about 0.015 SEK/kW·h (about 2.0 US mills/kW·h) and is paid to the state and is used for the activities of SKB.

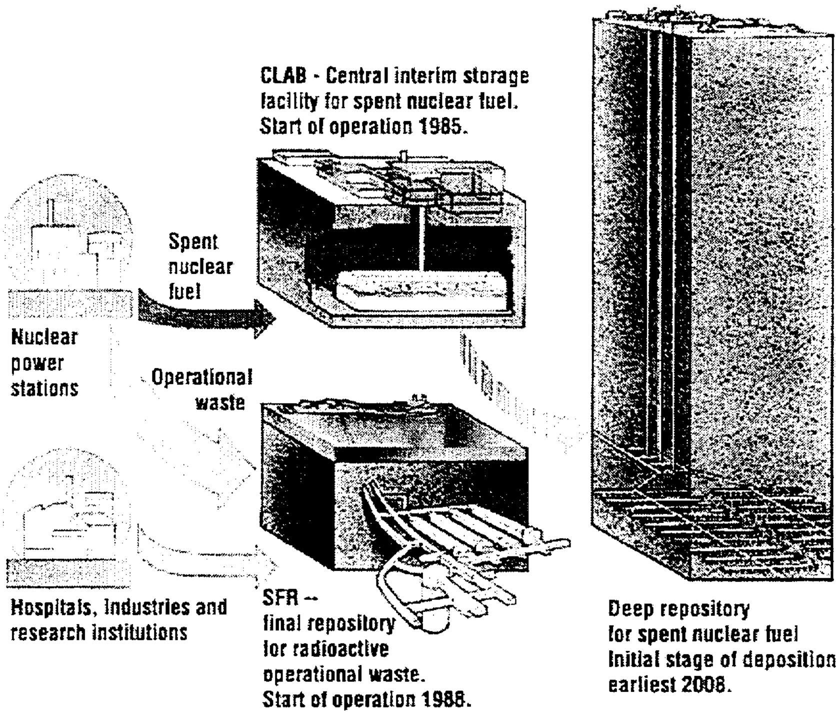


*FIG. 2. The Swedish waste management system*

## 2. SWEDISH FINAL REPOSITORY FOR RADIOACTIVE WASTE (SFR)

SFR, the Swedish Final Repository for Radioactive Waste, is a central disposal facility for all short-lived low and intermediate level waste from the operation of the nuclear power plants. SFR is located close to the Forsmark nuclear power plant on the east coast of Sweden. The storage capacity of SFR is at present 60,000 m$^3$. Later, it will be extended to accommodate also the waste from the decommissioning of the nuclear power plants. Similar waste from the use of radioisotopes in medicine, research and industry will also be disposed of in the SFR.

## 2.1. Repository design

The SFR is built in bedrock at about 50 meters depth. It is built underneath the sea-bottom about one kilometer off shore from the harbour at Forsmark. The repository has been designed so that the waste will be isolated to prevent any escape of toxic components into the environment in harmful quantities even after the facility has been sealed and abandoned. No institutional control is foreseen after sealing. The isolation is accomplished by different barriers surrounding the waste.

In the SFR, the waste is disposed of in different rock caverns (Fig. 3), that have been adopted to the different waste types and their different demands on handling and barriers [1]. The most active waste is disposed of in a concrete silo built in a cylindrical rock cavern. The silo is about 50 m high and has a diameter of 25 m. The thickness of the concrete wall is about 0.8 m. The waste is deposited in the silo by remote handling and successively surrounded by concrete grout. In between the silo wall and the rock bentonite clay is placed, which after closure of the facility will ensure that the nuclide transport is governed by diffusion, i.e. very slow.

Waste with less activity is disposed of in four rock chambers. The design of the caverns are different and depends on the type and dose-rate of the waste packages. The caverns are about 160 m long and 14 -18 m wide. In one cavern, for intermediate-level waste with a potential for gas generation, remote handling is used as in the silo, while in the other caverns a forklift truck is used for handling.

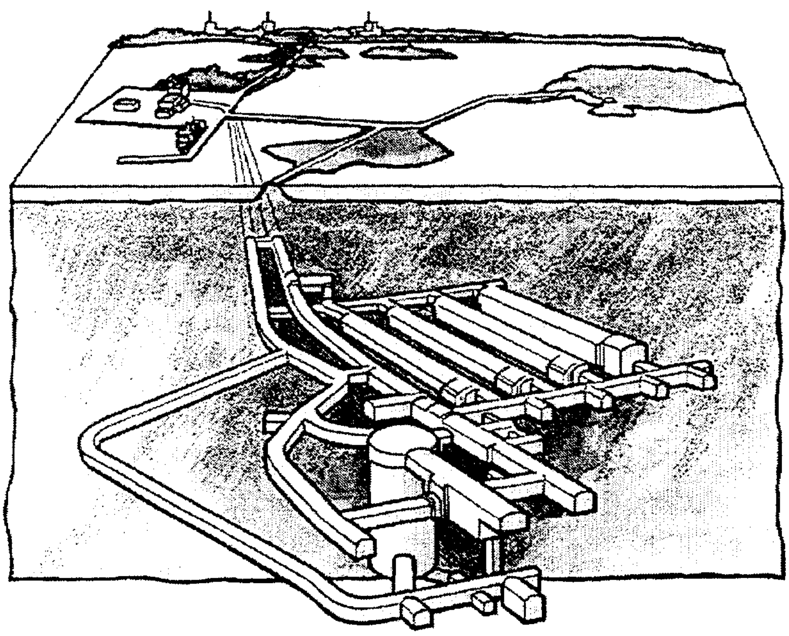

*FIG. 3. SFR - The Swedish final repository for radioactive waste*

## 2.2. Long-term safety

The long-term safety of the facility is ensured by the combination of good waste packages, engineered barriers and the isolation provided by the rock [2]. The only conceivable escape mechanisms for the radionuclides from the repository are by moving groundwater or by a well intruding into the repository. SFR has been built underneath the sea bottom because there the groundwater is practically stagnant as no topographical driving forces are present. Also there is no risk for a well being drilled as long as the repository is covered by sea water.

Most of the radioactivity, about 90 %, will be deposited in the silo, which has been equipped with the most extensive barrier system. These are:

- the immobilized waste, and the waste package;
- the chemical sorption on the material in the waste and in the silo;
- the concrete wall of the silo;
- the bentonite clay; and
- the rock mass.

Even in a case of moving groundwater the bentonite will act as a seal and prevent water from flowing through the waste in the silo. In the rock chambers the isolation is provided by the waste itself, the rock mass and in some cases by concrete structures around the waste.

Due to the land uplift in Sweden, about 6 mm/a in the area, the sea bottom above SFR will become dry land in 1500 to 2000 years time, and the hydraulic regime in the rock will change. At this time, however, the most important radionuclides in the waste, Cs-137 and Co-60, will have decayed and only the small amount of long-lived radionuclides like nickel-63 and plutonium will remain. The total amount of plutonium in the repository is expected to be less than 0.5 kg distributed in a large volume.

### 2.3. Operation of the SFR

Active operation of the SFR started in April 1988. So far about 23,000 $m^3$ of waste have been disposed of. The facility is operated by a staff about 15 persons, working day shift only. The operation has been subcontracted to the operators of the nearby Forsmark nuclear reactors and is closely integrated in the local organization [3].

Much of the handling in SFR is done by remote control from a central control room. Only during transport of the containers, handling of low-level waste and some closure operations will the operators be exposed to radiation from the waste. The doses to the personnel have thus been very low, on the order of a few mmanSv/year.

## 3. TRANSPORT

For the transport of spent fuel and waste, SKB has developed a sea transport system. It consists of a special ship, the M/S Sigyn, transport containers and terminal vehicles. Spent fuel is transported from the nuclear power plants to CLAB, and low- and intermediate level waste to SFR [4]. The M/S Sigyn is a roll-on/roll-off ship which makes the waste handling extremely expedient (Fig. 4). She only stays in the harbours at the power plants for a few hours for unloading and loading.

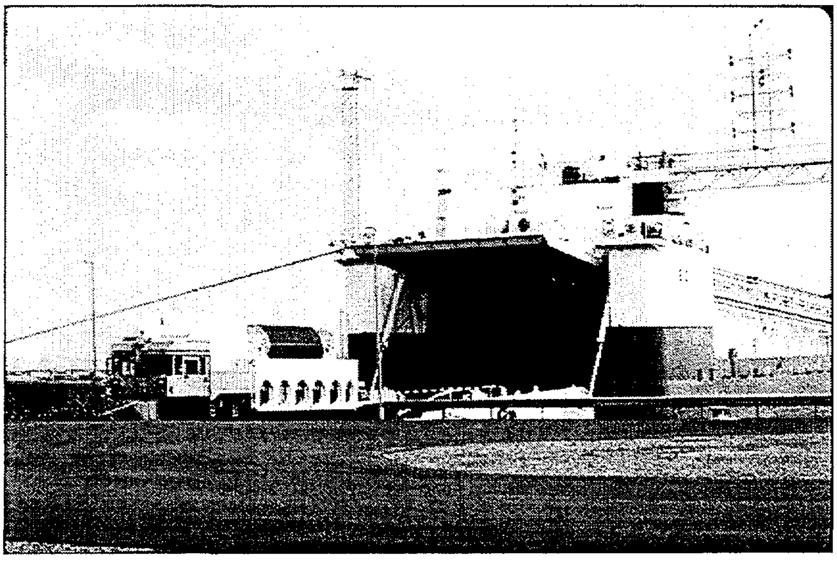

*FIG. 4. A transport cask is being loaded on board M/S Sigyn*

The low level waste is transported in standard freight containers. For the intermediate level waste that requires shielding a set of large transport containers have been developed. Each container weighs with load about 120 tonnes and can take 20 - 25 $m^3$ waste. The container has thick shielding walls and can thus take waste packages with a high dose rate. The strongest container used at present can take packages with a dose rate of about 70 mSv/h. By using these shielding containers the waste can be concentrated and the volumes can be lowered. Also a new type B-container that can take packages with even higher dose rates has been developed.

The spent fuel is transported in standardised transport casks of the French TN17/2 design. SKB has a fleet of 10 transport casks that are used for the transports from the nuclear power plants to CLAB. In total some 2,800 tonnes have been shipped to CLAB, corresponding to 970 fuel cask movements.

The performance of the transport system has been excellent with very low doses to the personnel, corresponding to the background radiation. Some factors that contribute to these low doses are the fast and efficient handling and lashing operations for the cask on the transport vehicle and on the ship.

## 4. CENTRAL INTERIM STORAGE FACILITY (CLAB)

CLAB, the central interim storage facility for spent nuclear fuel is located close to the Oskarshamn nuclear power plant on the Swedish east coast. In CLAB the spent fuel is stored in water filled pools in a similar way as at the nuclear power plants (Fig. 5).

CLAB comprises two parts, one above ground and one under ground. The main building above ground is the receiving building, where the transport casks are received, prepared and unloaded. The unloading is performed in pools under water. The actual storage section is located below ground in a rock cavern, the roof of which is 25-30 metres below the surface [5]. Interconnected with the receiving building are buildings for auxiliary systems, e.g. for water-cooling and purification, and for the electric power and control system.

The storage section, underground, consists of 4 storage pools in a 120 metres long rock cavern. Each storage pool can hold about 1,200 tonnes of fuel in canisters that serve as storage racks. A fifth pool in the cavern is used for transfer of canisters and as a reserve in case of problems with a storage pool. The storage canisters are transported down to the rock cavern in a fuel elevator with a water filled cage.

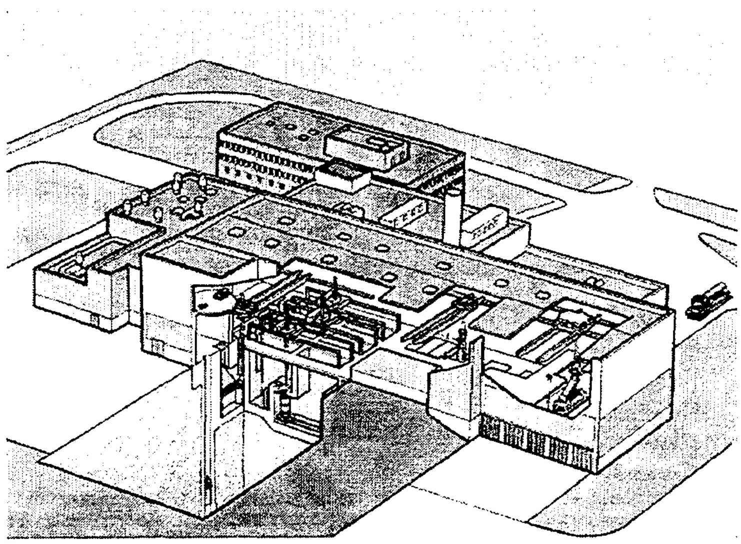

*FIG. 5. CLAB, Central interim storage facility for spent nuclear fuel*

CLAB was put in operation in 1985, and has thus now almost thirteen years of operation. In total some 2,800 tonnes of fuel have been received, and in addition about 80 transport casks with activated core components, e.g. control rods. The performance of the plant has been excellent and due to improvements, the operating costs have successively been reduced with about 40 % since the start in 1985.

The present capacity of CLAB is about 5,000 tonnes of uranium. This covers the needs until around 2005. From the Swedish nuclear programme, a total of almost 8,000 tonnes are expected to be generated. To accommodate this, CLAB will be extended by building storage pools in a new rock cavern parallel to the existing one. This was foreseen already during the construction of the facility and certain preparations for the extension was made to make it possible to construct a new rock cavern with the existing storage pools filled. The first blasting work for the extension is planned to take place in October 1998. The new pools will be in operation by the year 2004 according to current plans.

In CLAB the spent fuel is planned to be stored for 30-40 years. The fuel will then be encapsulated before being sent to final deep disposal. The encapsulation facility is planned to be built adjacent to CLAB as a direct extension of the facility.

## 5. DISPOSAL OF SPENT FUEL

In the Swedish system the spent nuclear fuel is planned to be disposed of directly as such. The safety of the repository will be achieved by the application of the following three principles:

- **Level 1 - Isolation.**
  Isolation enables the radionuclides to decay without coming into contact with man and his environment.
- **Level 2 - Retardation and retention.**
  If the isolation is broken, the quantity of radionuclides that can be leached and reach the biosphere is limited by:
  very slow dissolution of the spent fuel;
  sorption and very slow transport of radionuclides in the near field;
  sorption and slow transport of radionuclides in the bedrock.
- **Level 3 - Recipient conditions.**

The transport pathways are controlled to a great extent by the conditions where the deep groundwater first reaches the biosphere (dilution, water use etc.). The safety functions at level 1 and 2 are the most important. They are achieved by means of requirements on the properties and performance of both engineered and natural barriers and on the design of the repository.

To isolate the fuel it will be encapsulated in a canister with good mechanical strength and long-term resistance against corrosion [6]. The conceptual design adopted is a copper canister with a steel insert (Fig. 6). The copper provides a good corrosion resistance in the geochemical environment foreseen in a deep repository in Sweden. The steel insert provides the mechanical strength needed. The function of the canister is to provide adequate enclosure and radiation shielding of the fuel during handling before and during disposal and then provide an absolute barrier against radioactivity release for a considerable time after disposal. The isolation time needed cannot be universally defined. It will depend on the properties of the other barriers, but should at least be 1000 years to provide isolation for the period when the temperature is substantially elevated and when the most toxic fission products Cs-137 and Sr-90 still remain.

Each canister contains about 2 tonnes of spent fuel. The canisters are placed in deposition holes drilled from the floors of tunnels at about 500 m depth in the crystalline, granitic bedrock. Each canister is surrounded by blocks of compressed bentonite. When the bentonite absorbs water from the surrounding bedrock it will exert an intense swelling pressure and completely fill all void space in the near vicinity of the canister with bentonite clay. The clay barrier will contribute to the isolation by preventing or delaying dissolved corrosive species, that may exist in minor amounts in the ground water, to reach the canister. The clay will also provide some mechanical protection for the canister. The tunnels will eventually be backfilled by some material like a mixture of crushed rock and bentonite.

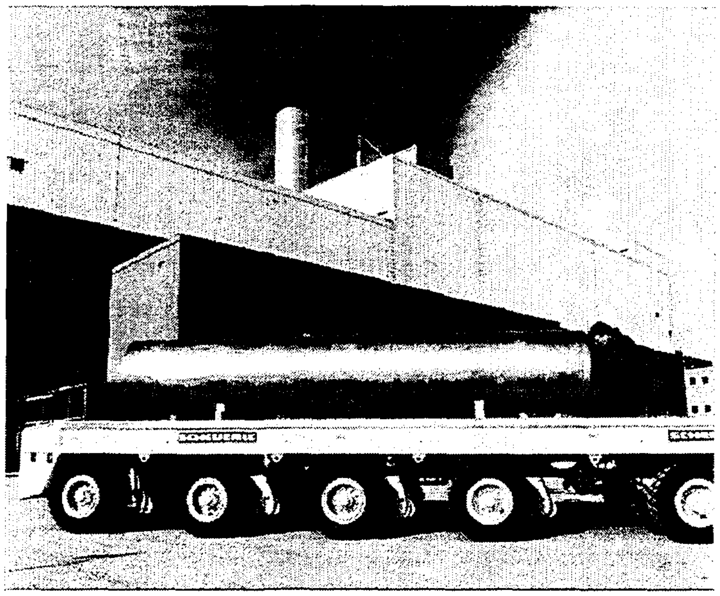

*FIG. 6. The copper canister*

## 6. ENCAPSULATION FACILITY

The encapsulation facility is planned to be built as an extension of the CLAB facility. A critical function in the encapsulation facility is the welding of the copper canister. This should be done remotely with a high accuracy, and in a way that can afterwards be controlled by non destructive testing. Different alternatives are being tested. The alternative preferred at present is electron beam welding at reduced atmospheric pressure. The welding technology has recently been demonstrated in full scale. Further development work will be performed to industrialise the process and a welding laboratory at Oskarshamn will be in operation this autumn.

The fabrication of copper canisters of the size needed is by no means an industrially available technology. Full scale canisters have, however, recently been manufactured and further tests are going on.

## 7. DEEP REPOSITORY

The final step in the spent fuel management chain is the deep disposal of the encapsulated fuel. The disposal will be made at about 500 metres depth. The canisters are deposited in holes drilled from the floors of drifts at a centre to centre distance of about 6 metres. In the holes the canisters are surrounded by highly compacted bentonite (Fig. 7).

The siting of a deep repository is politically sensitive in Sweden as in many other countries. The prime criterion for siting is that the safety requirements can be fulfilled. This puts requirements on the chemical environment, the mechanical stability and the transport conditions in the rock, as well on the risks for future intrusion into the repository. These requirements are expected to be fulfilled at many sites. Other factors such as transports, infrastructure, employment situation and political aspects will also be important for the siting.

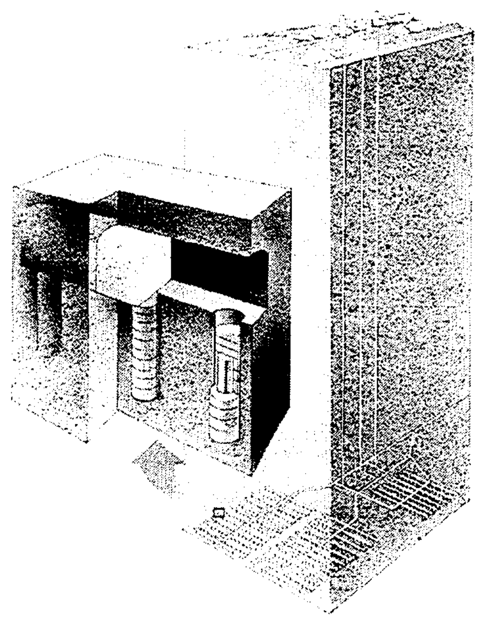

*FIG. 7. Deep repository for spent fuel*

To facilitate the siting and to clarify that the decision process for siting a repository is a stepwise procedure SKB is planning to start the disposal by first building the deep repository for deposition of a limited amount of spent fuel (about 400 canisters). Then the results will be evaluated before a decision is made whether or not to expand the facility to accommodate all the waste. This plan also makes it possible to consider whether the deposited fuel should be retrieved for alternative treatment. During the first phase it will be possible to demonstrate the siting, licensing, design and construction, handling of the canisters and operation of the facility. The long-term safety of the facility cannot, however, be demonstrated. This must always be based on a technical-scientific assessment.

The deposition could start at the earliest the year 2010. Since a few years SKB has started the work to site the repository. The first step includes general overview studies and feasibility studies in municipalities that show an interest for co-operating with SKB.

A repository is an unfamiliar facility for most people and thus raises a lot of anxiety among many people. One important aspect of the feasibility studies is thus to explain what a repository would mean for a local community. Both positive aspects such as employment opportunities, improvements of infrastructure etc. and negative aspects such as increased traffic, possible influence on tourism, and the risks of the facility itself are covered in these studies. The purpose is to provide a wider base for decision both for SKB and for the Municipality. Feasibility studies are in progress in three communities, all these have other nuclear activities. Feasibility studies in two communities have been abandoned after negative outcome of referendums in 1995 and 1997. All studies are based on a voluntary approach. In total, feasibility studies are planned to be made in at least five municipalities. Two of these will then be chosen for geological site investigations before one is chosen as the repository site. Before the final decision is made to use this site as a repository a detailed site investigation including a tunnel to repository depth will be made.

## 8. RESEARCH AND DEVELOPMENT

In order to prepare for the siting and construction of a deep repository SKB has built the Äspö Hard Rock Laboratory (HRL). The planning of this facility started some 10 years ago in 1986. The work at the laboratory has proceeded in three stages - planning and site investigations, construction and operation. The first two stages have now been completed and the operational stage has started. A basic objective in the planning of the laboratory was to create a facility for research and development in a realistic and unperturbed environment at a depth planned for the future repository.

The Äspö HRL is designed to meet the requirements on R&D. The underground construction starts with a tunnel from the site of the Oskarshamn nuclear power plant heading north down to about 220 m depth under the island of Äspö. The tunnel then goes down in a spiral with some 150 m radius down to 450 m depth. The total length of the tunnel is about 3,600 m. The last 400 m were excavated by a Tunnel Boring Machine (TBM) as opposed to the first part, that was drilled and blasted. The cross section of the tunnel is about 25 $m^2$.

Overall objectives for the research conducted at Äspö are to:

- increase the scientific understanding of the safety features and function of a repository;
- develop and test technology that will simplify the disposal concept and decrease costs at retained quality and safety;
- demonstrate the technology that will be used for disposal of spent fuel and other long-lived wastes.

## 9. INTERACTION WITH THE PUBLIC

The radioactive waste management is not only a technical issue, but has over the years also been a very controversial political issue. In order to proceed with the remaining facilities and notably the deep disposal facility for spent fuel and other long-lived waste it is necessary to gain society's confidence in the methods developed. The difference in opinions of the scientists and of the general public about the safety of radioactive waste disposal has been shown to be very great. It is therefore important to spread knowledge within the Swedish society about radioactive waste and its dangers, about the research being done and about the solutions which have been found.

Open and factual information is a prerequisite for the democratic decision-making process. SKB is contributing to this by a broad distribution of publications and other material, study visit to SKB facilities and various types of exhibitions. Most successful has the mobile exhibitions been that has been arranged on board the M/S Sigyn and with a separate trailer. In total, half a million persons have so far visited the exhibitions. The purpose of the exhibitions is to inform about the waste management, but also, not least, to listen to the worries of the people. The information trailer is also used in SKBs school programme.

In the actual siting work also extensive contacts are taken with the affected population. Information meetings are held, study groups are organized and visits to the existing waste management facilities. During the preparation of the Environmental Impact Assessment an extensive consultation of the public is planned in accordance with the intentions of the Swedish legislation. This work is in progress for the encapsulation plant.

## 10. CONCLUDING REMARKS

During the last decade a complete system for the management of all radioactive residues from nuclear power production has been developed in Sweden. With the existing systems and facilities the spent fuel and other radioactive waste can be handled safely for a very long time period. The experiences from the operation of the transport system, the interim storage facility for spent fuel and the final repository for short-lived low- and intermediate level waste have been very good.

In parallel an extensive R&D effort has been undertaken to develop a concept for the safe disposal of spent fuel and other long-lived waste in the Swedish bedrock. This work is now going over into an implementation and demonstration phase, with the design of an encapsulation facility and the first phase of a deep disposal facility. In all, this work the political and public acceptance aspects will have an important role. The stepwise implementation is a key element. It must be stressed that no step should really be irrevocable - it should always be possible to step back and reconsider and even take another route.

It is the responsibility of our generation - which has benefited from nuclear energy - to provide the facilities to take permanent care of the radioactive residues. It will be up to the following generations to decide on how to use, extend or change the system we have provided. In this way we can take our responsibility without depriving future generations of their possibilities to take their own actions.

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# STATUS OF SPENT FUEL STORAGE FACILITIES IN SWITZERLAND

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**Abstract**

Planning of a dry spent fuel storage facility in Switzerland started already 15 years ago. The first site considered for a central interim storage facility was the cavern of the decommissioned pilot nuclear plant at Lucens in the French-speaking part of Switzerland. This project was terminated in the late 80ies because of lack of public acceptance. The necessary acceptance was found in the small town of Würenlingen which has hosted for many years the Swiss Reactor Research Centre. The new project consists of centralised interim storage facilities for all types of radioactive waste plus a hot cell and a conditioning and incinerating facility. It represents a so-called integrated storage solution. In 1990, the new company "ZWILAG Zwischenlager Würenlingen AG" (ZWILAG) was founded and the licensing procedures according to the Swiss Atomic law were initiated. On August 26, 1996 ZWILAG got the permit for construction of the whole facility including the operating permit for the storage facilities. End of construction and commissioning are scheduled for autumn 1999. The nuclear power station Beznau started planning a low level waste and spent fuel storage facility on its own, because in 1990 its management thought that by 1997 the first high active waste from the reprocessing facilities in France would have to be taken back. This facility at the Beznau site, called ZWIBEZ, was licensed according to a shorter procedure so its construction was finished by 1997. The two facilities for high level waste and spent fuel provide space for a total of 278 casks, which is sufficient for the waste and spent fuel of the four Swiss nuclear power stations including their life extension programme.

## 1. INTRODUCTION

About 60% of the electric power capacity in Switzerland is covered by nuclear and 40% by hydroelectric power. In terms of electricity generation the proportion is reverse, with approximately 60% of the electrical output being produced by hydroelectric stations and about 40% by the four Swiss nuclear power stations (5 reactors). The reactors have a total installed capacity of 3,077 MWe and have accumulated about 115 reactor years experience. There location of the stations can be found in Fig. 1 and they consist of:

- Beznau NPP, a twin-unit plant of 2x 365 MW commissioned in 1969/71;
- Mühleberg NPP, 355 MW commissioned in 1971;
- Gösgen NPP, 970 MW commissioned in 1979;
- Leibstadt NPP, 1,030 MW, upgraded to 1,060 MW commissioned in 1984.

The Swiss atomic law stipulates that the producers of nuclear waste be fully responsible for its treatment and interim storage as well as final disposal. Against the background of this legal requirement, the history of the planning of interim storage in Switzerland started already in the early eighties.

## 2. THE PROJECTED LUCENS INTERIM STORAGE FACILITY

Already in those days, the idea of a centralised Swiss interim storage facility was given preference over a decentralised storage solution at the reactor sites and since then Switzerland has basically adhered to the principle of centralised storage.

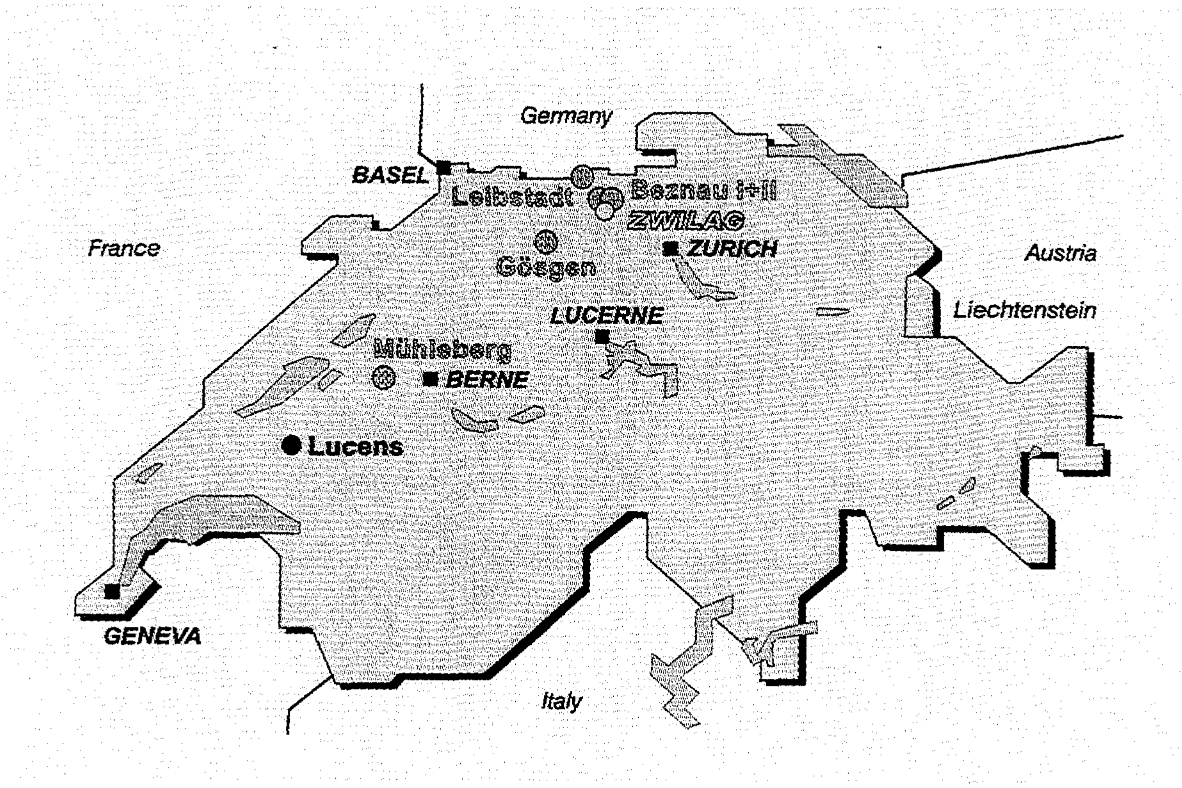


*FIG. 1. Location of the Swiss nuclear facilities*

In 1984, a pre-engineering study for a centralised Swiss interim storage facility in Lucens was presented. The town of Lucens was the site of the first Swiss test reactor which had to be decommissioned in 1971, following a core meltdown accident. In those days, the Lucens site was considered the optimum site for an interim storage facility, since public acceptance based on the familiarity with nuclear technology in the region was assumed. This first project was based on the concept of storing high-level, medium-level and low-level wastes in the same facility. The persuasive feature of the project was its compact layout, providing for cask storage of high-level wastes with heat removal by natural convection and for drum storage of the other waste types.

At the end of the eighties, however, it had to be recognised that, due to lack of public acceptance, the project could never be realised in the French-speaking part of Switzerland where no nuclear power plants are in service. The Chernobyl accident contributed further to the opposition to nuclear energy. This rejection was a severe setback to the project into which quite a great deal of effort, time and money had been invested.

The issue of centralised or decentralised interim storage was inevitably put again on the agenda by the search for a new site, all the more since for some plants the start of taking back high-level wastes from reprocessing was scheduled already for the mid-nineties. Optimisation of operational aspects, concentration of long-term interim storage at a single site and the pressure of the federal authorities were then the reasons for standing by the concept of centralised interim storage option originally pursued with the Lucens project.

A new possible site was found in the small town of Würenlingen, located in the valley of the river Aare, where the Paul Scherrer Institute (PSI) – in those days Swiss Reactor Research Centre – had done nuclear research already for decades. Compared to the rest of Switzerland, the acceptance of a nuclear interim storage facility by people and municipal authorities in the region seemed to be relatively high, due to the fact that, with the PSI and the two nuclear power stations at the Beznau and Leibstadt sites, a certain dependency on nuclear technology had emerged over the years.

## 3. THE BEZNAU-BASED ZWIBEZ INTERIM STORAGE FACILITY

As already mentioned, the failure of the Lucens project entailed a lot of trouble for the operators, because of the imminent return of radioactive wastes. In particular the management of the Beznau NPP had to assume that already in 1994 the first high-level wastes from reprocessing would be returned from reprocessing in La Hague.

Therefore, prior to the founding of a shareholder company for interim storage by the four nuclear power station operating companies, Baden-based Nordostschweizerische Kraftwerke decided to have a separate interim storage facility for high-, medium- and low-level wastes planned at the Beznau power plant site and to initiate the nuclear licensing procedure. Since this interim storage facility is located within the site boundaries of the nuclear power station, no time-consuming general licensing procedure had to be passed through, promising a considerably shorter licensing procedure.

In 1989, engineering of the ZWIBEZ facility was taken up by the Consortium NOK Engineering (Baden) and STEAG Kernenergie (Essen), adopting for the high-level waste storage facility the storage principles of the Lucens project, in particular heat removal by natural convection, since STEAG had already been involved in drawing up the Lucens project. Construction and operating permits for the ZWIBEZ were already obtained by 1991. In the same year, construction work on the low-level waste storage building began which was commissioned in 1993. At the same time, the decision was taken to refrain from building the ZWIBEZ medium-level storage building and to store all medium-level wastes in the Würenlingen storage facility to be built, which meanwhile had entered the general licensing procedure.

A special feature of ZWIBEZ is the storage technology used in the low-level waste storage building, where the individual drums are stored in steel-lattice storage containers and then stacked up in stacks of 16 m height. This storage system developed by NOK/STEAG – designed to resist a 1000-year earthquake and remain stable even under such great horizontal and vertical impact – permits highly efficient storage and is recommendable for drum storage (Fig. 2). The same type of system has been adopted for storage of low-level and medium-level wastes in the Würenlingen interim storage facility.

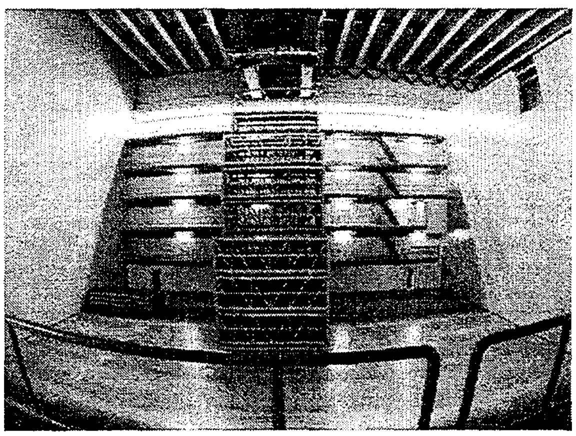

*FIG. 2. Beznau low-level waste storage building*
*Stack storage of drums in steel-lattice containers*

The high-level storage building has been designed to provide space for a total of 48 casks for high-level wastes and fuel elements. Moreover, this building provides space for the interim storage of replaced steam generators of the Beznau 2 nuclear power unit which are to be prepared in the storage facility for disposal. Construction of the high-level waste storage building commenced in 1996 and was completed in 1998.

On taking the decision in 1991 to build the facility, nobody could know that the date of return of casks with high-level waste from reprocessing which had originally been foreseen for 1994 has meanwhile been postponed to the year 2000. Obviously the return dates of wastes from reprocessing are rather difficult to forecast.

## 4. THE WÜRENLINGEN-BASED ZWILAG INTERIM STORAGE FACILITY

In January 1990, "ZWILAG Zwischenlager Würenlingen AG" (ZWILAG Interim Storage Co Ltd, Würenlingen) was established. The object of the company is the interim storage of the radioactive wastes of all categories arising from all Swiss nuclear power stations and to treat and condition low-level waste from the whole country. Shareholders of the company are:

| | |
|---|---|
| • Leibstadt nuclear power station | 33,8% |
| • Gösgen nuclear power station | 31,2% |
| • Nordostschweizerische Kraftwerke AG (Beznau NPP) | 24,3% |
| • Bernische Kraftwerke Beteiligungsgesellschaft AG | 10,7% |

ZWILAG is the first nuclear project in Switzerland licensed under the new Atomic Energy Act of 1978 (see Fig. 3 for a model of the facility). The legal procedures provide for a general licensing procedure, a construction licensing procedure and an operation licensing procedure for specific sections (conditioning, incineration). The nuclear licensing procedure described hereinafter took 6 years (from 1991 – 1996, see Fig. 4 for a detailed time schedule).

1 Conditioning plant
2 Incineration and melting plant
3 High-level storage building
4 Medium-level storage building
5 Hot cell
6 Low-level storage building
7 Federal interim storage building
8 River Aare

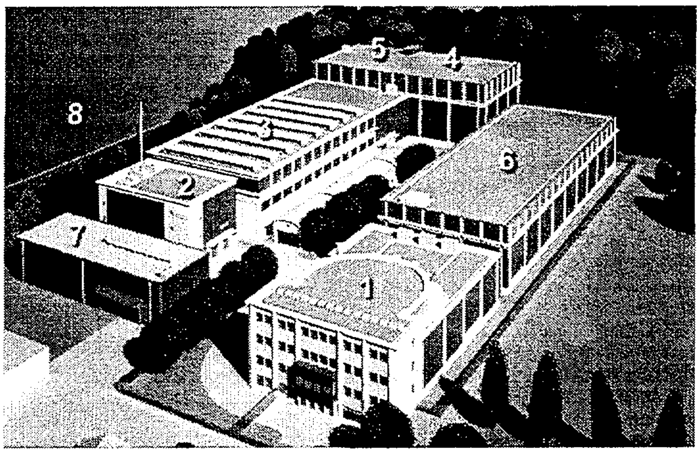


*FIG. 3. Model of ZWILAG*

In contrast to the first project in Lucens, which had been projected as an interim storage facility for high-, medium- and low-level wastes, the ZWILAG facility includes the following processes:

1. Storage of high-level wastes and spent fuel in transport and storage casks, space for a total of 200 casks;
2. Storage of medium-level wastes in drums, based on the earthquake-proof storage technology developed by NOK/STEAG using steel-lattice containers as in the above-mentioned ZWIBEZ facility;
3. Storage of low-level waste in drums, also based on the steel-lattice container storage technology;
4. Handling of radioactive material and fuel elements in a hot cell;
5. Conditioning of low-level wastes in a conditioning plant;
6. Treatment of radioactive wastes in an incinerating and melting furnace.

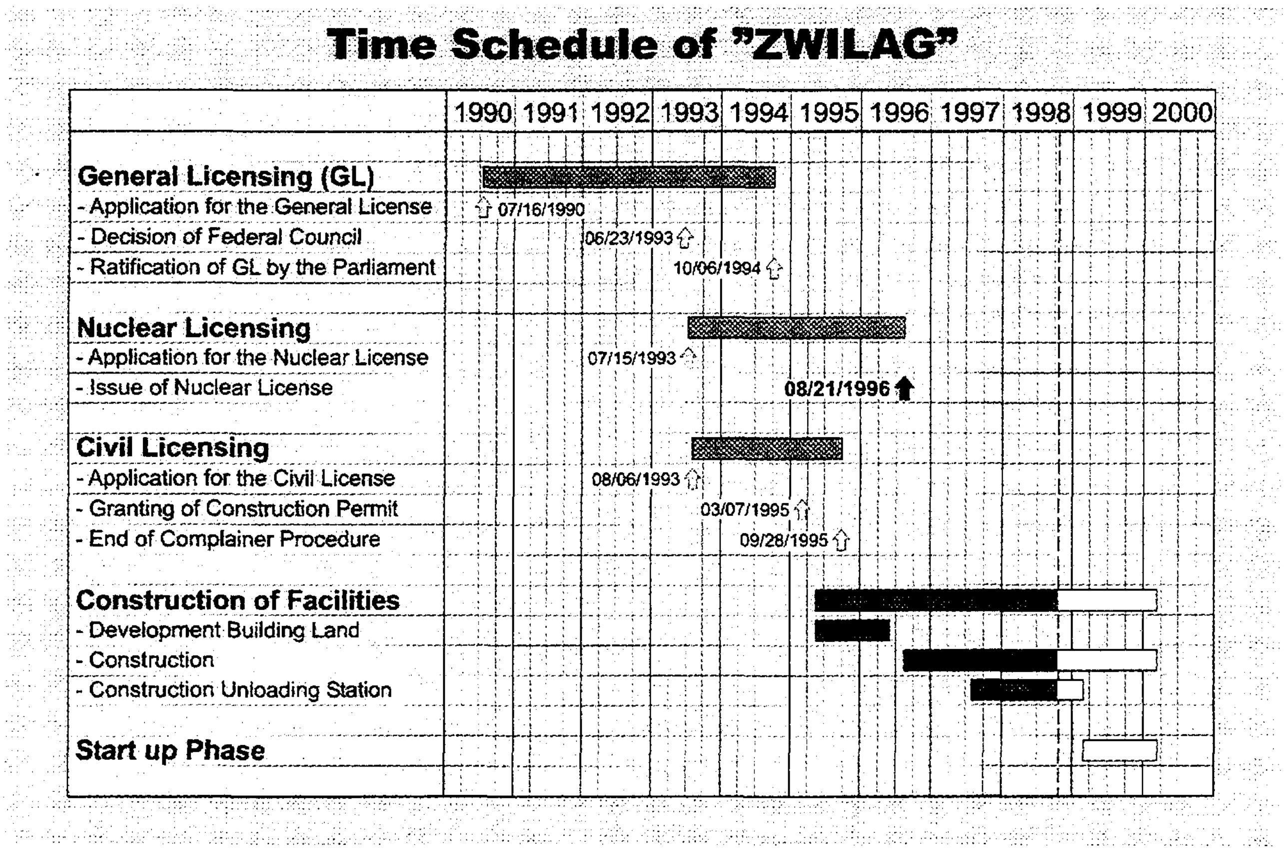


*FIG. 4. Time schedule of ZWILAG*

Construction work on the overall plant started in August 1996. Step-by-step, commissioning of the overall plant is scheduled for fall 1999. With an implementation time of about 38 months and a capital expenditure of approximately 320 million Swiss francs, excluding capital costs and costs of owner-rendered services, an average of 8 million francs per month were turned over, a figure that underlines the great performance by the enterprises involved. Below, specifically the storage facility for high-level wastes is described:

Casks loaded with high-level waste arrive mainly by rail transport via a transfer station at a distance of approx. 1.5 km from the storage facility. In the transfer station, similar to Gorleben, Germany, the casks are transferred onto special transport trucks and then transported on the road to the storage facility. In the acceptance hall the casks are removed from the truck by means of a crane. Following checking of leak-tightness and preparation for storage, the casks are transferred into the storage hall and placed in the provided positions.

The storage hall of ZWILAG has a dimension of 68 m x 41 m x 18 m and is designed to provide space for 200 casks of different types. It is based on the principle of heat removal by natural convection. With ZWILAG's 200 cask positions and the above-mentioned 48 positions in the ZWIBEZ facility of the Beznau nuclear power station, storage space for a total of 248 is available in Switzerland. Thus, there is sufficient storage capacity available in Switzerland to discontinue reprocessing if required, and to accommodate all wastes arising from the nuclear power stations throughout the planned operating period.

## 6. ZWILAG DESIGN FEATURES

Compared to the centralised interim storage facilities in the Federal Republic of Germany, the ZWILAG facility includes some specific design features:

- Acceptance area and hot cell;
- Design against external impact;
- Security concept.

The storage building for irradiated fuel elements, the storage building for medium-level wastes and the hot cell for handling of fuel elements and radioactive waste have a **common acceptance area**. In this area, transport trucks are unloaded and transport casks transferred to the transport systems of the specific storage facility. The acceptance area includes work stations for preparing the transport casks for unloading or storage. The specific facilities are functionally independent, except for the common acceptance area.

The integrated **hot cell** allows:

- Reloading of fuel elements from transport into storage casks;
- Reloading of fuel elements for required repair work on transport or storage casks;
- Inspection of fuel elements and HAW.

Thus, with regard to handling possibilities, the ZWILAG facility is independent from the further operation of the nuclear power stations after their decommissioning. The hot cell enables economic optimisation of cask management with regard to fuel element storage, particularly by reloading fuel elements into economically more efficient storage casks with higher capacities. Additional equipment provided in the hot cell permits basically the treatment of medium-level and high-level wastes.

All buildings and all safety-relevant components and systems have been given an **earthquake-proof design**. The other aspects of safety design of the specific facilities against external impact vary depending on the different storage technologies used:

- In case of the cask storage facility, protection against airplane crashes is provided by the storage casks. The required proof has to be rendered for each specific cask type.
- In case of the hot cell and the MAW storage facility, protection against airplane crash is provided by the building structures. Design against airplane crash does also cover the effects of other types of external impact.

**Security issues** are seldom discussed in symposiums although they have become a factor that considerably pushes up costs and are in some cases as decisive for the design as the nuclear requirements. It even seems that no longer the technical issues are in the focus of applied nuclear technology but rather the security issues. Further reference to the problems with the transports in Germany or the occupation of railway tracks in Switzerland will not be necessary here. It goes without saying that no confidential facts can be discussed in public, so this presentation is limited to discussing the obvious security features.

Security provisions of nuclear facilities, including the reference plant in Ahaus or the Gorleben facility are normally based on a fortified fence, difficult to surmount and arranged at a sufficient distance from the buildings of the facility. So, already the fence prevents potential agitators from accessing the area. A person climbing over the fence would be detected, set off the alarm activating numerous guards before additional external assistance would be called.

ZWILAG pursues a different concept, licensing of which was possible owing to the flexibility of the competent authorities. The ZWILAG facility will not be surrounded by a heavy fence fortified with barbed wire, water-cannons or the like. Only a simple fence of wire mesh marks the area's legal boundaries. Access to the ZWILAG will thus be similar to accessing a normal, fenced-off industrial estate. This is very interesting for the operation of the entire facility, since beyond the security-relevant area of the interim storage facility, the facility includes also the conditioning and incineration plants, operations which require optimum, i.e. free access conditions (see Fig. 5).

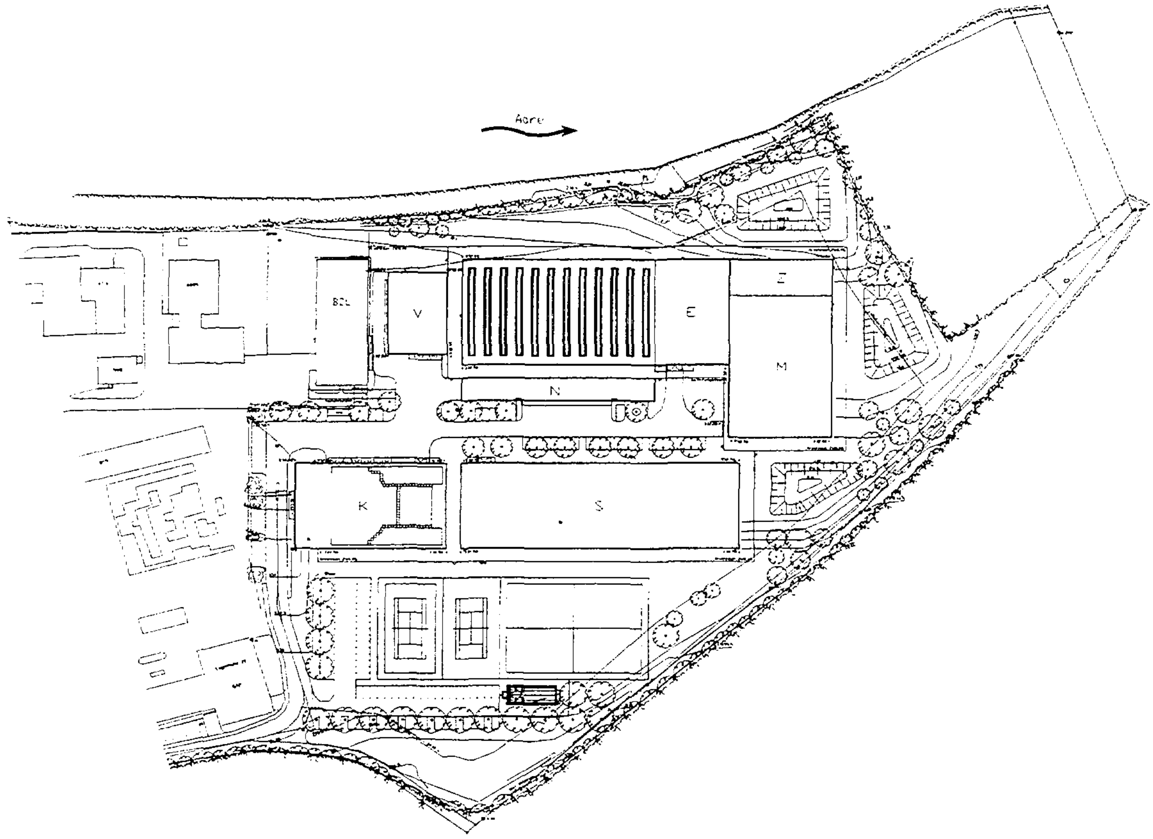

*FIG. 5. Overall view of the ZWILAG facility, with secured area around the H/E/Z/M building complex*

Around the security-relevant building complex, however, where the high-level wastes will be stored, a restricted-access zone close to the buildings will be demarcated by an additional fence. Any movement in this area is detected by electronic systems. So, what about security against intrusion of aggressors into the secured area? On the one hand, security is ensured by the concrete building structure with a suitable thickness and exhibiting the necessary resistance and by heavy grating in the openings, and, on the other hand, by additional detection systems at the building shell, providing for electronic area surveillance, which, of course entail a certain investment of capital. In addition, the acceptance area had to be designed as an access lock system.

The feasibility calculation shows that investment into the secured area immediately surrounding the building makes sense if it permits a reduction in operating costs. Security-related operating costs are mainly incurred by the guards payroll. Any investment into systems and buildings must thus bring forth a reduction in the number of guards. This was possible in ZWILAG by keeping the number of guards very small, owing to the installed electronic systems.

## 7. LICENSING PROCEDURE

The revised Swiss Atomic Energy Act divides the nuclear licensing procedure into 2 or 3 steps. In a first phase the so-called "*Rahmenbewilligung*" (general license) has to be applied for. This license specifies the site, and covers the proof that the facility is needed and the general layout. The formal application of the future owner and the expert opinion of the licensing bodies HSK and KSA are both open to the public for objections. In the case of ZWILAG, these procedures took place in 1990 and 1992. The federal government granted this first permit in June 1993. It was ratified 16 months later by the parliament.

The second phase was dedicated to the actual nuclear licenses for construction and operation. ZWILAG got the 2 licenses in one step on August 21, 1996 for the storage complex (high, medium and low level storage buildings including acceptance area, hot cell and railway transfer station). For the conditioning and the incineration plants only the construction license was issued at that date. The operation license is subject of a separate procedure now in progress. According to the latest predictions, the government will take its decision in fall 1999. ZWILAG is the first Swiss nuclear installation, which will be licensed entirely under the new law. The total delay of approximately 1-½ years as against the original time schedule can therefore at least partly be attributed to this pioneer role.

In parallel to the nuclear licensing process, a regular conventional license for construction granted by the town of Würenlingen had to be issued. It involved the town council and the general assembly of the voters and dealt mainly with the building layout, facades, electric power and water supply and sewage water treatment. The assembly of the citizens accepted this license without opposition. This was mainly due to the contract, which had been signed six years earlier between ZWILAG and the town of Würenlingen. This 35-year contract provides, among other things, for a yearly compensation payment of 1 Million Swiss francs and another 600 000 francs for the 4 neighbouring villages, paid by the ZWILAG company.

## 8. ZWILAG FINANCING

The total investment for the ZWILAG complex, including interest, taxes, fees, compensation payments etc. will amount to 500 million Swiss francs. It has been paid out of provisions of the 4 shareholder companies. These provisions have been accumulated over the last 15 years for waste storage and disposal.

The Swiss Confederation has paid an investment amount of 30 Million Swiss Francs, entitling the Confederation to use the conditioning and incineration facility without paying capital costs. Having paid for the total investment without using borrowed money, the shareholders can use all the facilities with no capital costs either.

In the case of the high-level storage building, each shareholder owns a certain number of positions for their casks. They only pay for handling and operating costs pro rata to the number of stored casks. A preliminary calculation shows that these costs break down as follows:

| | | | |
|---|---|---|---|
| 1. | Unloading and transport to the acceptance area | Sfr. 4,500.-- | per cask |
| 2. | Inspection and transport to the storage position | Sfr. 5,000.-- | per cask |
| 3. | Operating costs per year (incl. security measures and other fixed costs, assuming 100 stored casks) | Sfr. 40,000.-- | per cask |

The chosen financing mode ought to have the advantage of low capital taxes, owing to the low shareholder capital. In reality, however, the state tax authorities do not accept this relatively low-capitalised company balance sheet. ZWILAG will have to pay taxes on a fictitious shareholder investment, which still is subject to negotiation.

## 9. CONCLUSIONS

The countdown is on for spent fuel interim storage in Switzerland, with the aim to store in fall 1999 the first cask in the new storage hall. Some years after decommissioning of the Leibstadt nuclear power station around the year 2024, occupying should by completed. However, after starting to take back the wastes from reprocessing, occupation of storage positions will initially proceed rapidly.

The ZWILAG interim storage facility in the Swiss town of Würenlingen provides the link between the production of nuclear wastes in the nuclear power stations and their final disposal. Thus, also in Switzerland, interim storage of wastes arising from today's generation of power plants is ensured.

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# THE PROVISION OF SAFE STORAGE OF SPENT FUEL FROM POWER REACTORS IN UKRAINE



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**Abstract**

The nuclear energy option is the preferred source for electric power generation due to insufficient energy resources in the Ukraine. Because of the developed national nuclear power industry (the nuclear electrical capacity is 12,880 MWe and consists of 14 nuclear power units), the Ukraine faces a significant increase in the amount of spent fuel. The current general and economic situation requires the implementation of a spent fuel storage strategy that is based on interim away-from-reactor spent fuel storage, because the possible expansion of the at-reactor storage pool capacities is very limited. Implementation of the required arrangements for safe interim storage of spent fuel, allows to make the adequate selection between reprocessing and direct disposal at a later stage. At present, the licensing process of the Ukrainian interim spent fuel storage facilities is under way to ensure public and environmental safety. The review gives information about the spent fuel arisings and the measures for expansion of the spent fuel storage capacity.

## 1. PRESENT STATUS OF SPENT FUEL MANAGEMENT

The storage period for spent fuel depends highly on the individual national strategy for the nuclear fuel cycle. Taking into account the reduction of unoccupied capacity in at-reactor (AR) storage pools and the evident necessity of a balanced development of the national nuclear energy industry, the Ukraine is planning to build away-from-reactor (AFR) interim spent fuel storage facilities (ISFSFs) for storage of spent fuel for a period of more than 50 years. This option of intermediate storage will provide sufficient time for thorough considerations of the back-end of fuel cycle strategy at some point in time in the future, keeping open all alternative options.

Today, the amount of spent fuel from the Ukrainian Power Reactors stored in the Ukraine is about 3,393 t U (as of January 1998). This includes:

- about 2,051 t U of spent fuel from RBMK-1000 (i.e. a Water Channel Reactor) placed both in AR storage pools and in the AFR wet storage facility SFSF-1 on the Chernobyl NPP site;
- about 1,260 t U of spent fuel from WWER-1000 (i.e. a PWR type reactor) placed in AR storage pools of 11 units of WWER-1000;
- about 82 t U of spent fuel from WWER-440 placed in AR storage pools of two units of the Rovno NPP.

Approximately 805 t of spent fuel (SF) from Ukrainian WWER reactors were transported to reprocessing plants in Russia, including:

- about 153 t U from WWER-440;
- about 652 t U from WWER-1000.

In addition to the above stated, it is necessary to note that:

- for the present time, it is proposed not to reprocess RBMK spent fuel because it is not economical;
- the ecological and economic outlook for reprocessing or direct final disposal of WWER spent fuel (into a geological repository) are kept under consideration;
- as before, now only WWER-440 spent fuel remains the real subject of reprocessing.

## 2. DEFINING FACTORS OF SPENT FUEL ACCUMULATION DYNAMICS

The RBMK spent fuel arisings will increase to a level of about 2,500 t U (21,800 fuel assemblies, FAs), because of shutdown of the Chernobyl NPP in the year 2000.

During the next decade, the annual spent fuel arisings will be essentially characterized by the discharge of the WWER-1000 spent fuel as a result of commissioning of the new units Rovno-4 and Khmelnitski-2 in 1999/2000, as well as Khmelnitski-3 and Khmelnitski-4 in 2004. The arisings will be somewhat reduced due to the permanent improvement of fuel use and the use of the advanced fuel. The use of WWER-1000 fuel assemblies (FAs) for the fourth-year in the reactor, which has been allowed by the Ukrainian regulatory body since 1997, increases the fuel efficiency and is the transition to the commercial implementation of the four-year WWER-1000 fuel cycle. The four-year WWER-1000 fuel cycle can decrease the annual spent fuel arisings by 20% by the year 2004.

## 3. IMPLEMENTED AND SCHEDULED ACTIONS ON SAFE STORAGE OF SPENT FUEL FROM POWER REACTORS

### 3.1. Reracking of spent fuel ponds for WWER

Reracking of the AR spent fuel pools was done for all WWER-1000 power units in order to expand the storage capacity, except for the Zaporozhe units 3 to 5. The implementation of these measures has permitted to expand the storage capacity of the AR spent fuel pools of the WWER-1000 units by 30-70% compared to the initial design capacity. (See Figure 1.) Now the Rovno NPP plans the reracking of the AR spent fuel pools of the units 1&2

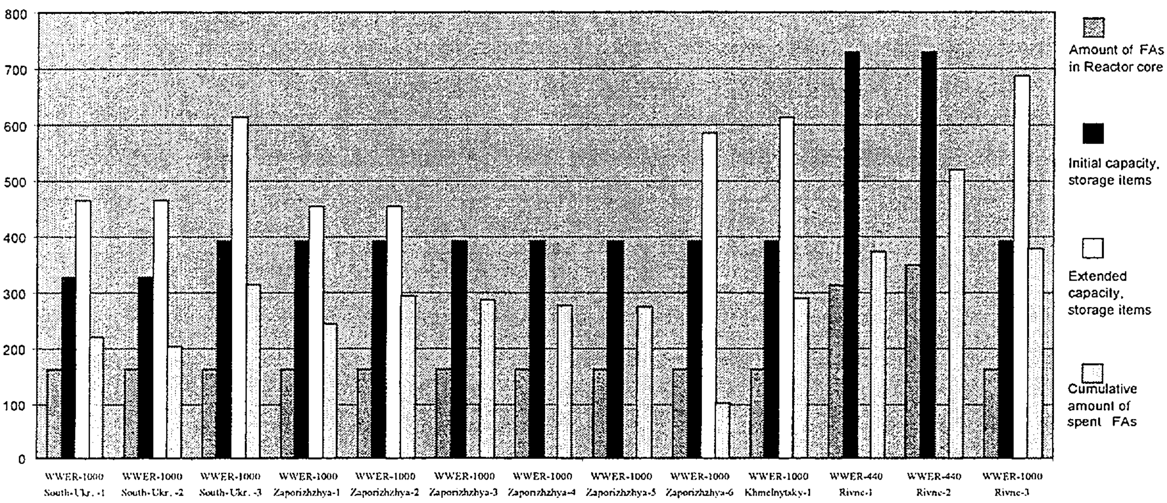


*FIG. 1. Reracking of spent fuel ponds for WWER units*

### 3.2. Construction of intermediate spent fuel storage facilities

Storage facilities (for a storage period of more than 50 years) will permit the Ukraine to gain sufficient time for thorough considerations of the back-end of the nuclear fuel cycle strategy and decrease its dependence on foreign services. According to the approved «National Energy Programme of Ukraine to 2010» [1], the commissioning of a centralized interim SF storage facility (CISFSF) is the best technical and economic option for interim storage of WWER SF. This decision took into account the large number of the NPPs sites, visible advantages for decommissioning from the point of view of NPP safety, as well as decreasing operation and maintenance costs. Nevertheless, due to the urgent necessity of storage capacity, the construction of dry WWER interim SF storage facilities (ISFSFs) are planned at the NPPs sites in using dual purpose systems (for storage and transportation).

This approach reduces the time of putting into operation the needed interim storage facilities (because of delay of the CISFSF due to uncertainties in the site selection, the construction start and the lack of initial funds) and avoids extra charges during decommissioning. The storage capacity of the ISFSFs should be limited to capacities of 3-5 annual spent fuel discharges per operating unit. Their unified design shall be selected by a tender procedure. The construction of AFR ISFSFs is planned on each NPP site with WWERs.

*3.2.1. Interim spent fuel storage facility at the Zaporozhe NPP*

The biggest progress has been obtained from the construction of the Zaporozhe NPP (ZNPP), which has been on-going since 1994. There are 6 WWER-1000 units in operation on the site. The NPP management had undertaken active measures with respect to the construction of an ISFSF. After the assessment of proposals from different companies, the decision was made to build a storage facility on the basis of the VSC-24 casks of the US company Sierra Nuclear. Designs of some main components for this ISFSF are shown in Figures 2 to 5.

The Zaporozhe NPP signed an agreement with the engineering company Duke Engineering & Services to perform the design of the storage containers for the WWER-1000 SF and guide the design of the ISFSF. The designing for the remaining components was carried out by the Kharkov Design Institute "Energoproject", which performed functions as the general designer for ZNPP.

The ISFSF licensing, which started in 1995 [2], is under completion. According to a special permit, the construction of the storage pad has been finished. The license for manufacturing of elements important for safety has been issued. In the framework of this license, the construction of the first three concrete casks has been completed. According to requirements of the safety authorities, which were based on the results of examination of the ISFSF SAR, the ISFSF design underwent some corrections. The safety examination of the design modifications was carried out. Some questions are still unresolved. The safety examination of the operation documentation and the certification of elements important for safety (Ventilated Storage Cask, VSC, Multi-seater Storage Basket, MSB, welding system and transfer cask) is currently be done. The commissioning of the Zaporozhe ISFSF is scheduled for a one year period. The State Committee on commissioning of the Zaporozhe ISFSF was set up this year.

*3.2.2. Interim spent fuel storage facility at the Chernobyl NPP*

The irradiated fuel assemblies and reactor components are currently stored in a wet storage facility (ISFSF-1), in existing at-reactor storage pools (two pools per reactor block 1, 2 and 3), in the reactor cores of unit 1, 2 and 3. As of January 1998, there were 17,870 spent fuel assembles in ISFSF 1 and in the AR storage pools of units 1 to 3. It is estimated, that approximately 11% of the total fuel assemblies to be stored will have defects. The existing storage facility may be full by the end of 1999 based on current reactor operating schedules. The existing storage facility's design life expires in 2016.

Construction of a dry type ISFSF-2 is planned in the area of the Chernobyl NPP. The ISFSF-2 will be located within the exclusion zone surrounding the Chernobyl NPP. The NPP requires a storage facility to store approximately 25,000 RBMK fuel assemblies (approximately 3,000 t SF) and approximately 3,000 discharged absorber rods for 100 years.

The interim storage facility project is part of the Grant Agreement signed between the Ukrainian Government, the European Bank for Reconstruction and Development (EBRD) and the Chernobyl NPP. The Chernobyl NPP Project Management Unit (PMU) has been set up to assist in the development of the Invitation For Tender and to assist in the management of the contract. It is expected that the technology to be implemented will be the dry storage technology (dry well, casks, silos or vaults). The stored fuel shall be safely retrievable during the entire storage period.

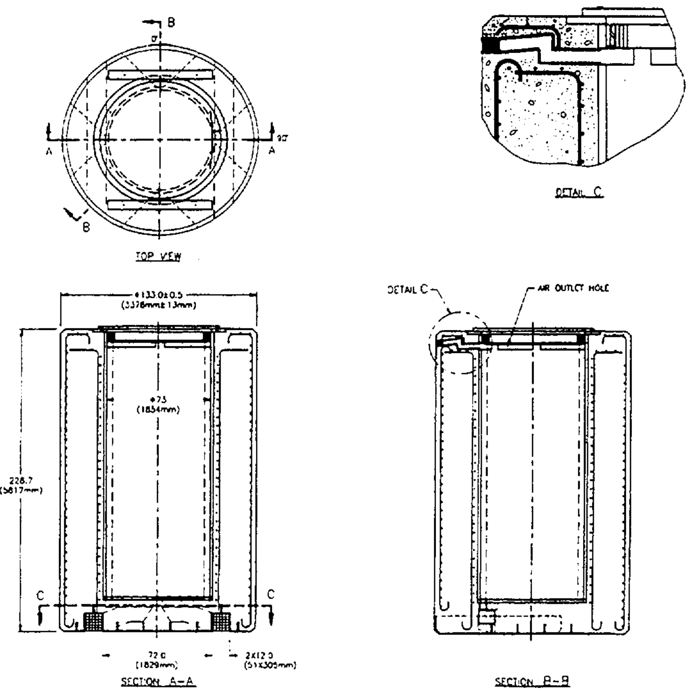


*FIG. 2. Design of Ventilated Concrete Cask*

To establish the licensing programme for the Interim Storage Facility, a special document "Licensing and Certification Programme" was prepared. This document could be considered as a kind of good practice or an agreement between the Applicant and the Regulatory Authority. This approach has to be in line with the Ukrainian legislation on one hand and has to comply with the requirement from the EBRD, under which the Grant has been provided to Ukraine (e. g. the open tender, the "turnkey" delivery), on the other hand. This practically implies the possibility of the use of Western codes and standards for the design and the construction of the facility and requires a comparison (reconciliation) of the applied codes and standards with the corresponding Ukrainian ones.

The invitation to tender was announced at the beginning of this year. The tender award is expected at the beginning of next year. Ukrainian laws, regulations, normative documents and standards define the licensing requirements for the ISFSF-2 as a nuclear facility. Other standards (national or international) that ensure equivalent or higher quality or performance than Ukrainian standards are also acceptable.

## 4. CONCLUSIONS

The present spent fuel management strategy in the Ukraine does not envisage alternative options of interim storage and subsequent reprocessing of spent fuel or of extended storage and direct disposal of spent fuel. The use of interim storage will be the primary spent fuel management option for the next ten to twenty years. It is expected, that in this period the selection of the optimum nuclear fuel cycle and the applicable spent fuel management scheme in Ukraine will be chosen. Consequently, the primary step of all decisions consists of the selection of the appropriate strategy for the national nuclear energy development.

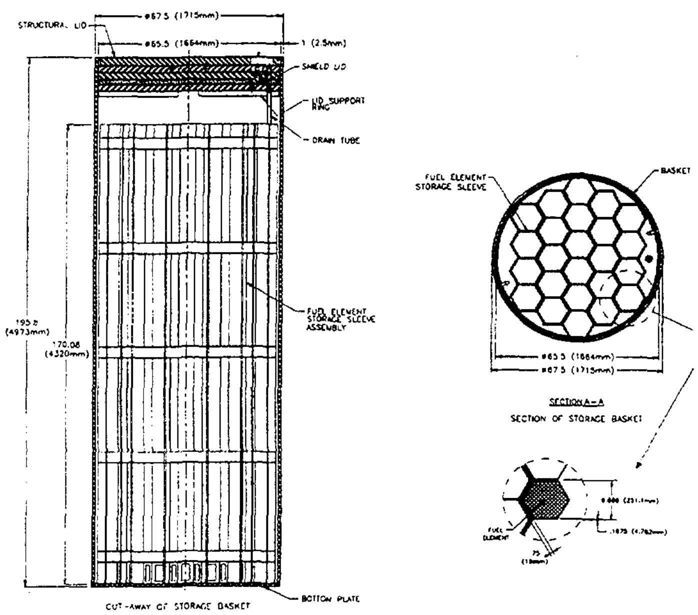


*FIG. 3. Design of Storage Basket*

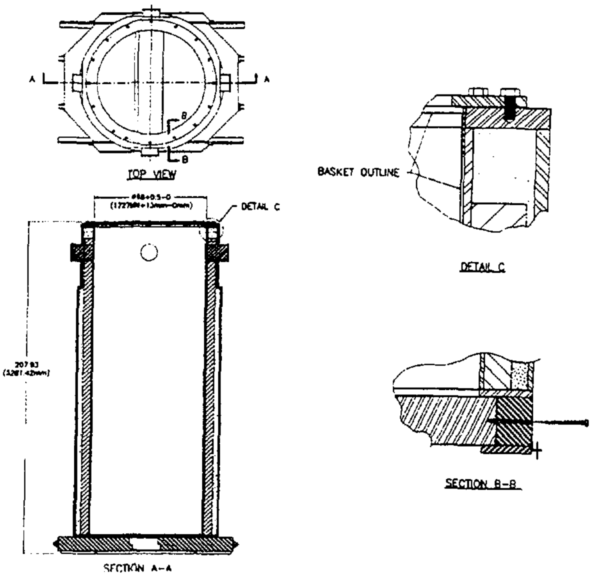


*FIG. 4. Design of Transfer Cask*

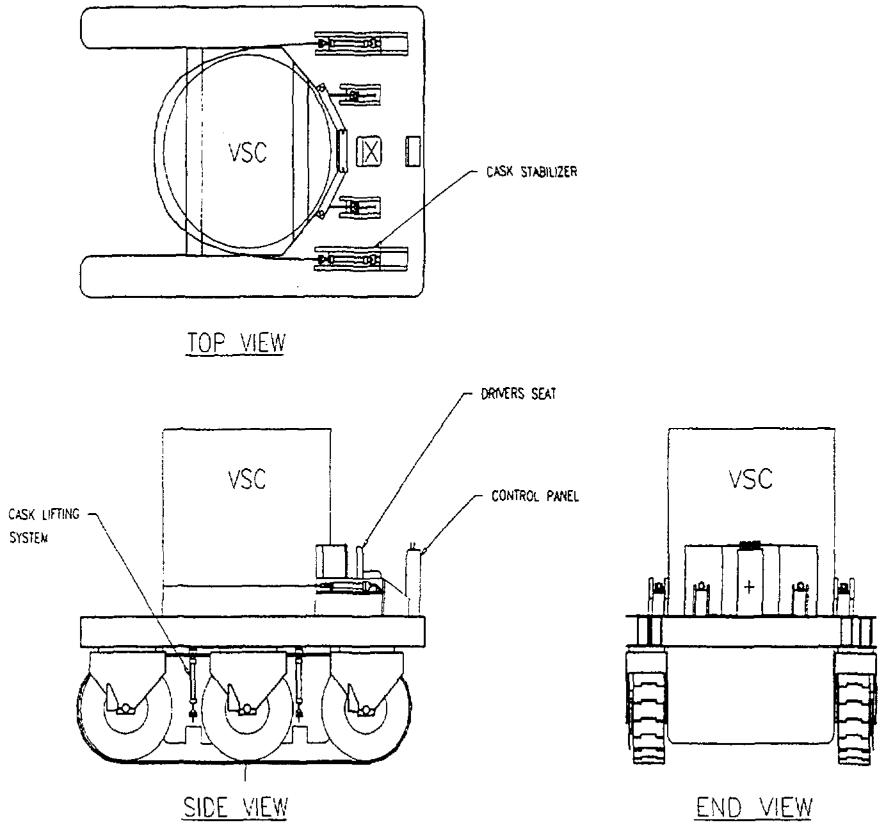


*FIG. 5. Design of Transporter*

There were some safety issues identified during the licensing process of ISFSF of Zaporozhe Nuclear Power Plant. The main issues can be formulated as follows:

- Due to the lack of a regulatory document system for nuclear facilities, such as the ISFSF, the regulation developed for NPP had to be applied. Some of these regulation requirements are of specific to NPP systems. At the same time, some of the reulations are not applicable to an ISFSF, some are maybe too strict and some of them can not be uded at all. The development of specific regulations for nuclear facilities, other than for NPPs, could resolve this problem.
- One of the main principles of nuclear safety assurance was not completely met, i.e. that "technical and organisational decisions being made to provide for safety shall be verified by gained experience or tests, appropriate researches, experience of prototypes operation". Thus, experience of WWER SF storage within gaseous media under the proposed conditions is unavailable. Both the applicant and the regulator need such experience.
- Currently, the Ukrainian legislation is under development. Some regulation and legislative documentation provisions need to be in better compliance with other ones. Both the applicant and the regulator need well-shaped, compatible and well-described nuclear legislation.
- The most important issue of the construction and commissioning of the interim dry storage facilities is the safety case (justification) for the long-term dry storage of spent fuel. This task could be solved jointly by the Ukrainian authoritative bodies and the leading institutions for research and design of nuclear fuel.

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# THE STATUS OF SPENT FUEL STORAGE IN THE UK



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**Abstract**

Nuclear generating capacity in the UK is static with no units currently under construction. There are three main nclear fuel types used in the UK, for Magnox reactors, AGRs and PWRs. All Magnox fuel will ultimately be reprocessed following a short period of interim storage. AGR fuel will either be reprocessed or long term stored in ponds. PWR fuel will be stored underwater at the reactor site for the foreseeable future, with no decision as yet made to its ultimate management route.

## 1. BACKGROUND AND GENERAL ISSUES

Nuclear generating capacity in the UK is static with no units currently under construction. The UK's nuclear generating capacity comprises some 8,400 MWe AGRs and one 1,200 MWe PWR operated by British Energy (BE), and 3,350 MWe Magnox reactors operated by British Nuclear Fuels plc (BNFL). The details of the nuclear power stations, currently in operation in the UK, are given in Table I.

TABLE I. OPERATIONAL NUCLEAR POWER STATIONS IN THE UK

| Name | Type | Net Capacity (MW) | Start of Operation | Current Accountancy Lifetime (Years) |
|---|---|---|---|---|
| Calder Hall | Magnox | 200 | 1956 | 50 |
| Chapelcross | Magnox | 200 | 1959 | 50 |
| Bradwell | Magnox | 240 | 1962 | ↑ |
| Dungeness A | Magnox | 440 | 1965 | Average of |
| Hinkley Point A | Magnox | 460 | 1965 | 37 years |
| Oldbury | Magnox | 440 | 1967 | operating |
| Sizewell A | Magnox | 420 | 1966 | lifetime |
| Wylfa | Magnox | 950 | 1971 | ↓ |
| Dungeness B | AGR | 1120 | 1983 | 25 |
| Hartlepool | AGR | 1205 | 1984 | 25 |
| Heysham 1 | AGR | 1060 | 1984 | 25 |
| Heysham 2 | AGR | 1340 | 1988 | 30 |
| Hinkley Point B | AGR | 1270 | 1976 | 35 |
| Hunterston B | AGR | 1195 | 1976 | 35 |
| Torness | AGR | 1210 | 1988 | 30 |
| Sizewell B | PWR | 1200 | 1994 | 40 |

Nuclear power in the UK represents some 18% of the total installed electrical capacity, but currently supplies some 31% of the electricity produced.

The long history of nuclear power in the UK means that there is now over 40 years operating experience with a number of different fuel types and a variety of storage systems in the UK. Spent fuel storage facilities in the UK currently consist of a mixture of at-reactor stores and large, centralised ponds associated with the reprocessing activities which take place at the Sellafield site.

The Government policy in the UK is that it is for the owners of the spent fuel to decide on the appropriate spent fuel management option based on their own commercial judgement, subject to meeting the necessary regulatory requirements.

## 2. MAGNOX FUEL

Magnox fuel elements consist of bars of natural uranium metal, approximately 1m long, which are clad in a magnesium alloy (giving rise to the name Magnox). The Magnox system was designed with a wet discharge route and interim pond storage of fuel in anticipation of early reprocessing. Magnox fuel is generally reprocessed after about 6 months storage because the cladding degrades when wet and this prevents the fuel from being stored for long periods underwater.

The spent Magnox fuel is received at the Sellafield site in the Fuel Handling Plant (FHP) which has a capacity of some 2,700 tU. During the wet storage which takes place prior to reprocessing, cladding degradation is minimised through careful control of pH of the pond storage water. Just prior to reprocessing, the fuel is fed into a decanner in FHP which remove the cladding and loads the uranium bars into magazines for transport to the Magnox reprocessing plant which is known as B205. Figure 1 shows this route schematically.

Although in principle Magnox fuel could be dry stored, the retrofitting of expensive drying facilities or modifications to station fuel discharge routes would prove uneconomic, a fact acknowledged by a UK Government select Committee in 1986, which accepted prompt reprocessing as the only practical option for dealing with Magnox fuel.

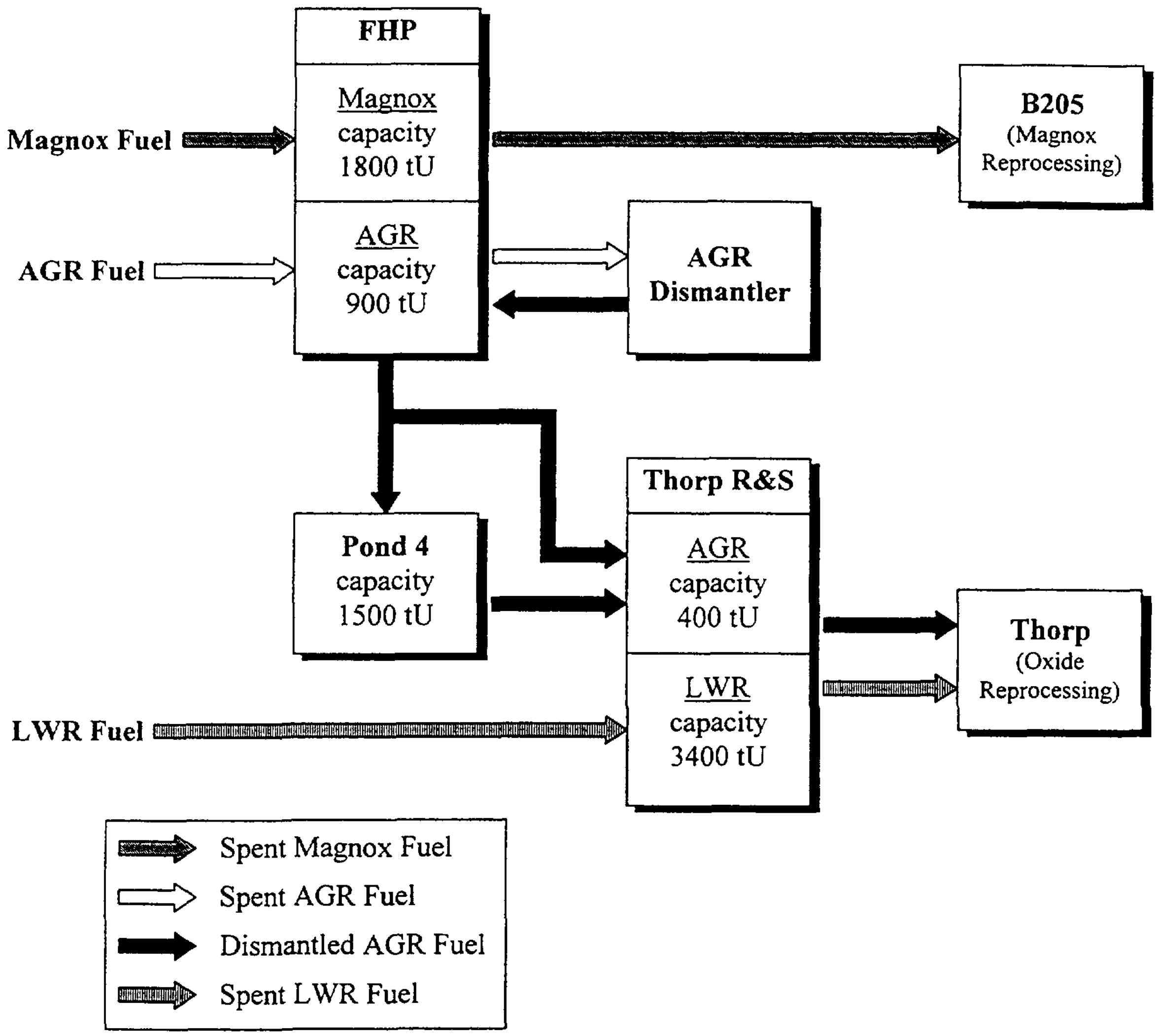


*FIG. 1. Outline of spent fuel storage and treatment at Sellafield*

Wylfa power station is the only Magnox reactor site which does not use wet storage. Fuel is discharged from the reactors into three fuel storage cells and stored dry in a Carbon Dioxide atmosphere. The total storage capacity is about 18,000 fuel elements.

However, two further air-filled stores were constructed at the Wylfa reactor site in the 1970's to act as a buffer store against any potential reprocessing throughput constraints. Fuel may be transferred to the secondary dry fuel cells by a dedicated fuelling machine after 150 days cooling in a primary dry store cell. The secondary dry store cells provide storage for over 50,000 fuel elements.

## 3. AGR FUEL

AGR fuel pins are approximately 1m long and consist of enriched $UO_2$ pellets clad in a stainless steel tube. The fuel elements consist of 36 pins arranged in a circular lattice and sheathed in a graphite sleeve. The AGR power stations have small at-reactor pond stores, as early reprocessing was envisaged during the design of the reactors, and hence all spent AGR fuel is sent to Sellafield where it is stored underwater in large centralised ponds.

The AGR stations pond capacities are detailed in Table II.

TABLE II. AGR STATION POND CAPACITIES

(t)

| Station | Nominal pond Capacity |
|---|---|
| Heysham 1 | 15 |
| Hartlepool | 15 |
| Hinkley Point B | 19 |
| Hunterston B | 19 |
| Dungeness B | 32 |
| Heysham 2 | 46 |
| Torness | 46 |

The contractual relationship between BNFL and British Energy covers the lifetime arisings of AGR fuel. It provides for a near maximum commitment to reprocessing in Thorp over the first 20 years of operation. Options for further reprocessing, following the first 20 years, or long-term storage (circa 80 years) also exist.

AGR fuel is currently received at Sellafield in the Fuel Handling Plant (FHP) in 15 compartment skips, containing 15 fuel elements, the skips are first placed into lidded storage containers. After an initial period of cooling the fuel is dismantled and the fuel pins are placed into stainless steel slotted cans. Each can holds 108 pins and are stored in 20 compartment skips, in lidded storage container. The dismantled fuel is then transferred to the AGR storage pond until it is a minimum of three years cooled after which time it can be transferred to Thorp Receipt and Storage for reprocessing (see Figure 1).

There is sufficient capacity at Sellafield to store the AGR fuel arisings which are likely to be delivered outside the window for reprocessing in the first 20 years of Thorp operation. The fuel will be wet stored in water dosed with a corrosion inhibitor at pH 11.5 Sodium Hydroxide. It is anticipated that there will be no insurmountable technical problems associated with the long term storage of AGR fuel. Fuel delivered during this period, will be stored undismantled in the 15 element skips. Dismantling of fuel and consolidation for long term storage would be considered in terms of overall cost optimisation in the future.

Depending upon the lifetime of the AGR stations up to several thousand tU of AGR fuel may be long term wet stored at Sellafield.

AGR fuel could, theoretically, also be stored in dry stores and, indeed, Scottish Nuclear (now part of BE) undertook a study of the feasibility of this option and applied for planning permission to build a dry store at their Torness site. However, following a review of the options they concluded that a mixture of reprocessing and long term storage at Sellafield provided the most cost-effective spent fuel management solution and signed lifetime arisings contracts with BNFL in 1995.

4. PWR Fuel

The UK only has one PWR, Sizewell B, and in contrast to the other UK reactors it was designed with a relatively large on-site spent fuel storage pond, capable of storing approximately 18 years spent fuel arisings. However, this has been extended by re-racking so that it will now accommodate up to 30 years spent fuel arisings and hence decisions on the ultimate fate of the spent fuel do not have to be taken for some considerable time. BE will consider in due course arrangements for further management of spent PWR fuel in the light of the prevailing commercial and regulatory environment.

5. OTHER FUEL TYPES

The UK also has small quantities of spent fuel from UK test and prototype reactors such as the SGHWR and WAGR. It is anticipated that these fuels will also ultimately be reprocessed, but in the interim they are also being stored at the Sellafield site.

6. OVERSEAS LWR FUEL

BNFL has contracts with a number of overseas utilities for the reprocessing of over 5,000 tU of spent LWR fuel, the majority of which is already in storage at Sellafield. Spent LWR fuel is received at Sellafield in the Thorp R&S facility which has a capacity of 1,000 tU and is stored in Multi-Element Bottles (MEBs) into which the fuel elements had been loaded at the reactor sites prior to transport to Sellafield (see Figure 1).

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# STATUS AND CURRENT SPENT FUEL STORAGE PRACTICES IN THE UNITED STATES

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**Abstract**

Brief discussions are presented on the history and state of spent fuel generation by utilities that comprise the United States commercial nuclear power industry, the current situation regarding the Federal government's nuclear waste policy, and evolving spent fuel storage practices. These evolving spent fuel storage practices are the result of private sector initiatives, but appear to be influenced by various external factors. The paper is not intended to provide a comprehensive appraisal of the storage initiatives being conducted by the private sector. The focus, instead, is on the Federal government's role and activities related to spent fuel management. Although the Federal government has adopted a policy calling for deep geological disposal of spent fuel, the US Congress has recently begun to consider expanding that policy to include a centralized interim storage facility. In the absence of such an expanded policy, the Department of Energy has performed some preliminary activities that would expedite development of a centralized interim storage facility, if Congress were to enact such a policy. The Department's current activities with regard to developing a centralized interim storage facility, which are consistent with the current policy, are described in the paper. The paper also describes two important technical development activities that have been conducted by the Department of Energy to support improved efficiency in spent fuel management. The Department's activities regarding development of a burnup credit methodology, and a dry transfer system are summarized.

## 1. INTRODUCTION

More than 100 commercial nuclear power reactors located on 73 sites are licensed in the United States (US). These nuclear power plants supply over 20% of the country's electric power. About two-thirds of these reactors are pressurized water reactors (PWR), and the remaining one-third are boiling water reactors (BWR). There is no expectation of new nuclear power plants being constructed in the US in the near future. These existing plants are projected to produce 87,000 metric tonnes of uranium (tU) in their design lifetimes of 40 years. This projection is generally considered an upper bound because early closures are not fully considered. However, a recent initiative to extend the service life of reactors beyond the 40-year initial design is being developed in the US. One utility, Baltimore Gas and Electric Company (BG&E), has tentative plans to pursue this alternative. Normally, US power reactors are licensed for 40-years, but under life extension, as the practice is called, BG&E would seek a 20-year license from the Nuclear Regulatory Commission (NRC) to continue operations. To be granted an extension, the licensee would have to demonstrate the adequacy of the plant's safety systems to be licensed for the additional 20-year period.

US commercial nuclear power reactors are operated using a once through fuel cycle. The policy adopted in the US for dealing with spent fuel generated under this choice of fuel cycles is disposal in a deep geological repository. Legislation passed in 1982 has set this policy [1], legislation enacted in 1987 [2] has amended the earlier legislation, and identified a specific site at Yucca Mountain in Nevada for characterization as a repository. These legislative acts are known as the Nuclear Waste Policy Act (NWPA) and the Nuclear Waste Policy Amendments Act (NWPAA). It is now estimated by the Department of Energy (DOE) that, if Yucca Mountain proves suitable, it will begin receiving spent fuel about 2010 [3]. At the present time, spent nuclear fuel is being stored at commercial nuclear power plants at various locations throughout the US. Many of the older reactors, having limited pool storage space, have opted for dry storage technologies once pool space has been fully used.

DOE, which is part of the executive branch of the US Government, is responsible for developing a deep geological repository, under the NWPA. Congress has recently entertained the idea of enacting legislation that would require DOE to develop a Centralized Interim Storage Facility (CISF) to store up to 40,000 tU of spent fuel until a permanent disposal site is available. Bills recently proposed in Congress, but not enacted into law, have identified early development of such a facility at Yucca Mountain. In the absence of legislation, DOE cannot develop such a facility, but may conduct activities related to contingency planning to prepare for such a mandate. Therefore, in preparation for the possibility of such legislation, and in light of an expected fast schedule, DOE has designed a generic CISF that is not site specific. The generic CISF, which is described in this paper, is designed for siting almost anywhere in the contiguous US. The generic CISF design is intended to allow early identification and resolution of technical and regulatory issued related to development of such a facility. The strategy is intended to expedite site-specific design, development, licensing, and construction of such a facility should legislation be enacted.

## 2. BACKGROUND

Many of the early reactors built in the US were designed with the expectation that spent fuel would be recycled. For these reactors, pools were sized to hold spent fuel for short periods awaiting shipment to a reprocessing facility. By the late 1970s, reprocessing of spent fuel was abandoned in the US.

As utility companies realized that reprocessing would not be pursued in the US, on-site storage plans for new reactors were modified. Later plants generally included sufficient pool space to handle lifetime generation of spent fuel. For older reactors with limited storage capacity in existing spent fuel pools, no reprocessing option, and no near term disposal available, alternative on-site storage options were sought.

Some of the obvious solutions to the developing storage needs of utilities were pool expansion, more efficient use of pool space, on-site dry storage, centralized off-site storage, and disposal.

Expansion of existing storage pools and building of new pools on existing sites were expensive and filled with regulatory and licensing difficulties. More efficient use of pools had limited success for a number of utilities, allowing time to implement more comprehensive solutions to satisfy their storage needs. The key to more efficient use of existing pools generally involved reracking storage regions to allow closer packing of spent fuel. To assure criticality safety in such modified arrangements of spent fuel required the use of neutron absorbers in fuel racks, and in some cases, burnup credit [4]. An interesting approach was used by Carolina Power and Light Company (CP&L).

CP&L has three nuclear facilities in North and South Carolina. The oldest, H.B. Robinson, Unit 2, began service in 1971. The second plant, Brunswick Steam Electric Plant has two units which began service in 1975 and 1977. The newest plant, Shearon Harris Nuclear Power Plant, was built for four reactors in the late 1980s. The plant was built with storage capacity for all four units, but only one is in service. The other three were not brought into service. CP&L has used this extra storage capacity at its Shearon Harris plant to store spent fuel from all three of its operating reactors. Fig. 1 shows the currently operating spent fuel sites.

The option currently being selected by many utilities anticipating constraints on storage capability is dry storage. The approach uses existing, limited capacity, pools for cooling newly discharged spent fuel. Dry storage is used for the oldest and coldest spent fuel as the reactor site's pool capacity limit is approached. A number of dry storage concepts have been pursued, including storage only vaults and casks, storage-transport (dual purpose) cask systems, and multi-purpose systems (i.e., storage, transport, and disposal). Some of the dual purpose and all of the multi-purpose systems use canisters to hold the spent fuel in casks or separate overpacks for storage, transport, or disposal.

The early dry storage systems were storage only systems. More recently, utilities have been selecting dual purpose systems to satisfy their current and anticipated storage needs. Some utilities with older storage only systems are entertaining the possibility of seeking NRC approval for transport; thereby, converting them to dual purpose systems. Fig. 2 shows the location of potential near-term spent fuel storage sites.

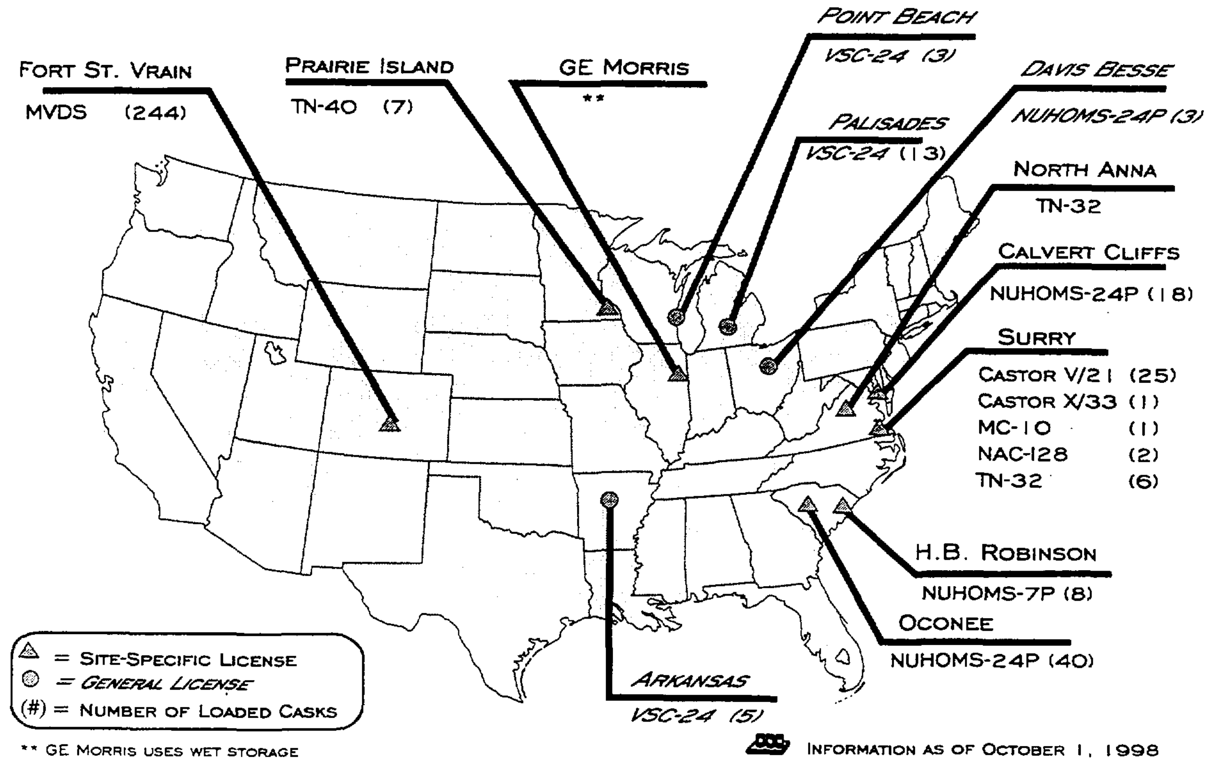


*FIG. 1. Operating spent fuel storage sites*

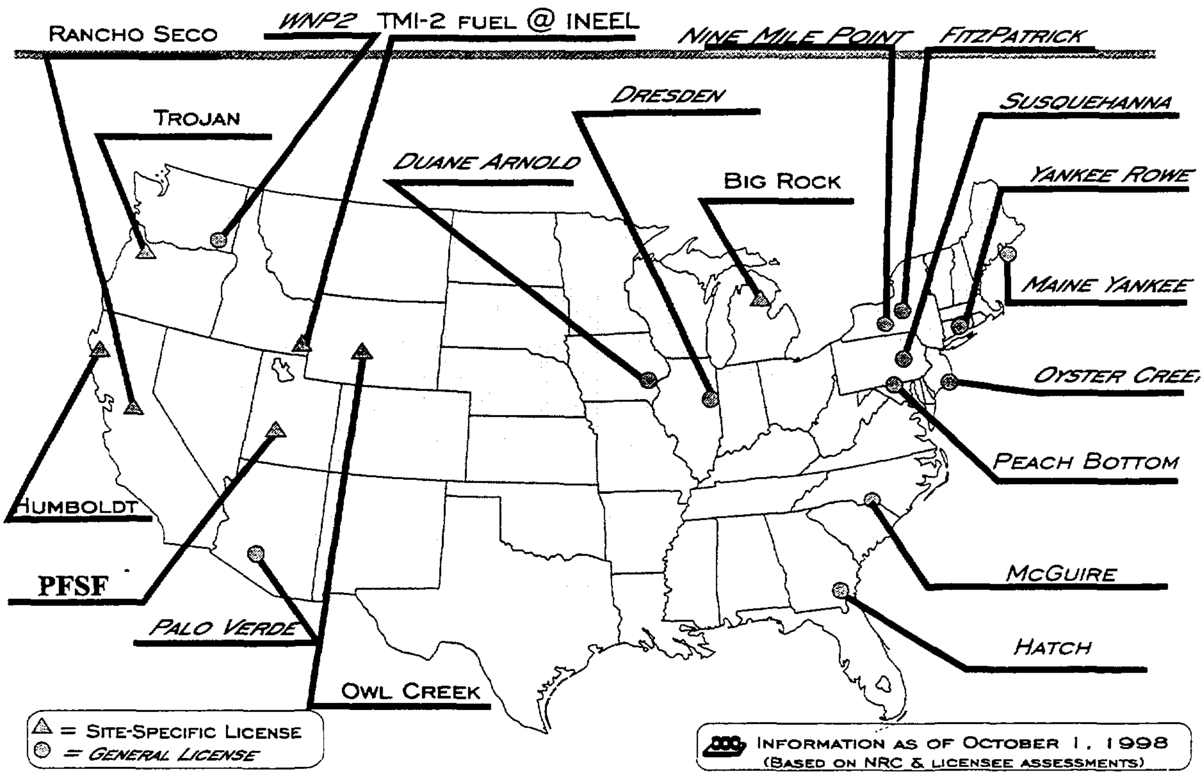


*FIG. 2. Potential near-term spent fuel storage sites*

## 3. THE U.S. NUCLEAR WASTE PROGRAMME

The NWPA required the DOE to establish a Federal Waste Management Programme (Programme) to develop, construct, and operate a system for spent nuclear fuel and high-level radioactive waste disposal. The system identified by NWPA includes a permanent geologic repository, interim storage capability and a transportation system. The NWPAA, which amended the NWPA in 1987, specified a site at Yucca Mountain, Nevada for evaluation as the site of a first repository. Study of the site is still being conducted, and the planned date for receipt of fuel at Yucca Mountain, if it proves satisfactory for a repository, is 2010.

In the early days of the Programme, DOE recognized a need to develop advance technology transport only casks. In 1988, DOE contracted with five cask vendors for development and NRC certification of two truck casks and three rail casks [5]. Later DOE contracted for development of a multipurpose canister system (MPC) which was based on a single canister that would be used in conjunction with various overpacks for transport, storage, and disposal of spent fuel [6]. Due to competing priorities within the Programme, changing schedules for the start of operations of the repository, changing industry needs, and other factors, DOE abandoned its direct involvement in development of these cask systems. DOE's current strategy is to place primary reliance on the private sector to provide storage and transport services as they are needed by the Programme. This approach will rely on the private sector to provide such services and equipment, as needed, for transport and possible storage at a Federal facility [7]. The services will be acquired through a contracting mechanism with appropriate private sector vendors.

### 3.1. Geological repository for spent fuel disposal

Under the 1982 NWPA DOE was evaluating several potential sites for construction of a repository that was expected to be available by 1998. The 1987 amendment to the NWPA limited site characterization to one site at Yucca Mountain, Nevada. Also, in 1987, DOE announced a five year delay in the opening of the repository, and in 1989 a further delay was announced bringing the revised opening to 2010. The delays in opening a repository attest to the various difficulties associated with developing such a facility.

The DOE is currently on schedule to submit a Viability Assessment (VA) for the Yucca Mountain site to the President and Congress. As the name implies, the VA will assess the viability of the Yucca Mountain site as a suitable location for a repository. Subsequent activities for completion of the repository include completion of an environmental impact statement in 2000, and if suitable, recommendation of the site to the President in 2001 and submittal of a license application to the NRC in 2002.

### 3.2. Centralized interim storage

By 2010, the planned date for a repository opening, many utilities will have to store some of the spent fuel they have generated during power production on their reactor sites. A way to avoid such a situation, that has been considered by some members of the US Congress, is the use of a Centralized Interim Storage Facility (CISF). A CISF could serve as a collection point for temporary storage of some of the nation's spent nuclear fuel until a permanent repository is licensed constructed, and started to operate. Although DOE continues to focus its efforts on developing a repository, it must be ready to respond to possible policy redirection related to interim storage. One step already taken to prepare for such redirection has been preparation of a generic CISF design which could provide an early start for development of a site specific CISF (see Fig. 3). The design has been describe in a Topical Safety Analysis Report (TSAR) which has been submitted to the NRC for review in May 1997 [8]. The NRC has finished its initial review and submitted comments to DOE. DOE has revised the TSAR to address NRC's comments [9]. The revised TSAR was submitted to the NRC in September 1998 for final review. The NRC is expected to issue an Assessment Report in 1999, describing the results of their final review.

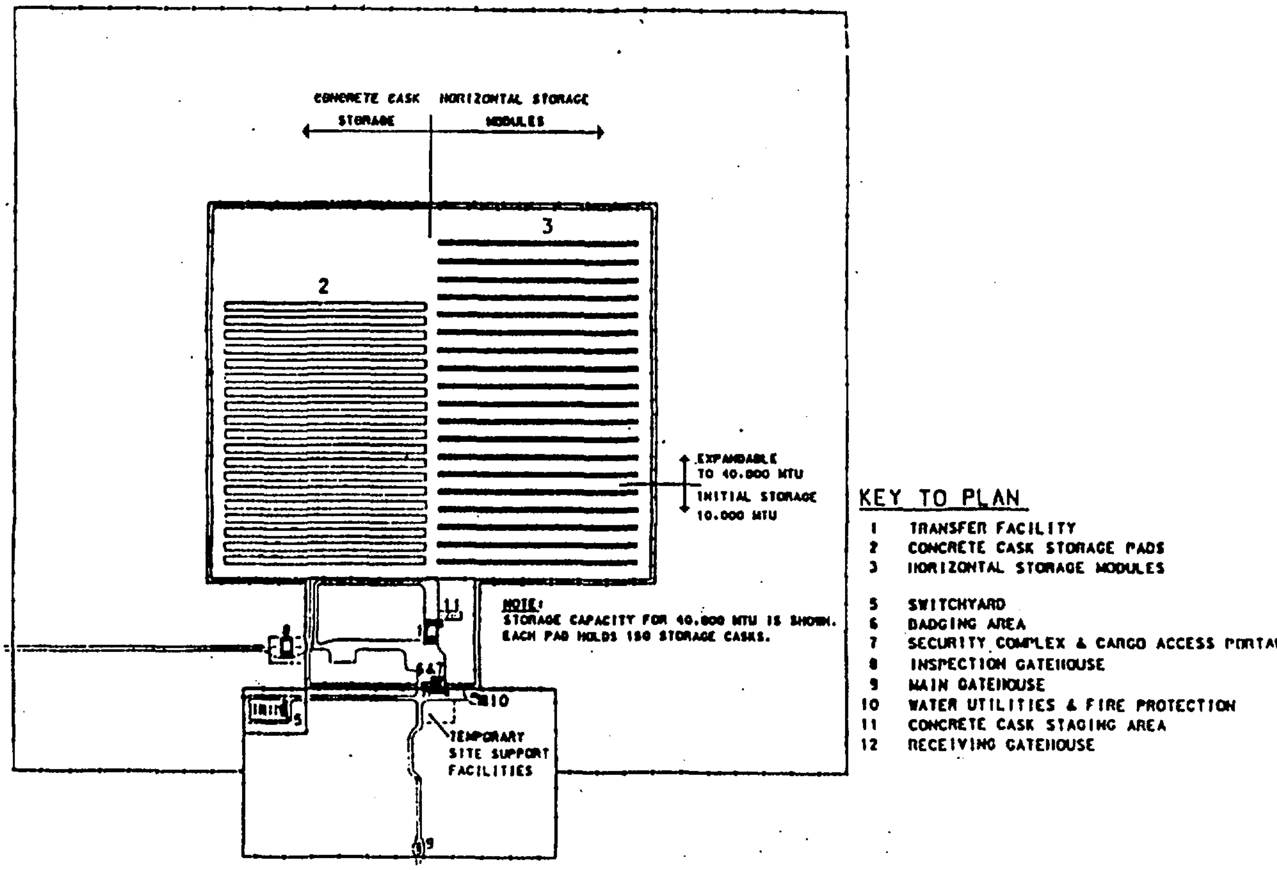


*FIG.3 Overview of the Centralized Interim Storage Facility (CISF)*

The CISF is designed as a temporary, above ground facility for storage of up to 40,000 tU of spent fuel. The facility is designed without a specific site for construction. It is designed and developed so that can be located almost anywhere within the contiguous US. The CISF TSAR covers the Phase I CISF, which will receive canistered spent fuel or dual purpose cask systems. A later phase (i.e., Phase II) will be designed to acceptuncanistered spent fuel.

The CISF will receive, handle, and store spent nuclear fuel. Only NRC approved dual purpose systems will be received at the Phase I CISF. The Phase I facility is designed to accept spent nuclear fuel at the rate of 1200 tU per year for the first three years, 2000 and 2700 tU per year for the fourth and fifth years, and 3000tU per year for the sixth year and beyond.

Storage of spent fuel at the Phase I facility will be based on the use of dual purpose casks and canister based dual purpose systems that have been certified by the NRC. For the Phase I CISF design and TSAR, five systems currently being developed and/or certified by the NRC as dual purpose systems have been considered. The systems considered for the CISF include: 1) VECTRA's NUHOMS, 2) Holtec's HI-STAR, 3) Sierra's TranStor, 4) Westinghouse's MPCs, and 5) NAC's STC.

The main components of the CISF are the transfer facility, the storage area, and support facilities for such things as fuel receipt, fuel inspection, power supply, security, water supply, and fire protection. The transfer facility is a reinforced concrete building which is 250 ft (77 m) long, 88 ft (27 m) wide, and 75 ft (23 m) high. The transfer facility is designed to accommodate transfers of canistered spent fuel between storage and transport overpacks. The Phase I facility is not designed for routine handling of bare fuel. The storage area is comprised of concrete pads for placement of storage casks. The storage area will be constructed in stages, allowing expansion as needed.

### 3.3. Technical development activities

DOE has been involved in establishing a method for using burnup credit for spent fuel transport, and developing a dry transfer system (DTS). Both of these activities could prove beneficial to US utilities as they address their storage needs. Burnup credit is the practice of considering the actual reactivity of the spent fuel when demonstrating a systems criticality safety. Current practice for transport of spent fuel in the US assumes that the fuel is in its most reactive state, that is, unburned or fresh. The use of burnup credit for transportation of spent fuel will permit designers to increase capacities of transport casks over those of current designs which use the fresh fuel assumption. The DTS can be used for cask to cask transfers of bare spent fuel which would allow greater flexibility for fuel handling.

#### *3.3.1. Burnup Credit*

DOE's transportation burnup credit programme began in the mid-1980's to support development of advanced technology spent fuel transportation casks. One of the goals of the advanced technology cask development activity was to increase cask capacities over existing systems. At the time, system capacities were typically one PWR or two BWR assemblies in a truck cask, and 10 PWR or 24 BWR assemblies in a rail cask. The advanced technology casks were being designed for up to four PWR and nine BWR assemblies for truck casks, and 21 PWR and 52 BWR assemblies in a rail cask. Although many other factors contributed to the increased capacities expected for these designs, burnup credit was generally found to be needed to achieve the maximum predicted capacity increases for the PWR systems.

Discussions between the DOE and NRC began on the issue of transportation burnup credit in 1988. Topical reports describing the general methodology proposed by DOE for an actinide only approach to transportation burnup credit were submitted to the NRC for review in 1995, 1997, and 1998. Each of these reports has been a revision of the same basic report, which was modified to reflect DOE's responses to comments and questions that NRC issued after conducting their reviews of the reports. The latest revision of the Topical Report is the DOE's final submittal for transportation burnup credit [10]. Following final submittal of its Topical Report, DOE has been working with the private sector to advance the general methodology by using it for specific cask designs. The current plan is to have a cask vendor use the latest revision of the Actinide-only Burnup Credit Topical Report, along with NRC's comments and recommendations, as a basis to develop a cask licensing case. This licensing case would then be submitted to theNRC by the cask vendor for review and approval.

Although the burnup credit approach just described is intended for use in transport casks, it has a definite impact on current spent fuel storage practices in the US. The importance of transportation burnup credit for storage comes about because of the trend in the US of using dual purpose (i.e., storage-transport) systems for storage at a reactor site. Several US reactors currently use on-site dry storage systems that have been developed for storage only. In some cases, these systems have been loaded at capacities and configurations that would require burnup credit if used for transport. The regulatory basis for these loading schemes for storage is an assurance of moderator exclusion (that is, no water is present in the storage unit under storage conditions), and a requirement that loading is done in a boron controlled pool environment (that is, a specified minimum boron concentration must be maintained during spent fuel loading operations). A number of these storage only systems could meet conditions necessary to be certified for transport, except for criticality requirements that would be met if burnup credit were used and approved. With the move toward dual purpose systems, many utilities and cask vendors would like to convert these storage only systems to dual purpose systems. Furthermore, without the possibility of using burnup credit for spent fuel transport, utilities and cask vendors are faced with the dilemma of either electing to use higher capacity storage only systems or dual purpose systems with less than optimum capacity ratings. Estimates have been made that show that the use of burnup credit could lead to significant increases in cask system capacities which result in reduced public and worker exposure, reduced risk and lower costs [11].

*3.3.2. Dry Transfer System (DTS)*

The Dry Transfer System (DTS) is intended to allow transfer of bare fuel assemblies between casks without the need for complex and costly facilities such as pools or hot-shops [12]. The DTS has been designed and developed under a co-operative arrangement with the Electric Power Research Institute (EPRI). In addition to DOE and EPRI, other participants in this activity include Battelle Pacific Northwest Laboratory and Transnuclear, Inc. The DTS is being developed as a generic design that can be used at a reactor site or storage facility. The system is described in a Topical Safety Analysis Report which is being reviewed by the NRC [13]. The approach, which is similar to that used for the CISF TSAR, is intended to resolve non-site specific issues, allowing expeditious licensing for use at specific reactor sites or storage facilities (see Fig. 4).

The DTS has several potential uses that could significantly benefit the US Federal waste management activities, and private sector initiatives. Some of these benefits include:

1) Spent fuel transfers between casks at reactor facilities that have size or weight restrictions on cask handling, allowing the use of larger storage or transport systems at such facilities.

2) Allowing economic spent fuel transfers between casks at a centralized interim storage facility, avoiding the necessity of constructing transfer pools or hot-shops.

3) Provide an economic alternative to continued operation of spent fuel pools at shutdown reactors.

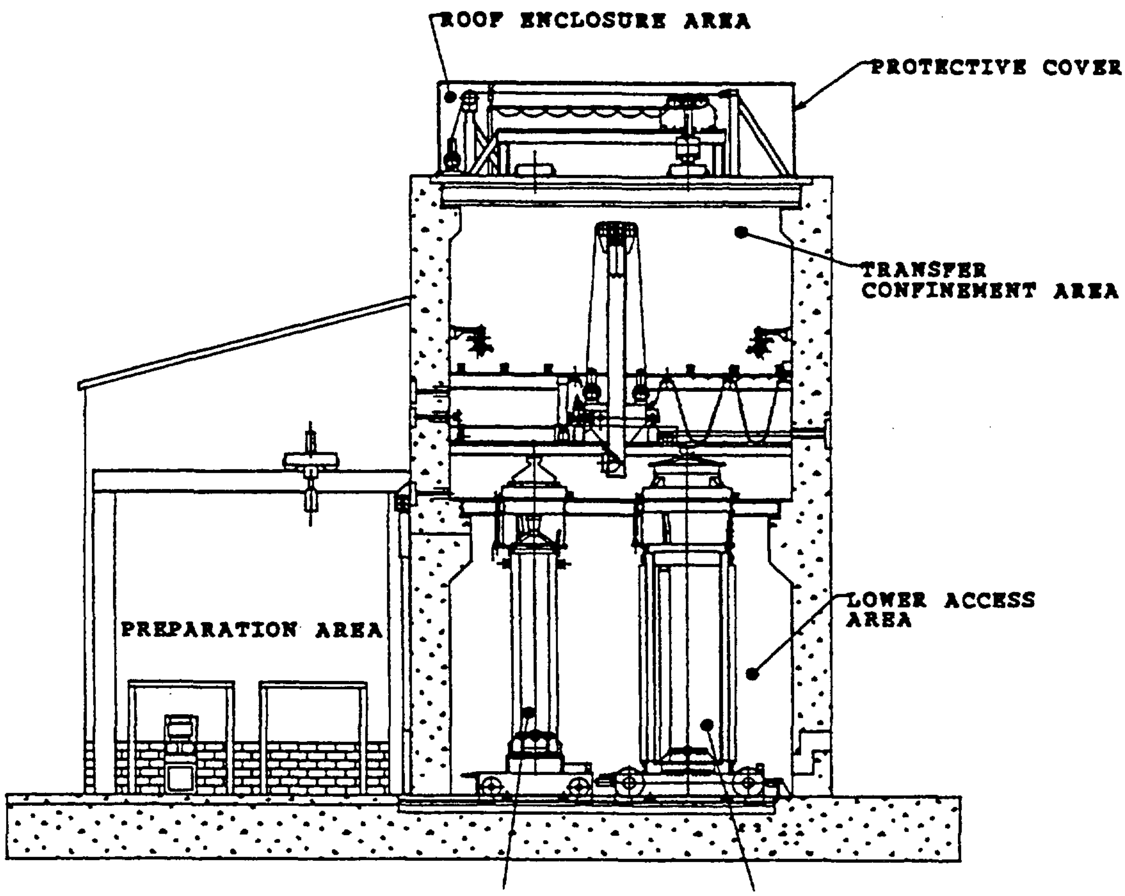


*FIG. 4. Overview of the Dry Transfer System (DTS)*

## 4. CONCLUSION

The US Federal government has adopted nuclear waste policy that relies on deep geological disposal of spent nuclear fuel that is generated in a once-through fuel cycle. However, development of a repository that is necessary for implementation of this policy has proven difficult, and delays have resulted. A number of historical factors have conspired to shape the current spent fuel management practices being used at US nuclear reactor sites. The delay in start up of a repository has further impacted this evolving situation. Although the US Federal government continues to address issues related to spent fuel management, spent fuel currently remains held at reactor sites, and many utility companies are pursuing various storage technologies to handle their developing spent fuel storage needs. The most recent trend that has been observed regarding private sector spent fuel management initiatives, is the selection of dual purpose and multipurpose canister based system for storage of spent fuel at commercial nuclear reactors. The use of these systems, which at minimum can accommodate storage and later transport of spent fuel, provides a capability to move these units off-site with minimal additional handling once a facility to receive the spent fuel becomes operational. The two technology development activities conducted by DOE, i.e., burnup credit and the DTS, have the potential to increase the efficiency of some spent fuel management activities. Both burnup credit and the DTS are currently undergoing NRC review, but once they are accepted, they can be used by the Federal government or the private sector for their respective spent fuel management activities.

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# SESSION 2

# TECHNOLOGY

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# INFLUENCE OF LOCAL REGULATIONS ON TN DUAL PURPOSE BWR CASKS

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**Abstract**

Transnucléaire (Paris, France) and Transnuclear, Inc. (Hawthorne, New York, United Sates) have both developed Pressurized Water Reactor (PWR) spent fuel casks for storage and transport purposes. The products are supplied in Europe by Transnucléaire and in the United States by Transnuclear, Inc. Now the TN Group is working on a design for Boiling Water Reactor (BWR) spent fuel assemblies: the TN 52 L cask is designed for transport and storage is Switzerland, the TN 68 cask is designed for transport and storage in the United States. For storage purpose, national regulatory requirements have to be met: each country has specific demands and criteria. As a consequence, differences between the TN 52 L design and the TN 68 design for rather similar contents appear in several fields: the design work, the licensing process, the manufacturing and the operational life.

## 1. INTRODUCTION

Transnucléaire (Paris, France) and Transnuclear, Inc. (Hawthorne, New York, United States) have both developed high performance spent fuel casks for storage and transport purposes. The products are supplied in Europe by Transnucléaire and in the States by Transnuclear, Inc. They rely on the use of massive steel parts for structural gamma shielding and have proven for two decades a trouble free means to transport and/or store spent fuel assemblies on an industrial scale (see Fig. 1).

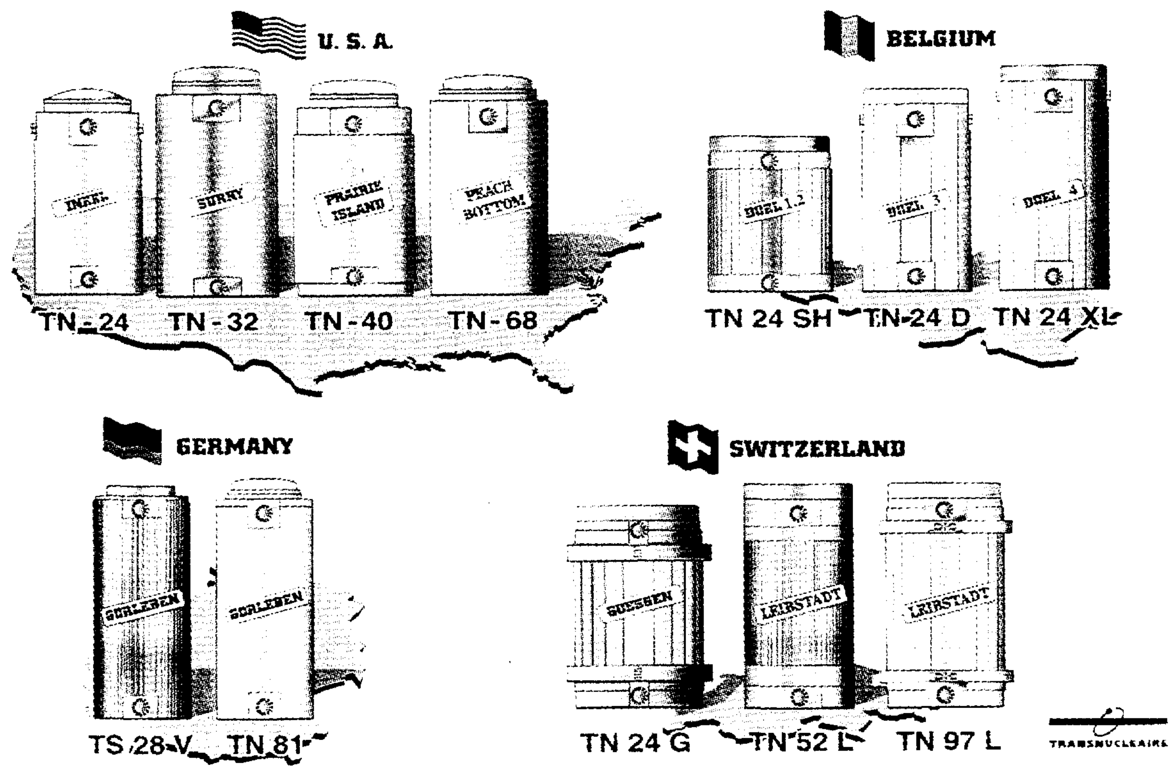


*FIG. 1. Transport and storage casks for many users*

Indeed, one specific advantage offered by this steel cask technology is the possibility of easily adapting the gamma and neutron shielding thickness to any given batch of spent fuel assemblies with given burnup and cooling times, while keeping the same basic concept so that the safety analyses necessary for storage and for transport follow the same path.

### 1.1. TN group experience on PWR spent fuel

TN steel storage casks have been used:

- in Europe with, in particular, the dry dual purpose transport/storage TN 24 D casks and the TN 24 XL casks. They have been licensed, fabricated and they are loaded with Pressurized Water Reactor (PWR) spent fuel assemblies from Doel 3 and 4 power plant in Belgium;

- in the United States with the TN 24 P, TN 32 and TN 40 PWR fuel dry storage casks which have been licensed, fabricated and are in use at Northern States Power's Prairie Island Station and Virginia Power's Surry station. TN 32 will be soon stored at Virginia Power's North Anna Station, Duke Power's McGuire Station and Wisconsin Electric's Point Beach Station too.

Table I shows the number of each cask belonging to the TN 24 family.

TABLE I. THE TN-24 CASK FAMILY

| Name | Site | Number | Status |
|---|---|---|---|
| TN 24 D<br>(TN 24 DH) | Doel 3 | 10<br>(6) | 8 Loaded<br>(Under final licensing work) |
| TN 24 XL<br>(TN 24 XLH) | Doel 4 | 8<br>(6) | 4 Loaded<br>(Under final licensing work) |
| TN 24 G | Gösgen | 4 | Delivery end 1998 |
| TN 24 SH | Doel 2 | 6 | Under final licensing work |
| TN 24 P | INEEL | 1 | Loaded |
| TN 32 | Surry<br>North Anna | 7<br>1<br>(31) | Loaded<br>Loaded<br>(Under manufacturing) |
| TN 40 | Prairie Island,<br>Minnesota | 7<br>(5) | Loaded<br>(Under manufacturing) |

A wide range of fuel characteristics can be concerned (fuel type, residual heat power, burnup, initial U235 enrichment) because of the high adaptability of the TN group steel technology.

### 1.2. TN group experience on BWR spent fuel

Now, the TN group is working on designs for Boiling Water Reactor (BWR) spent fuel assemblies based upon the TN 24 concept:

- the TN 52 L cask is designed to transport and store up to 52 fuel assemblies from Leibstadt (KKL) (Switzerland) with a maximum initial enrichment of 4.9% w/o U235, 50,000 MW·d/tU average burnup and a minimum cooling time of 30 months;
- the TN 68 cask is designed to transport and store up to 68 fuel assemblies with a maximum enrichment of 3.3% w/o U235, 40 000 MW·d/tU maximum burnup and a 10 year minimum cooling time. First use is foreseen at the Philadelphia Electric Company's Peach Bottom Atomic Power Station;

- the TN 97 L cask is designed to transport and store up to 97 fuel assemblies from Leibstadt (KKL) with a maximum initial enrichment up to 3.95 % w/o U235, 26000 MW·d/tU average burnup and a minimum cooling time of 10 years.

The concerned BWR fuel assemblies are GE type (7 x 7, 8 x 8, 9 x 9 and 10 x 10) or SVEA type (Array 8 x 8 for SVEA 64 and array 10 x 10 for SVEA 96). The fuel assemblies can be loaded with or without their external channels and the central cross channel of the SVEA type fuel assemblies can be present or not.

### 1.3. Comparison TN 52 L/TN 68

Both TN 52 L and TN 68 designs meet the IAEA requirements for transportation. The licensing process for obtaining of a B(U)F transport license is underway for both cases. The TN 68 cask also relies on the American regulations (as implemented by the US Nuclear Regulatory Commission).

For storage purpose, national regulatory requirements have to be met. Because there are no international storage rules called or considered as "standards", each country has specific demands and criteria. In addition, some countries have a regulatory package for storage, some rely on their general regulations on nuclear facilities.

After a description of both TN designs and a recall of applicable storage regulations, this paper will list the main differences between the designs due to local regulations in the States and in Europe (Switzerland).

## 2. TN 52 L CASK

The TN 52 L cask belongs to the TN 24 cask family. It is composed of (see Fig. 2):

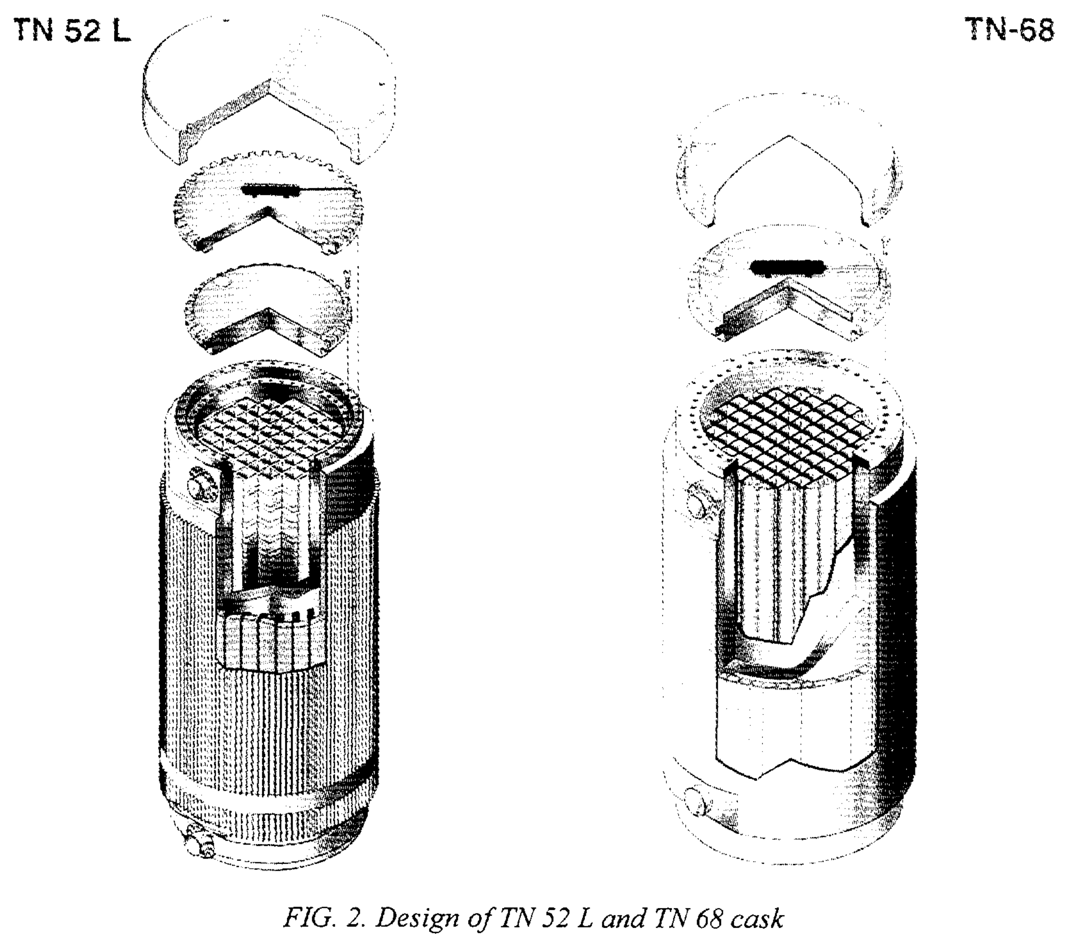


*FIG. 2. Design of TN 52 L and TN 68 cask*

- A cylindrical body, the cavity of which holds an aluminium basket, itself receiving the irradiated fuel assemblies;
- A bottom disk shaped end welded to the body;
- Two lids (primary lid and secondary lid);
- A fuel basket;
- An aircraft crash protection cover for the storage;
- A pressure monitoring system.

This cask is designed for interim storage and transportation. It complies with IAEA transport regulations, and Swiss requirements for interim storage at Zwilag.

### 2.1. Containment

Forged carbon steel is used to manufacture the 3 main components making up the containment with its good mechanical behaviour at low temperature (cf. brittle fracture resistance): a thick cylindrical body, a welded on bottom end, and a topside primary lid. A secondary lid that can be fitted with either metallic gaskets or elastomer gaskets, is used either as secondary barrier when the primary lid carries only one metallic gasket or as a transport lid, the leak-tightness of which is easily checked before shipping after a long storage period.

Thanks to metallic gaskets the lids bolted to the upper end of the cask ensure the leak-tightness of the cavity during interim storage, for a duration lasting from 40 to 60 years. The drain/vent orifice cover is bolted and provided with metallic gaskets too.

Permanent monitoring of the interlid space pressure allows any leak-tightness performance decrease detection long before any release is possible: the pressurization of the monitored space ensures that no gas can flow from the inner cavity to the atmosphere. The TN 24 family casks are in fact no-release systems.

### 2.2. Shielding

Outside the thick steel wall, there is a layer of neutron shielding resin. The gamma shielding stems from the thick steel forgings and also from the contribution of the resin that has a high density. The external thin steel shell that encases the resin also contributes to the radiation protection by absorbing capture gamma rays emitted in the resin.

### 2.3. Criticality control

The cask cavity is fitted with a basket designed as a structural support for the fuel assemblies. It guarantees the subcriticality of the loaded cask. It consists of mechanically assembled partitions in boronated aluminium and in stainless steel defining an array of cells or lodgements, one for each fuel assembly.

### 2.4. Heat dissipation

Embedded in the resin, longitudinal copper plates connect the forged steel wall to the external envelope to transfer the decay heat of the fuel assemblies to the outer shell, and then to the atmosphere. The external envelope is fitted with cooling fins. The aluminium basket conducts heat from the assemblies to the inner surface of the forging body and keeps the maximum fuel cladding temperature well below 350°C. The helium atmosphere within the cask also contributes to heat transfer to the cavity.

## 3. TN 68 CASK

The TN 68 cask belongs to the TN 24 cask family (see Fig. 2). It consists of:

- A cask body (shell, bottom, one lid);
- A fuel basket;
- A protective cover;
- A pressure monitoring system.

This cask is designed for interim storage and transportation. It complies with NRC regulations.

### 3.1. Containment

The containment vessel for the TN 68 cask consists of:

- An inner shell which is a welded carbon steel cylinder with an integrally welded carbon steel bottom closure;
- A welded flange forging;
- A flanged and bolted carbon steel lid;
- Two orifice covers for venting and draining.

Double metallic O-ring seals are used for the lid closure and the orifices. The O-rings interspace is monitored for leakage in storage configuration. The cask cavity is pressurized with helium. The interspace between the metallic seals is monitored and pressurized with helium so that any seal leakage would be towards the cavity or the atmosphere.

### 3.2. Shielding

The TN 68 cask consists of a relatively thin containment vessel surrounded by gamma shielding. The gamma shielding is provided around the walls of the containment vessel by a carbon steel shell that is welded to a bottom shielding plate and to the closure flange. Neutron shielding is provided by a polyester resin compound. The resin compound is enclosed in aluminium boxes that surround the cask body. These boxes are encased in a painted carbon steel shell.

### 3.3. Criticality control

The fuel basket is a plate and box structure that uses existing proprietary laminating methods to produce a light basket with the heat transfer and strength properties of a heavier basket. The basket structure consists of an assembly of stainless steel cells joined by fusion welding to stainless steel bars. The stainless steel bars hold the boxes together and provide structural support. Between the stainless steel bars are placed poison (borated aluminium or equivalent) plates which form an egg crate structure. The poison plates are used for criticality control and heat transfer. No structural credit is taken for the poison plates.

## 4. REGULATIONS/SITES

### 4.1. Switzerland

In Switzerland, there is a centralized interim storage site (Zwilag) dedicated for both spent fuel assemblies and high activity level wastes canistered in glass. This site will be in operation by middle 2000 and all the Swiss utilities participate in the management of this Site.

The Swiss Competent Authorities, HSK, issued guidelines for nuclear facilities that are also applied to interim storage purpose on Zwilag Site. As long as spent fuel assemblies are concerned, the main references of these regulations are HSK-R-14, HSK-R-41 and HSK-R-102. In addition, specific requirements ("Referenzanforderungen") have been set for Zwilag by HSK (see Table II).

TABLE II. SPECIFIC REQUIREMENTS FOR ZWILAG

| | FIELD | MAIN CRITERIA |
|---|---|---|
| Normal conditions | Thermal | Ambient temperature: 32°C,<br>temperature of cladding < 350°C - 390°C<br>(according to fuel type).<br>Natural convection |
| | Activity Release | Double barrier, each guaranteeing a leak rate less than $10^{-8}$ $Pa.m^3.s^{-1}$ |
| | Radiation Protection | Maximum equivalent dose at surface < 0.5 mSv/h (neutron + gamma)<br>Surface contamination $\alpha < 0.37$ $Bq/cm^2$<br>$\beta/\gamma < 3.7$ $Bq/cm^2$ |
| | Criticality | $k_{eff} + 2\,\sigma < 0.95$<br>$\sigma$ = standard deviation |
| Accident conditions | Thermal | Fire 60 min, 600°C.<br>+ Temperature of cladding < 350°C - 390°C<br>After aircraft crash (fighter F18) integrated dose to population < 100 mSv |
| | Mechanical | |
| | Radiation Protection | |
| Environment Earthquake | Mechanical | No tip-over nor domino effect |

**4.2. United States**

The situation in the United States is different because there are many "interim storage" sites, they are generally located at the power plant site itself.

The American Competent Authorities, NRC, issued generic regulations reference 10 CFR Part 72, that opens the choice to license the storage system either on a site specific or on generic basis. Moreover, the main requirements for the storage are given in Table III. One has to point out that the sites are open-air ones; in Switzerland, Zwilag is provided with a building.

5. MAIN DIFFERENCES

Differences between the TN 52 L design and the TN 68 design for rather similar contents appear in several fields:

- The design work;
- The licensing process;
- The manufacturing;
- The operational life.

**5.1. Design work**

The criteria to be met are different as long as radiation protection and thermal behavior of both the cask and the content are concerned.

TABLE III. MAIN US REQUIREMENTS FOR STORAGE

| | FIELD | MAIN CRITERIA |
|---|---|---|
| Normal conditions | Thermal | Ambient Temperature: 115°C max<br>Temperature of cladding < 357° C |
| | Containment | Double barrier<br>No leakage of gas out (test leak rate to be assumed) |
| | Radiation Protection | Dose at site boundary less than 25 mrem/year to the nearest resident<br>(External dose rate (on the storage pad) is usually less than 400 mrem/h at surface) |
| | Criticality | $k_{eff} + 2\ \sigma < 0.95$ |
| Accident conditions | Thermal | In case of burial, temperature of cladding < 570°C |
| | Mechanical | Free drop (0.5 m height) on concrete floor |
| | Radiation Protection | Same as normal conditions |
| | Criticality | $k_{eff} + 2\ \sigma < 0.95$ |
| Environment<br>Earthquake,<br>Tornado, Flood<br>Snow | | No tip-over, no sliding of cask, no water in-leakage into cask |

In Switzerland, the criterion of maximum surface dose rate is quite harsher than the one of the transport regulations. This criterion is also less severe for the American case. In a similar way, maximum allowable temperature of cladding is higher in the States than in Switzerland. Moreover, because casks are stored in a building in Switzerland, the mutual influence of heat power on all the stored casks has to be considered; when the site is an open-air site, this is less of a concern.

On the contrary, the natural events like tornado, storm, snow... have no influence on Swiss designs but can influence American designs: the TN 68 cask is provided with a torispherical weather cover for a better weather protection. This is not necessary for Zwilag.

However, the TN 52 L is provided with an heavy aircraft crash protection cover in order to guarantee a full protection against F18 fighter for the building is not designed to withstand such an impact. This item is not required in USA.

Considering an average initial U-235 enrichment in BWR fuel assembly is an accepted practice in the United States, not yet in Switzerland (nor in France): for a same type of loaded fuel assembly, the neutron absorbing dosage in the basket will be different.

Finally two different lids are a requisite of Zwilag interim storage. In USA, only two containment barriers are required: the TN 68 design has one lid, the cladding being considered as a first barrier; the TN 52 L cask is provided with two independent lids.

### 5.2. Licensing process

Each country Competent Authorities have a specific method of assessing a cask design before delivering any license. The approach and the list of documents submitted to Competent Authorities may be quite different.

In Switzerland, 4 steps of documentation are required (called Hierarchy 1, 2, 3, 4 documents). They are transmitted according to a specific time schedule. H1, for example, has to be submitted by the future owner before any commitment of purchasing a cask. This document gives an outline description of the design and explains how the designer intends to fulfil the regulatory requirements. HSK approval is necessary.

In USA, it is necessary to wait for a green light from NRC before launching manufacturing. After license application, NRC will return questions to the applicant: a strict process is defined. Two rounds of questions are usually necessary.

Concerning licensability demonstration, tests may be performed in order to prove the good behaviour of the cask in case of drop. In Europe these tests demonstrate both mechanical adequacy and leak-tightness of cavity. In the States, demonstration of containment is not directly derived from drop tests, but from bolts analysis under conditions derived from drop tests and from test bolts with strain gauges.

### 5.3. Manufacturing

Because demonstration of good resistance of the steel containment vessel to brittle fracture is different in USA and in Europe, chosen materials are different(*).

In America, strict intrinsic limits on material characteristics at low temperature are given, which leads to eliminate the "LF5" steel as possible material. The TN 68 is therefore composed of SA-250 LF3 material for the inner containment.

In all countries besides the USA, the approach to brittle fracture at low temperature is based upon the well-established theory and practice of dynamic fracture propagation. This theory has been successfully applied also to reactor vessels with cracks. On this basis, the European TN casks are composed of carbon steel A 350 LF5.

Borated aluminium or steel used in the rest of the world is not allowed in USA for structural purposes: the basket manufacturing process is different, as well as the material procurement.

### 5.4. Operational life

As long as release of activity is concerned, and although the number of lids is not the same for both TN 68 and TN 52 L designs, the philosophy is the same: a pressurized dynamic barrier guarantees no-release and an easy preventive monitoring. The difference between these two casks is that the overpressure is put in the interlid space for Switzerland and in the intergasket space for the United States.

From the overall surveillance point of view, the open-air storage requires visual additional controls due to the environment (property of cask, for example).

## 6. CONCLUSION

TN 52 L and TN 68 casks belong to the same family "TN 24" cask and are close cousins. The basic technology is also the same in both cases. However each cask could not be licensed in the country of the other one unless modified.

With a basic philosophy of placing the greater part of interim storage/transport safety in the cask, different regulatory set of requirements and approaches give birth to different evolution of the same basic concept.

(*) to be taken into account during the design work too.

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# ADVANCED TECHNIQUES FOR STORAGE AND DISPOSAL OF SPENT FUEL FROM COMMERCIAL NUCLEAR POWER PLANTS

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**Abstract**

Electricity generation using fossil fuel at comparatively low costs forces nuclear energy to explore all economic potentials. The cost advantage of direct disposal of spent nuclear fuel compared to reprocessing gives reason enough to follow that path more and more. The present paper describes components and facilities for long-term storage as well as packaging strategies, developed and implemented under the responsibility of the German utilities operating nuclear power plants. A proposal is made to complement or even to replace the POLLUX cask concept by a system using BSK 3 fuel rod containers together with LB 21 storage casks.

## 1. INTRODUCTION

Competition among various techniques of electricity generation under liberalised market conditions forces utilities and the nuclear industry to strictly control their costs and to finish unsettled issues as far as possible. In order to ensure competitiveness in the field of spent fuel and radioactive waste management, all cost saving potentials at all steps along the route toward direct disposal have to be explored. At the same time the highest levels of safety standards as well as the flexibility for future improvements, decisions and regulatory needs have to be maintained. Concepts designed in the past decades, therefore, have to be reassessed and confronted with new developments and requirements in view of their commercial use.

## 2. THE ACTUAL SITUATION IN GERMANY

Twenty nuclear power plants with a capacity of about 23 GWe produce some 450 t (HM) of spent fuel per year on an average. An amendment of the German Atomic Act in 1994 was a precondition for the direct final disposal of spent fuel without prior reprocessing. At present, two away from reactor interim storage facilities have been commissioned, a pilot conditioning plant is close to completion and a repository for heat generating waste is under exploration. The cost advantage obtained by direct disposal, which meanwhile for some nuclear power plants is a precondition for economic operation, initiated further improvements. Hereby, experience which has been gained for several years in the field of dry interim storage of spent fuel assemblies, served as a basis. The main components and facilities are described below. For detailed information see [1,2].

## 3. TRANSPORT AND STORAGE CASKS FOR SPENT FUEL ASSEMBLIES

Before being sent to an underground repository spent fuel needs to be cooled down in long-term interim storage facilities. To minimise related costs, big-sized casks of the CASTOR® V-type have been developed, which in future will be mainly used. Casks of the CASTOR® family consist of a thick-walled cylindrical cask body and a double lid system. The cask body is made of ductile cast iron with nodular graphite (ductile cast iron GGG 40). The lids are made of stainless steel. All lids are screwed and leak-tightened with long-lasting metal seals. Radial fins at the outside of the cask body improve passive heat dissipation to the environment. For neutron shielding polyethylene bars are assembled in uniformly distributed drillings in the cask wall. In the bottom area as well as on the lower side of the secondary lid plates of the same material are incorporated. Two pairs of trunnions are screwed onto the cask body. The basket bearing the fuel assemblies is welded and consists of borated special steel plates. To minimise the impact load at the lid and bottom ends of the cask body in case of accidents during the transport, shock absorbers of energy absorbing wood with a steel liner are installed. The casks meet the IAEA safety regulations for type B(U)F-packages. Furthermore, they comply with the acceptance requirements of the interim storage facilities at Gorleben and Ahaus. So far, some 600 casks of different types have been manufactured (Table 1). For the purpose dealt with here mainly CASTOR® V–types will be considered.

TAB. 1. OVERVIEW: PRODUCTION AND USE OF CASTOR® CASKS

| Cask Type | Contents | Produced | Loaded | Remarks |
|---|---|---|---|---|
| CASTOR Ia | 4 PWR FAs | 1 | - | |
| CASTOR Ib | 4 PWR FAs (short) | 7 | - | internal transports at KWO |
| CASTOR Ic | 16 BWR FAs | 11 | 1 | 1 TBL-G; 1 internal transports at KKP |
| CASTOR IIa | 9 PWR FAs | 1 | 1 | 1 TBL-G |
| CASTOR IIb | 8 PWR FAs (short) | 3 | - | internal transports at GKN |
| CASTOR KRB-MOX | defective WWER FAs in special basket | 4 | - | KGR fuel for storage at ZLN |
| CASTOR THTR/AVR | THTR/AVR FAs | 474 | 399 | 305 TBL-A; 94 FZJ |
| CASTOR 440/84 | 84 WWER-440 FAs | 32 | 6 | 3 KGR + 3 KKR to be stored at ZLN |
| CASTOR V/19 | 19 PWR FAs | 13 | 6 | 3 TBL-G; 3 TBL-A |
| CASTOR V/52 | 52 BWR FAs | 3 | 3 | 3 TBL-A |
| CASTOR MTR 2 | Research reactor FAs | 1 | - | VKTA fuel for storage at TBL-A |
| CASTOR HAW 20/28 CG | 28 HAW-canisters | 12 | 8 | 2 TBL-G; 6 at La Hague transports to TBL-G pending |
| TS 28 V | 28 HAW-canisters | 1 | 1 | 1 TBL-G |

| | | | |
|---|---|---|---|
| FZJ | Research Centre Jülich | KWO | NPP Obrigheim |
| GKN | NPP Neckarwestheim | TBL-A | Ahaus Interim Storage Facility |
| KGR | NPP Greifswald | TBL-G | Gorleben Interim Storage Facility |
| KKP | NPP Philippsburg | ZLN | Interim Storage Facility at Greifswald |
| KKR | NPP Rheinsberg | | |

### 3.1. CASTOR® V/19

The CASTOR® V/19 cask [3] is designed for transport and interim storage of spent fuel assemblies from 1300 MW type PWRs (Fig. 1). The capacity of the cask corresponds to 19 uranium fuel elements with a maximum average burnup of 55 GW·d/tHM and an initial U-235 enrichment of up to 4.05 (wt)% or 15 uranium fuel assemblies plus four special fuel assemblies, which could consist of MOX (with a maximum average burnup of 55 GW·d/tHM and a fissile material content (fissile Pu + U-235) up to 3.95 (wt)%, thereof a maximum of 3.7 (wt)% of fissile Pu) or uranium fuel assemblies with a maximum average burnup of 65 GW·d/tHM and an initial U-235 enrichment of up to 4.05 (wt)%.

### 3.2. CASTOR® V/52

The transport and storage cask CASTOR® V/52 is designed for a maximum of 52 fuel assemblies from German BWRs which can be disposed of together with their fuel channels. The dimensions of the CASTOR® V/52 take into account all designs of fuel assemblies that are currently being used in German BWRs and will be used in the future according to today's knowledge. The construction corresponds to that of the CASTOR® V/19. Among the 52 fuel assemblies, there may be up to 16 special fuel assemblies, e.g. containing high burnup Uranium or MOX respectively on specified basket positions. Payloads up to 32 fuel assemblies including up to 12 special fuel assemblies on specified basket positions together with twenty outer basket positions filled with steel dummies. With that additional shielding at the outer basket area a reduction of the minimum cooling time at the expense of the cask capacity can be achieved.

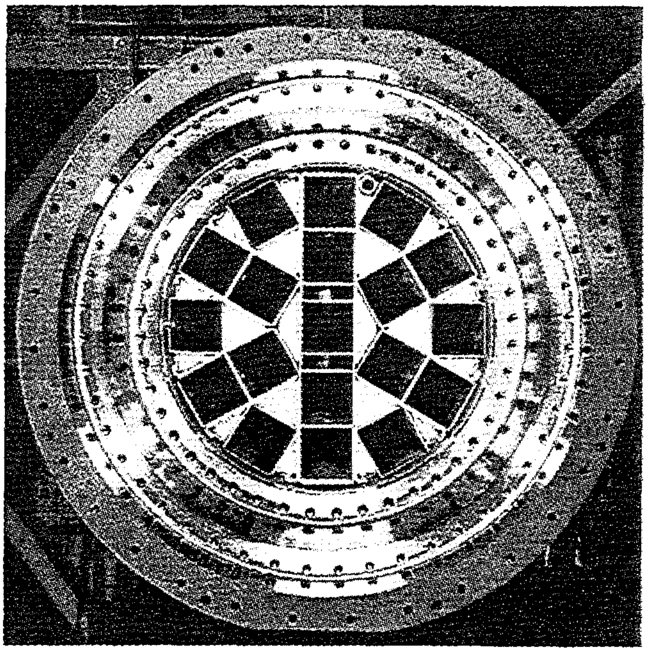

*FIG. 1. CASTOR® V/19*

**3.3. Final disposal cask POLLUX**

In the frame of R&D activities in the field of direct disposal of spent fuel a triple purpose cask for transport, storage and final disposal has been developed [4]. The POLLUX type cask (see Fig. 2) is designed to be disposed of in drifts of a salt dome repository. Precise and final design requirements, however, cannot be formulated before the exploration of the Gorleben salt dome is completed.

The safety analysis report and the licensing documents according to the regulations of the Atomic Act were submitted to the licensing authorities, Bundesanstalt für Materialforschung und -prüfung (BAM) and Bundesamt für Strahlenschutz (BfS) for obtaining the license according to the transport regulations, type B(U)F, and the storage license according to the acceptance requirements of the Gorleben interim store. A drop test programme was carried out in 1994 to demonstrate the cask safety under type B(U) conditions. All licenses for the POLLUX cask are expected to be issued in 1999.

The fuel assembly that formed the design basis is a standard PWR fuel assembly (which has also been used for the CASTOR® V/19 design). All design requirements which can be derived from the remaining PWR and BWR assemblies under consideration are covered by that type. The maximum load includes ten Uranium fuel assemblies or alternatively seven Uranium plus three MOX fuel assemblies with a fuel equivalent of about 5.5 tHM.

Fig. 3 shows the basic design of the POLLUX final disposal cask. It consists of the shielding cask with an inscrewed lid and an inner cask with bolted primary and welded secondary lid. The fuel rods to be stored is inserted in the final disposal cask in cans. The cylindrical wall and the bottom of the inner cask are extruded in one piece and made of fine grained steel. The body of the shielding cask also consists of one piece and is made of ductile graphite iron. Two rows of bore holes in the wall of the shielding cask are filled with moderator material. Besides its primary function, to keep the gamma and neutron doses rate at the surface below the licensed limits, the shielding cask also serves as overpack to meet disposal criteria.

Presently, a prototype cask undergoes cold handling testing in the Pilot Conditioning Facility (PKA, [5]) in Gorleben. Further development of the Pollux cask towards an universal system for transport, interim storage and final disposal could lead to a promising low cost multi-purpose cask concept in the future.

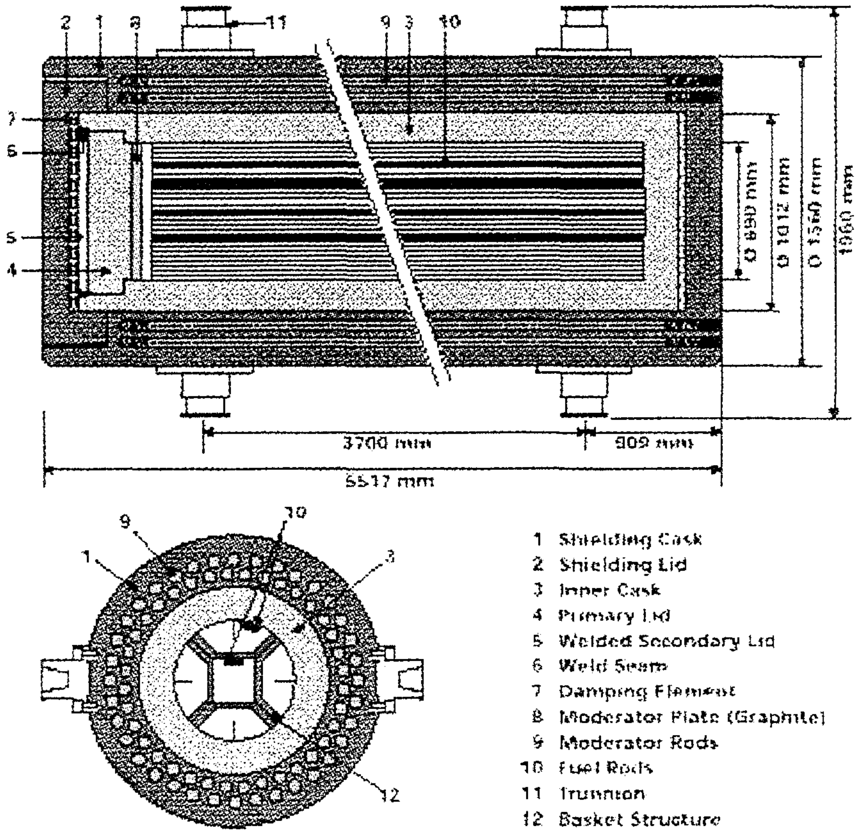


*FIG .2. Final disposal cask POLLUX*

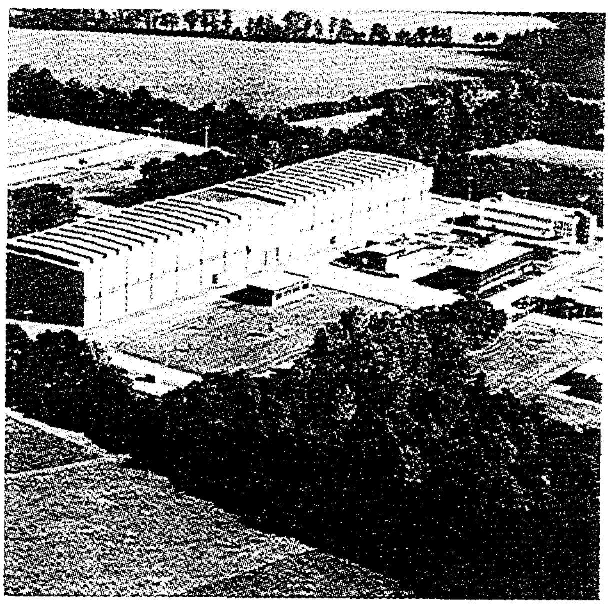

*FIG. 3. Ahaus Interim Storage Facility*

## 4. LONG-TERM INTERIM STORAGE OF SPENT FUEL

### 4.1. Ahaus Interim Storage Facility

The interim storage facility in Ahaus (Fig. 3) consists of a storage hall with the dimensions (l x w x h) 196 m x 38 m x 20 m. It is subdivided into a reception area in the center of the hall where casks are unloaded from the transport vehicles and prepared for storage, and two storage areas where casks are placed in upright position for interim storage. The dissipation of residual decay heat is effected by natural convection through openings in the roof.

From June 1982 to February 1995 305 CASTOR® casks with spent fuel of the decommissioned THTR were brought to the Ahaus facility. In 1998, further six casks of the CASTOR® V series with a total of about 55 tHM from light water reactors (BWRs) were added. The currently valid storage licence comprises a total of 3,960 tHM.

### 4.2. Gorleben Interim Storage Facility

The transport cask storage facility in Gorleben consists of a storage hall of the same construction and dimensions as the one in Ahaus with a capacity of 420 storage positions for CASTOR® LWR casks. The first license issued in September 1993, allowing the storage of 1,500 tHM LWR fuel, was extended in June 1995. Thus, it comprises now a total of 3,800 tHM and includes the storage of fuel assemblies with higher burnup in an extended cask spectrum as well as the storage of vitrified high level waste from reprocessing abroad. The facility was constructed between 1981 and 1983 and taken into hot operation on April 25, 1995 with the storage of a CASTOR® IIa loaded with 9 fuel assemblies of the nuclear power plant Philippsburg-2 (KKP-2). Later a TS 28 V and 2 CASTOR® HAW 20/28 CG casks each filled with 28 cans of vitrified HLW from reprocessing in France were added. Together with three further CASTOR® V casks with spent PWR fuel a total of 8 casks is currently stored.

### 4.3. Pilot Conditioning Facility Gorleben (PKA)

On behalf of the German utilities, GNS constructed a plant in Gorleben, which is supposed to demonstrate spent fuel handling and to test all general services for interim storage and final disposal of spent fuel and high level waste. The process building of the PKA has been completed and the installation of the technical equipment (e. g. hot cells) will be finished within 1998. The third partial licence (commissioning) is scheduled for 1999. Exemplary for the tasks of the PKA are all kinds of cask services, testing of reloading techniques of fuel assemblies from interim storage casks into final disposal casks including fuel rod consolidation [5,6].

## 5. SPENT FUEL DISPOSAL STRATEGIES

### 5.1. Reference concept

The "classic" German way of spent fuel disposal is shown in Fig. 4 (left part). The illustrated concept served as a basis for the development and planning of the described techniques, components and plants.

After a minimum cooling period of 5 years, fuel assemblies from German LWR are loaded into casks of the type CASTOR® V in the storage ponds of the power plants and transported to one of the away-from-reactor interim storage facilities in either Ahaus or Gorleben. Since CASTOR® casks comply with the transport requirements as well as with the acceptance requirements of the storage facilities for long-term interim storage, a reloading is not necessary. After the respective storage period needed for spent fuel cooling and under the precondition that a suitable final disposal facility is available, the casks are transported to the conditioning plant. Here, the fuel assemblies are unloaded, disassembled and packed into containers suitable for final disposal. Therefore, the fuel rods of 10

PWR-assemblies (or an equivalent of BWR-fuel) are loaded into 5 so called POLLUX-cans. The latter again are loaded into a POLLUX cask (Fig. 3). The remaining skeletons of the fuel assemblies, from which the fuel rods have been removed, will be compacted and space-sparingly disposed of, e.g. using MOSAIK casks. It is presumed that the CASTOR® V, used for transport and long-term interim storage, is scrapped.

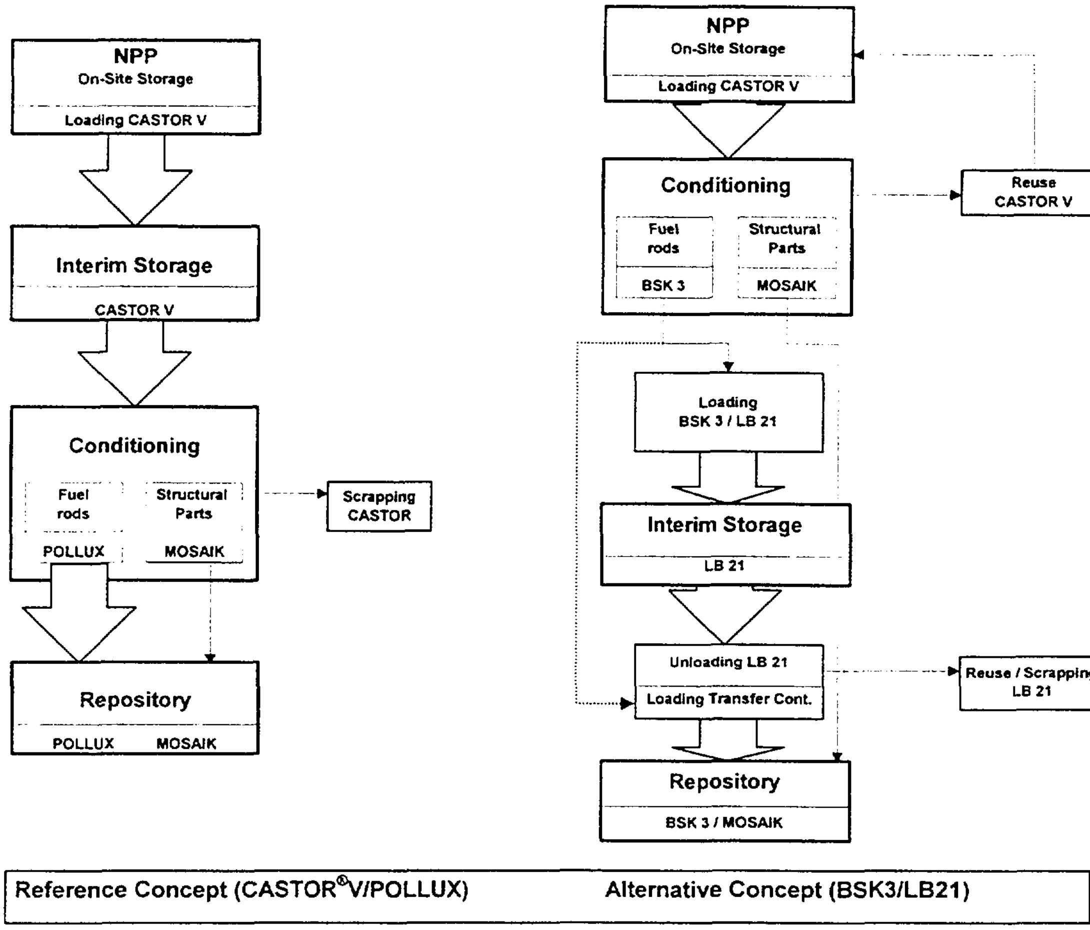


*Fig. 4. Disposal strategies*

## 5.2. The alternative concept

A simple break-down of costs for the entire route of disposal divided into costs for transports, casks, interim storage and conditioning (Fig. 5) prior to repository emplacement shows that those casks used for interim storage and final disposal represent most of the total costs. Investments already made for the existing interim stores, the conditioning facility as well as the repository costs, which are inevitable and independent from the route of disposal chosen, have not been taken into account. These considerations, though, only apply for the actual situation in the Federal Republic of Germany and, therefore, are not transferable to other countries.

Bearing in mind the aforementioned need of reducing spent fuel and waste management costs, an alternative approach is now being considered (Fig. 4, right part). Consequently, it puts emphasis on the avoidance of expensive interim storage and final disposal casks. The essential item is the packaging of fuel rods into a final disposal fuel rod canister, which, in contrast to the POLLUX cask, is not finally disposed of in repository drifts but in bore holes. Therefore, heavy shielding is not necessary. These final disposal fuel rod canisters can be tightly sealed using a qualified welding process in the conditioning facility. This allows interim storage, which is necessary until a repository becomes available, in casks of a simplified design without the otherwise necessary double lid system. Those fuel assemblies foreseen for final disposal are loaded into transport casks in the reactor after the respective cooling period and transported to the conditioning facility where they are unloaded. In contrast to the reference concept, the fuel rod container named BSK 3 (see Fig. 6) forms the final disposal container [6]. It contains the consolidated fuel rods of 3 PWR fuel assemblies (or of 9 BWR fuel assemblies). Seven BSK 3 containers are loaded into a storage cask of the type LB 21 (Fig. 7), which is bridging over the period until the availability of an underground repository. For emplacement of the BSK3 containers into repository bore holes the LB 21 is brought back to the conditioning facility. The containers are unloaded and transported inside a shuttle cask either individually or in groups to the underground facilities. The fuel rod skeletons are compacted in the same way as mentioned in the reference procedure, loaded into MOSAIK casks and finally disposed of.

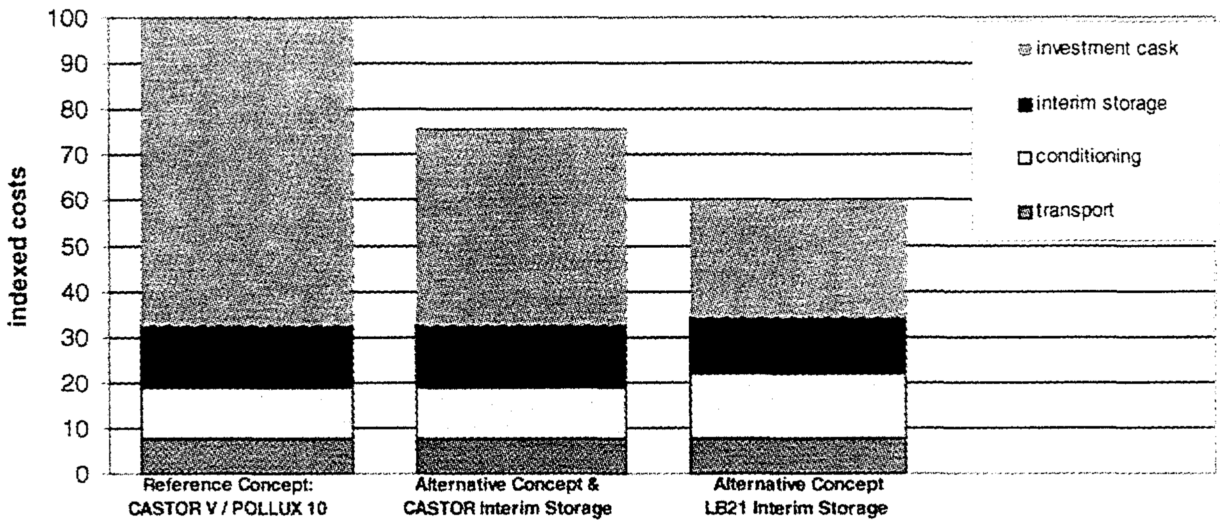


*Fig. 5. Cost comparison of different disposal strategies*

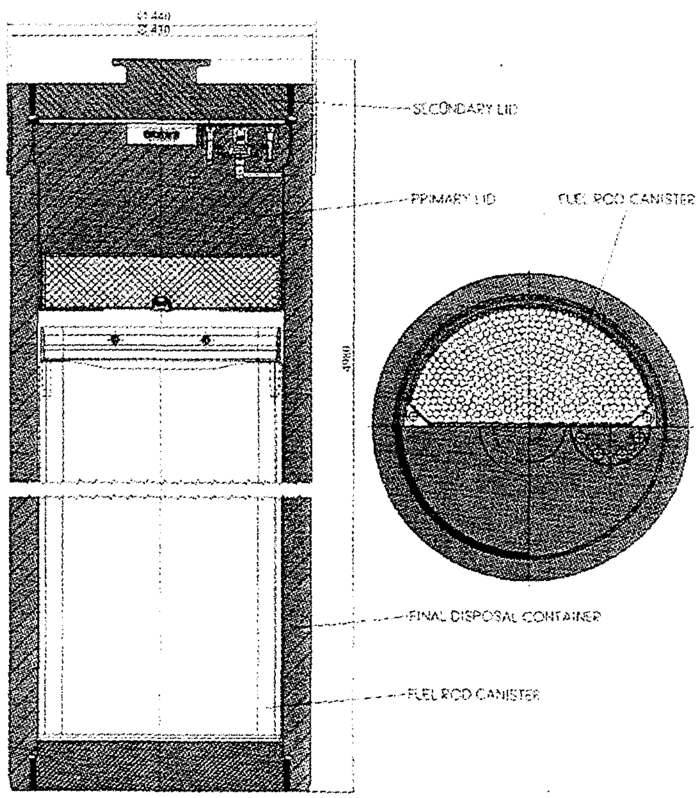


*FIG. 6. Fuel rod canister BSK3*

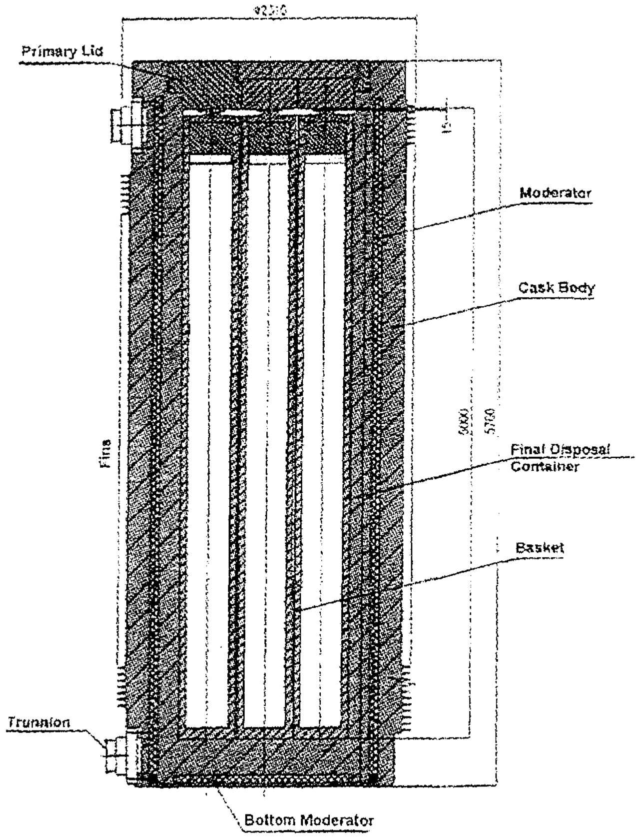


*FIG. 7. Storage cask LB21*

Apart from cost saving aspects, the alternative concept offers extensive flexibility. Positive aspects are:

- the comparatively expensive CASTOR® V cask is exclusively used for the transportation to the interim store, respectively to the conditioning facility. A multiple use of the cask leads to a higher cost efficiency;

- the diameter of the BSK 3 fuel rod canister corresponds to the diameter of vitrified high level waste cans already produced in high numbers and also destined for final disposal. This allows a standardized handling for HLW and spent fuel, also at the final disposal site;

- the BSK 3 fuel rod canister is tightly closed using a qualified welding process in the conditioning facility. Therefore, the interim storage cask LB 21 does not need the usual double lid system;

- the inventory of one LB 21 cask filled with 7 BSK 3 fuel rod canisters correspond to a capacity of 21 PWR-fuel assemblies. The utilisation of storage positions in the interim storage facility and thus the specific storage costs are reduced accordingly in comparison with a CASTOR® V with a capacity of 19 PWR assemblies.

According to the numbers shown in fig.4 the disposal costs of the presented alternative concept decrease to approximately 60 % of the reference concept. Of course, every intermediate solution taking into account short or medium-term interim storage of spent fuel in CASTOR® V casks is conceivable. The cost will then depend on the interim storage period. But they are in any case lower than those of the reference concept.

The transition from the reference concept to the alternative concept requires only minor modifications of the technical equipment of the conditioning plant. The design of the LB 21 storage cask is based on the CASTOR® HAW 20/28 CG for HLW-cans from reprocessing, which is already in use (see Tab. I). The demonstration of the described techniques will be included in the PKA commissioning programme.

Further, significant cost reductions are possible due to the flexibility of the proposed approach with respect to time:

- taking into account the favourable thermal conditions of the BSK 3 fuel rod canisters compared with the POLLUX cask emplacement in salt rock bore holes can be carried out much earlier (about 7 years after unloading instead of 20-30 years).

Upon availability of a repository site the entire period of interim storage following the conditioning process, including the use of the LB 21, can considerably be reduced or even be dropped. This could eventually reduce costs down to less than 30 % of the reference concept.

## 6. FUTURE IMPROVEMENTS

The development of advanced cask designs aims at increased spent fuel inventories together with higher burnup or a higher initial U235 enrichment. In addition, a growing amount of spent MOX fuel originating from the recycling of reprocessed Plutonium needs to be taken into account. Weights and dimensions of spent fuel casks reached already the limits set by the technical boundary conditions of power plants. Therefore, intelligent solutions are required. Among them, an optimized moderator configuration, compact basket designs, new materials and new fabrication processes need to be considered. Table II compares data of the actual CASTOR® V/19-design with an advanced type presently under development.

TABLE II. PROGRESSIVE CASK CONCEPTS

| | **CASTOR V/19** | **CASTOR Va** |
|---|---|---|
| Capacity (PWR FAs) | 19 | 21 |
| Max. Number of MOX-FAs | 4 | 8 |
| Max. Enrichment [%] | 4,45 | 5,0 |
| Burnup [GW·d/tHM] | maximum 55 (15 FAs)<br>maximum 65 (4 FAs) | average 65<br>maximum 75 |
| Typical cooling time [month] | 60 | 60 |

Licensing procedures which in general represent extensive and time consuming activities should also be subject to revision. At present, licences for casks together with their contents form part of the storage licence of the respective storage facility according to the German Atomic Act. For the future it is proposed to separate the cask licences from those of the storage facilities. The latter should then form an "umbrella licence" comprising all requirements which all individual casks have to comply with. Thus, when new cask designs or minor technical revisions are desired extensive reviews of storage licences can be avoided.

## 7. CLOSING REMARKS

Continuous improvements aiming at higher effectiveness and efficiency are common procedures in all industrial areas. Here, nuclear energy is no exception either. However, problems cause the long time periods needed to achieve changes or modifications that are either desired or required. In order to allow adaptations to changing situations in due time, future developments have to be considered and initiated well in advance. Redundancy and diversity are typical key words of the nuclear technology vocabulary. These terms usually used in the hardware sector of safety technology also apply to the spent fuel management strategies in order to ensure the economic operation of the German nuclear power plants. In Germany the technology of direct final disposal, as a complementary measure or as an alternative to reprocessing, is available. The paper presented here shows that the necessary casks and systems together with the typical facilities for long-term storage and conditioning already exist. The facilities are in operation or in progress of commissioning. However, the alternative concepts discussed here need to be developed in more detail. A prerequisite for the realisation, i.e. the necessary political support, has not been investigated here. It is especially this area where Germany currently has to face a difficult situation.

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# SGN MULTIPURPOSE DRY STORAGE TECHNOLOGY APPLIED TO THE ITALIAN SITUATION



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**Abstract**

SGN has gained considerable experience in the design and construction of interim storage facilities for spent fuel and various nuclear waste, and can therefore propose single product and multipurpose facilities capable of accommodating all types of waste in a single structure. The pooling of certain functions (transport cask reception, radiation protection) and the choice of optimized technologies to answer the specific needs of clients (transfer of nuclear packages by shielded handling cask or nuclearized crane), the use of the same type of storage pit to cool the heat releasing packages (vitrified nuclear waste, fuel elements) makes it possible to propose industrially proven and cost-effective solutions.

Studies carried out for the Dutch company COVRA (HABOG facility currently under implementation phase) provide an example of a multipurpose dry storage facility designed to store spent fuel, vitrified reprocessing waste, cemented hulls and end-pieces, cemented technological waste and bituminized waste from fuel reprocessing, i e. high level waste and intermediate level wastes.

The study conducted by SGN and GENESI (an Italian consortium formed by Ansaldo's Nuclear Division and Fiat Avio), on behalf of the Italian utility ENEL, offers another example of the multipurpose dry storage facility designed to store in a centralised site all the remaining irradiated fuel elements plus the vitrified waste.

This paper presents SGN's experience through a short description of reference storage facilities for various types of products (HLW and spent fuel). It continues with the typical application to the Italian situation to show how these proven technologies are combined to obtain multipurpose facilities tailored to the client's specific requirements.

## 1. SGN EXPERIENCE IN DRY STORAGE

### 1.1. Introduction

Whatever the policy adopted for the back end of the fuel cycle, spent fuel from reactors or waste from their reprocessing must be stored temporarily. A number of technologies have been developed and applied by SGN to meet these needs. They concern:

- Interim storage of waste generated by reprocessing, the alternative selected by France and other countries for civilian fuels. This includes:

  ⇒ medium level waste (MLW): hulls and end-pieces immobilized in concrete and bituminized waste are stored respectively in the EDS and STE3 facilities at La Hague. Both types of waste are stacked in ventilated concrete bunkers. These facilities respectively put into operation in 1990 (EDS) and 1989 (STE3) are covered in [1].

  ⇒ high level waste (HLW): fission products and transuranics are embedded in a glass matrix. The glass packages are stored in vertical pits, and cooled with air by forced convection (AVM, R7, T7 facilities at Marcoule and La Hague), or natural convection (EVSE facility at La Hague).

- Long-term storage provided for some spent fuels from the CEA's research reactors at the CASCAD facility pending a final decision. The fuels are stored dry in vertical pits cooled by natural convection.

### 1.2. HLW facilities: R7, T7 and EVSE facilities

The fission products separated by reprocessing operations are vitrified, poured in stainless steel canisters and stored on site in a dry storage building. At the La Hague site, the vitrification and associated glass canister storage take place in the R7 facility for the UP2 reprocessing plant and in the T7 facility for the UP3 reprocessing plant. These facilities were respectively put into operation in 1989 (R7) and 1992 (T7). The R7 and T7 facilities are almost identical. Each facility has three vitrification lines. In each facility, the following functions are performed:

- remote transfer of empty canister to the pouring cell;
- canister docking;
- canister filling and lid placement;
- canister transfer into the cooling cell;
- after cooling, canister transfer to the welding cell for lid sealing;
- canister contamination checking by a smear test;
- canister transfer to the storage building using a shielded handling cask. The storage building consists of several concrete vaults in which glass canisters are stacked within vertical and evenly spaced dry storage pits;
- canister cooling within the store by forced convection.

The main safety criteria specific to the R7 and T7 facilities are:

- containment obtained by the glass matrix and the glass canister (1st barrier) and HEPA filters on ventilation exhaust (2nd barrier);
- residual heat removal (average of 2,500 W/glass canister) to maintain glass matrix temperature less than or equal to 510°C.

To increase the storage capacity, the EVSE facility (extension of T7 facility) has been built and was put into operation in 1995. In the EVSE facility (Fig. 1), the released heat is removed by natural convection: a liner around each storage pit forms a double jacket and the cooling air circulates in the annular space thus formed. The leak-tight pit in which canisters are inserted provides the 2nd containment barrier. The main characteristics of R7, T7 and EVSE facilities are given in Table I.

TABLE I. MAIN CHARACTERISTICS OF R7, T7 AND EVSE

| | **Number of storage vaults** | **Storage capacity per vault** | **Number of pits per vault** | **Canister external dimensions** |
|---|---|---|---|---|
| R7 | 5 | 900 canisters | 100 pits of 9 canisters | D = 430 mm H = 1338 mm |
| T7 | 4 | 900 canisters | 100 pits of 9 canisters | D = 430 mm H = 1338 mm |
| EVSE | 2 | 2160 canisters | 180 pits of 12 canisters | D = 430 mm H = 1338 mm |

### 1.3. Spent fuel storage facilities

Typical SGN spent fuel dry storage facilities fulfil two main functions: unloading and canistering of fuel and dry storage.

As regards unloading, SGN has designed and constructed notably the T0 facility [2] for dry unloading of LWR spent fuel transport casks to be stored underwater in the La Hague pools, for decay heat prior to reprocessing. The T0 facility (Fig. 2), put into operation in 1986, has unloaded about 8,000 tU.

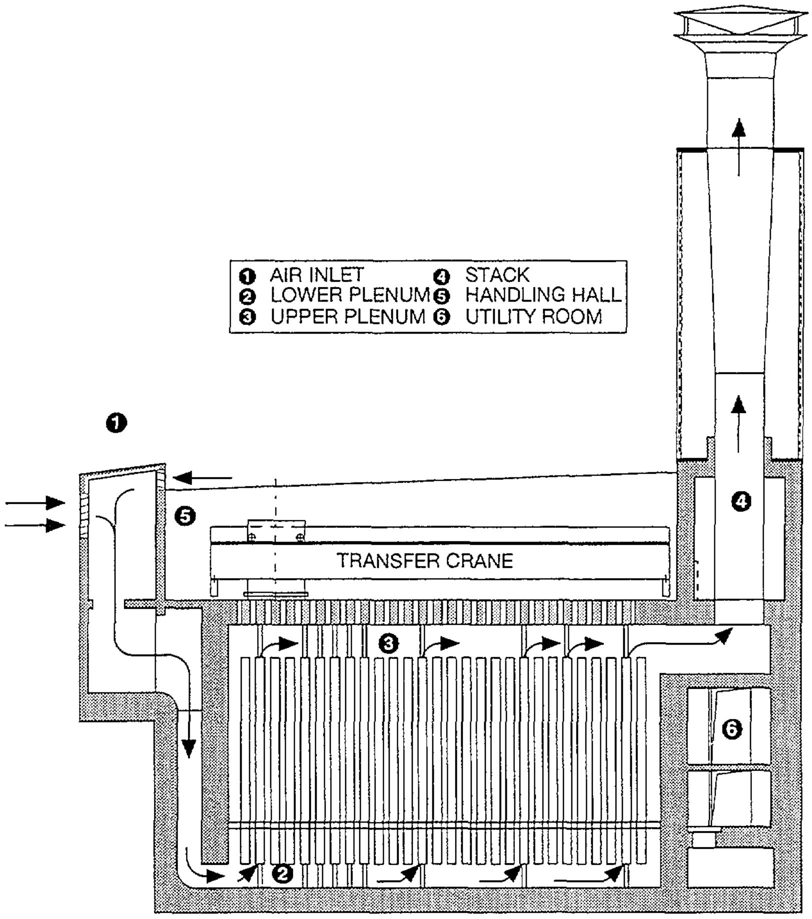



*FIG. 1. Glass canister storage in EVSE facility*

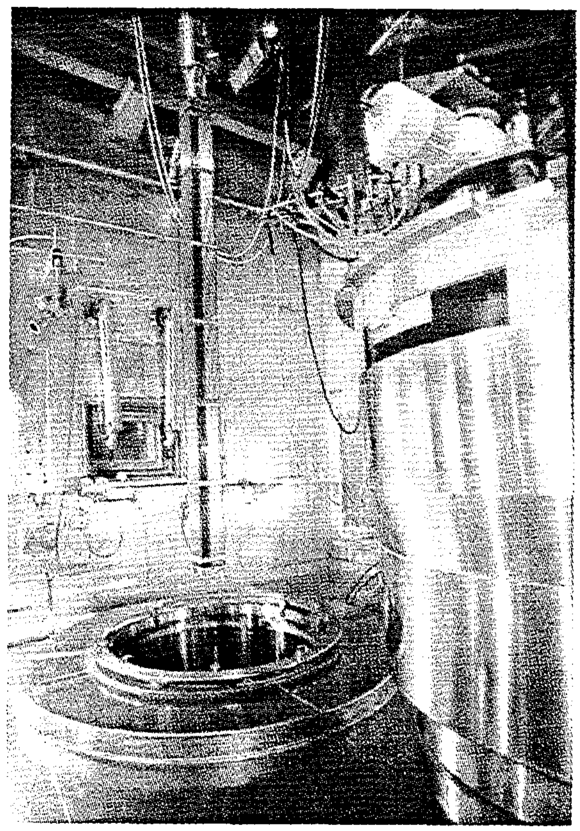


*FIG. 2. T0 Facility - Fuel handling*

Dry storage of fuel is also performed in the APM and Cascad facilities. APM (Atelier Pilote de Marcoule) is a pilot scale reprocessing plant for fast breeder reactor fuel, particularly spent fuel from the Phenix reactor. In the head-end unit of APM, called the TOR facility, leaktight canisters containing the fuel are received, unloaded and dry stored before mechanical and chemical treatment.

CASCAD (CASemate CADarache), located on the Cadarache site, is intended for the dry storage of exotic fuel which cannot be economically reprocessed because of its special properties. This facility, which has been operated since 1990, is planned for a storage period of 50 years after which the fuel will be rehandled according to final decision. Fuel to be stored comes from the CEA (French Atomic Energy Commission) research reactors, particularly from the Brennilis EL4 heavy water reactor. The operations implemented in the CASCAD facility are shown in Fig. 3.

The main safety functions are:

- containment by a multiple barrier system formed by leaktight canisters (1st barrier), leaktight pits (2nd barrier) and rod cladding;
- subcriticality ensured by pit arrangement;
- cooling of fuel to preserve cladding integrity.

Cooling of fuel by natural convection is a passive and thus inherently safe system (Fig.4). The cooling air enters the bottom of the pits, is heated along the pits and discharged to the atmosphere through a stack. The main characteristics of the storage vault are summarized in Table II.

TABLE II. MAIN CHARACTERISTICS OF STORAGE VAULT

| | **Number of storage vaults** | **Storage capacity per vault** | **Number of pit per vault** | **EL4 canister external dimensions** |
|---|---|---|---|---|
| CASCAD | 1 | Equivalent to 180 tU | 319 | D = 104 mm<br>H = 1,100 mm |

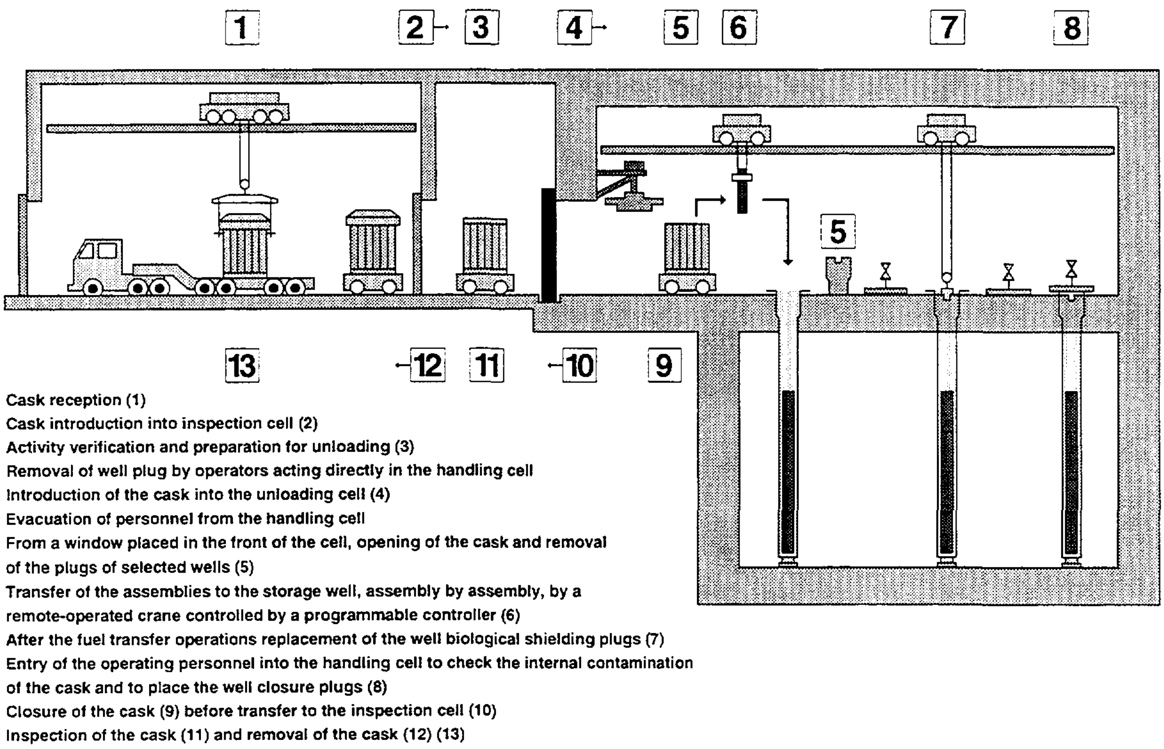



*FIG. 3. Sequence of operations of CASCAD facility*

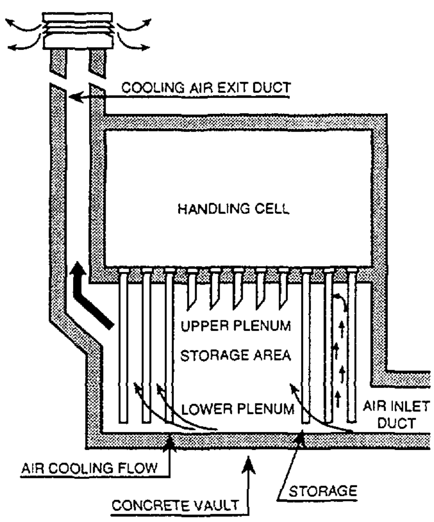



*FIG. 4. Principle of passive air cooling system by natural convection*

The T0 and TOR facilities put into operation in 1986 are not described in this paper because they are covered in references [2] and [3].

### 1.4. SGN multipurpose dry storage concept

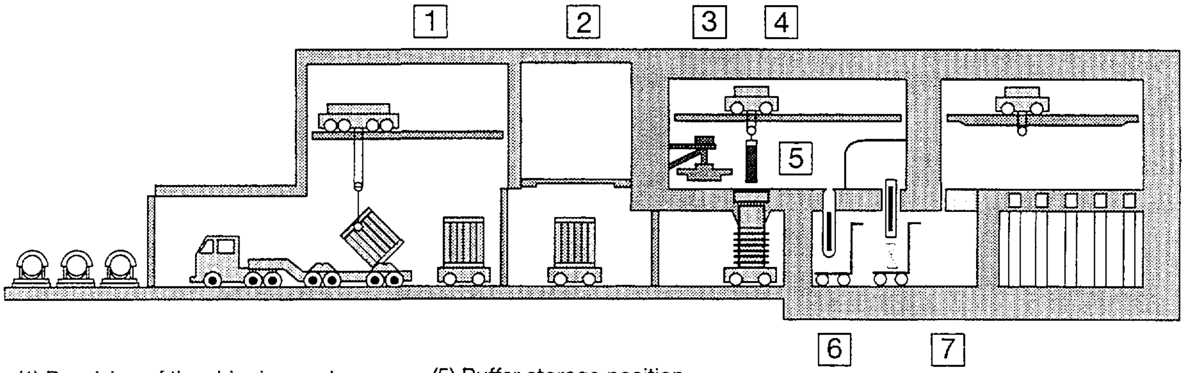
Through the experience SGN has gained in interim storage facilities, for spent fuel, HLW and MLW, the company can offer its clients solutions ideally tailored to their needs. The facilities proposed may be dedicated to a given type of product (fuel element or waste), or may be of a centralized type (various types of waste). They include:

- The unloading unit: this unit offers maximum flexibility, accommodating all types of cask (without any adjustment to reactor loading procedures) and waste and fuel elements. In the handling cell, spent fuel is inserted in canisters dimensioned and adapted to the fuel characteristics to be stored, including nuclear properties (residual power, enrichment, etc.). After the interim storage period has elapsed, this unit also serves to remove the waste packages and fuel canisters to their final destination. This operation requires no complementary installation (Fig. 5).
- The interim storage modules: these modules (vaults and/or bunkers) are built and added as the need arises. The waste packages and fuel canisters are transferred from the unloading unit to the modules by means of shielded transfer equipment (R7 type) or a crane (CASCAD type).

The fuel canisters and waste packages are stored in a concrete structure which protects both the personnel and the public against radiation, but also the fuel and waste against external phenomena, such as earthquake, explosion, etc.

Containment is guaranteed by a double barrier:

- For fuel, the first barrier is formed by the canister, which contains the fuel elements. The canister is inerted, tightly sealed, and checked for integrity. The second barrier is ensured by the storage pit.
- For HLW and MLW, the first barrier is constituted by the matrix of the embedded waste and the associated canister or drum which also offers mechanical protection. For HLW stored in vaults, the second barrier is made by the tight pit into which the canisters are placed. For MLW stacked in bunkers, the second containment barrier is formed by the walls and the building ventilation.




(1) Receiving of the shipping cask
(2) Preparation of cask
(3) Connection of unloading cell/cask
(4) Unloading of spent fuel assemblies
(5) Buffer storage position
(6) Canistering process
(7) Transfer to storage module

*FIG. 5. Handling operations performed in the CASCAD facility adapted to LWR and VVER fuel*

Studies carried out for the Dutch company COVRA (Centrale Organisatie Voor Radioactief Afval) provide an example of a multipurpose dry storage facility designed to store:

- spent fuel;
- vitrified reprocessing waste;
- cemented hulls and end-pieces, cemented technological waste and bituminized waste from fuel reprocessing;
- miscellaneous waste from research reactors and nuclear facilities.

The first two types of waste will be stored in three vaults, in vertical pits cooled by natural convection, using the technology implemented for the EVSE facility. The last two types, packaged in cemented containers or stainless steel drums, will be stored in three bunkers similar to those in the EDS facility. Currently, the project is in the implementation phase, tender specifications for equipment are being prepared and the construction permit is expected by the end of 1998.

## 2. APPLICATION TO THE ITALIAN SITUATION

### 2.1. Italian context

The Italian nuclear programme was abruptly halted in 1987 by a moratorium. The three operating nuclear power stations, one BWR, one PWR and one Magnox GCR, were all shut down. In addition, another small BWR nuclear plant was already in the decommissioning phase. All Magnox and part of PWR and BWR spent fuel elements were sent abroad for reprocessing before 1995.

At the end of 1995, an investigation was started regarding an Away From Reactor (AFR) Storage Facility to store all the remaining irradiated fuel elements plus the vitrified waste which should return from reprocessing. Furthermore, the possibility of storing the part of Superphenix's fast breeder reactor (FBR) spent fuel (one third of the core) charged to ENEL was also contemplated.

Besides the typical storage requirements (subcriticality, heat removal, radioactive material containment and radiation), the facility was also to comply with the following main requirements:

- dry type, passive cooling;
- proven, licensed technology;
- years design life;
- single centralised site;

- self sustaining, in order to allow maintenance in case of problems and shipment to final storage at the end of the interim period, also after decommissioning of all nuclear power plants in Italy;
- capability to store different types of fuel elements (BWR, PWR, old BWR in bottles, high power FBR) and Vitrified high level waste;
- capability to withstand severe accidents (in particular, earthquake and air crash);
- capability of monitoring containment.

Another important constraint was related to existing equipment and physical restrictions in the three current Italian storage sites (Caorso and Trino NPPs and Saluggia storage pool), limiting loads and dimensions of components which could be handled without significant modifications.

The GENESI consortium (Ansaldo Nuclear division and Fiat Avio) was first in charge to select the most suitable storage concept based on the above mentioned concept. As a result of this study, the SGN dry vaults storage was chosen.

**2.2. Description of the designed dry storage facility**

The interim dry storage facility is composed of one concrete building with a roughly parallelepiped shape of approximately 63 m length, 37 m width and 20 m height. A light structure metal building annex to receive the trucks carrying the casks completes the facility. Two stacks, partially made of concrete with upper part metal chimneys, are provided on the roof for cooling air discharge to the atmosphere; a third stack is also planned for building ventilation. The building has four main levels and can be basically divided in three parts:

1. the waste processing system area;
2. the storage area;
3. the services and control area.

The interim dry storage facility is designed for the following purposes:

- reception and unloading of different types of spent fuel and vitrified high level waste (vitrified HLW) transport casks from road trucks or rail wagon;
- unloading of spent fuel elements from transport casks, checking of fuel conditions and conditioning inside welded canisters filled with helium;
- unloading of vitrified HLW packages from transport casks and related contamination checking;
- storage of both fuel canisters and vitrified HLW packages in vertical pits cooled by natural air convection;
- retrieval of all stored waste for final disposal.

To perform the above tasks the facility has been functionally divided into five units, each one in charge of a part of the process.

*2.2.1. Cask reception, preparation and shipping*

Different types of transport casks are to be received in the interim dry storage facility. They arrive in horizontal position on special trailers and are tilted and placed in vertical position onto a transfer trolley after removal of shock absorbers.

The preparation for both spent fuel and glass canister transport casks consists of:

1. cask external contamination checking;
2. cask internal atmosphere checking by internal pressure measurement;
3. check for contamination of the internal cavity (glass canister only);
4. check of fuel integrity by krypton detection in internal cavity (spent fuel only);
5. cask lid unbolting by an unscrewing machine supported by an overhead crane;
6. transfer of the cask to its unloading position.

For transport cask return shipping, the same operations have to be performed in the reverse order. Moreover, a leak test on lid gaskets and a contamination check of the external surface are performed.

*2.2.2. Spent fuel handling in the unloading cell*

For spent fuel transport cask unloading, the cask is placed in and anchored to the unloading cell by means of a specially designed docking station, which is provided with inflatable seals to ensure a leak-tight connection and prevent contamination of the upper part of the cask lid and the lower surface of the unloading cell plug. The cask lid is then removed by the in-cell crane and the fuel elements are extracted from the cask and placed into the buffer storage in the unloading cell; depending on the type of fuel, some checking may also be performed. The cask lid is then re-positioned on the empty cask, which can be undocked and transferred to the preparation area for reshipment. A docking station, similar to the one for the transport cask, is provided for the canister, which is docked in the same way. The fuel elements can then be transferred by the buffer storage to the canister, which is closed after filling by repositioning the lid.

*2.2.3. Spent fuel encapsulation in storage canisters*

Following the operations performed above, the canister is undocked and brought by a transfer trolley to a position where the remote welding of the lid can be performed. After the welding, a helium leak test is performed on the weld, and the canister is vacuumed and inerted by filling with helium gas. A cover cap is then welded on top of the quick coupling connection used for gas filling, and the canister is moved by the trolley to a position under the penetration for access to the storage hall.

*2.2.4. Transfer of fuel canisters and vitrified HLW packages to the storage wells*

Once positioned by the transfer trolley under the storage hall penetration, fuel canisters are rehandled to their storage pit by means of a remote controlledbridge crane.

The crane performs the following operations:

1. storage well lid and shield plug removal;
2. canister recovery from the transfer trolley and transfer to its storage pit;
3. canister positioning;
4. storage pit closing by shield plug and lid repositioning.

After the repositioning of the plug, operators can enter the storage hall for lid bolting.

The same procedure is followed for vitrified VHLW packages, with the difference that the canister is extracted directly by transport cask positioned in a specific location under the storage hall and the packages are stacked on four levels inside the pits.

*2.2.5. Cooling of the storage vaults*

The air cooling system is based on the natural convection concept. Fresh air is supplied through air inlets into the bottom plenum, circulates around pits and is exhausted through the stack. The products are stored according to the decay heat to be evacuated:

- for the highest heat producing products (fast breeder reactor fuel and vitrified fission products), one specific vault equipped with double jacket vertical pits cooled by natural convection and using the technology implemented in the EVSE facility is provided;
- for the lowest heat producing product (BWR spent fuel), a separate vault equipped with single vertical pits cooled by natural convection using the technology implemented in the CASCAD facility is provided.

## 3. ADVANTAGES OF THE PROPOSED TECHNOLOGY

The main highlights and strengths of the proposed technology are:

- Flexibility:
  - the dry storage technology can accommodate any type of fuel and any type of cask can be handled;
  - special waste can be stored in the facility, including vitrified high-level waste (HLW) and other heat releasing waste;
  - the modular design of the storage concept allows storage capacity to be increased as needed;
  - canisters are completely retrievable in the storage concept.

- Safety:
  - double containment barrier;
  - double containment barrier monitoring;
  - passive air cooling;
  - proven technologies.

- A simple, cost-effective solution. Dry storage provides several benefits:
  - low capital and operating costs;
  - no effluents;
  - fewer operating personnel;
  - storing multiple products in a single facility also offers significant cost savings, since only one receiving unit and one set of utilities (electric power, ventilation, etc.) serve multiple modules.

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IAEA-SM-352/25

# FLEXIBILITY OF THE BNFL DRY STORAGE SYSTEMS



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**Abstract**

To widen its range of spent fuel management services, BNFL entered the fuel storage market in 1995; entry was by acquisition rather than internal product development. The need for a transportable product was identified very early, but represents only the first phase of a philosophy of continuous improvement. Strong synergy exists between the new business area and existing fuel handling and transportation expertise, which has been of considerable assistance to the new business.

## 1. BACKGROUND

In the early 1990's BNFL decided to enhance its portfolio of products and services and to provide a comprehensive range of spent fuel management options to its customers. To this end, dry storage was added to the portfolio. Rather than create a new system, to shorten the time to have a licensed product it was decided to buy in the technology. Following a series of desk studies and meetings with potential vendors/licensers, the technology marketed by Sierra Nuclear Corporation (SNC) was selected. Amongst several criteria, cost-effectiveness and development potential were considered particularly important. The SNC product came first on both counts. A series of agreements was therefore concluded with SNC in 1995, allowing BNFL to begin marketing the system outside the USA.

In April 1998 BNFL acquired all of the SNC stock; following the acquisition a new company - BNFL Fuel Solutions - was set up to build on the US business base already well established by SNC. The adaptability of the storage system is demonstrated by the contracts list in Table I, which already includes VVER and test reactor fuel in addition to PWR.

TABLE I. BNFL FUEL SOLUTIONS - DRY STORAGE PROJECTS

| UTILITY | FACILITY | PROJECT START | LOADED CASKS |
|---|---|---|---|
| USDOE | INEEL | Sept 1988 | 1 |
| Consumers Power | Palisades | Feb 1989 | 13 |
| USDOE | FFTF | Nov 1993 | 10 |
| Wisconsin Electric | Point Beach | Dec 1992 | 3 |
| Entergy | Arkansas Nuclear | May 1992 | 5 |
| Portland General | Trojan | June 1995 | [April 1999] |
| Ukratomenergoprom (Ukraine) | Zaporozhye | Dec 1993 | -------------- |

[ ] = anticipated dates

## 2. THE NEED FOR TRANSPORTABILITY

The first-generation SNC product, the Ventilated Storage Cask (VSC), was the first vertical canister/concrete cask storage system to be licensed for LWR fuel, and rapidly gained market share in the USA. Well before 1995 both BNFL and SNC had recognised the need for any future dry storage system to be readily transportable off-reactor site. Possible other locations included at-reactor (AR) or away-from-reactor (AFR) domestic sites, repositories or overseas reprocessing plants.

While metal Dual Purpose (DP) casks offer such transport capability, assuming both storage and transport licences can be obtained, it is generally recognised that DP casks are not so cost-effective as concrete casks, given the same ground rules. Also, each DP cask has in-built transport capability. This can be advantageous if off-site transport is anticipated to be at short notice, but in general that is an unlikely scenario. The in-built transportation capability may then become a liability if transport regulations are altered at some future date, time horizons of 50 years or more are possible.

In contrast, a canister-based fuel storage system with its own purpose designed and built shipping cask has a number of clear advantages:

1. changes in transport regulations can be taken into account before manufacture;
2. approved new structural or shielding materials can be used if appropriate;
3. the shipping cask/fuel canister ratio can be tailored to individual customers, perhaps 1:10 instead of the effective 1:1 ratio for DP casks; and
4. the transport element only needs to be bought near the time of off-site transport

Significant cost savings arise from all four of these items. A joint BNFL/SNC development programme was therefore set up to design and licence a transportable storage system - **TranStor™**.

## 3. OVERVIEW OF THE TRANSTOR™ SYSTEM

Operation of the TranStor™ system requires the spent fuel to be loaded into a specially designed canister. Fuel loading takes place within the fuel pool, during which time the canister is held within a steel transfer cask. Upon completion of fuel loading the canister has its shield lid emplaced and, in the transfer cask, removed from the fuel pool. The shield lid is then welded shut, and a second, structural, lid welded into place. The canister is then vacuum dried and back-filled with helium. If the fuel is to be shipped off-site the canister is transferred to a shipping cask. Otherwise, the canister is loaded into a concrete cask and stored outside on a concrete pad. The VSC system is identical, except for the absence of off-site shipping capability.

The primary components of the TranStor™ system, as shown in Fig. 1, are :

- The sealed canister, providing containment for the spent fuel;
- the concrete cask, providing physical protection and shielding for the canister during storage;
- The transfer cask, providing shielding during fuel loading and during transfer operations into or out of either the concrete cask or the shipping cask;
- The shipping cask, used to transport a loaded canister off-site to another storage site, a final disposal site or a reprocessing facility.

These components were developed using experience gained from previous concrete-based system designs and from metal cask designs. The TranStor™ canister contains an array of steel storage sleeves inside a circular steel shell. The base is a large steel disc welded to the shell. The canister has two lids, shield and structural. These are emplaced after fuel loading and are multi-pass welded to the canister shell. The transfer cask has a steel, lead and neutron shield composite wall and is used to move the canister around. It has thick steel bottom doors operated by detachable hydraulic arms. The concrete cask has a thick internal steel liner for enhanced shielding and strength and contains two rows of reinforcing bar. It provides shielding during storage and creates a natural circulation air flow path for cooling the canister - air taken in at the base passes up the annulus between the canister outer wall and the cask inner wall and is ejected at the top. The use of vertical storage increases the heat transfer capability compared to horizontal storage. The concrete cask/canister combination can be easily transported around the reactor site.

The shipping cask, plus its payload of the loaded canister, is a package designed to meet US Code of Federal Regulations and IAEA Safety Series 6 requirements. It is thus compatible with shipping to reprocessing, storage or disposal sites anywhere in the world, keeping all spent fuel management options open for the reactor operator.

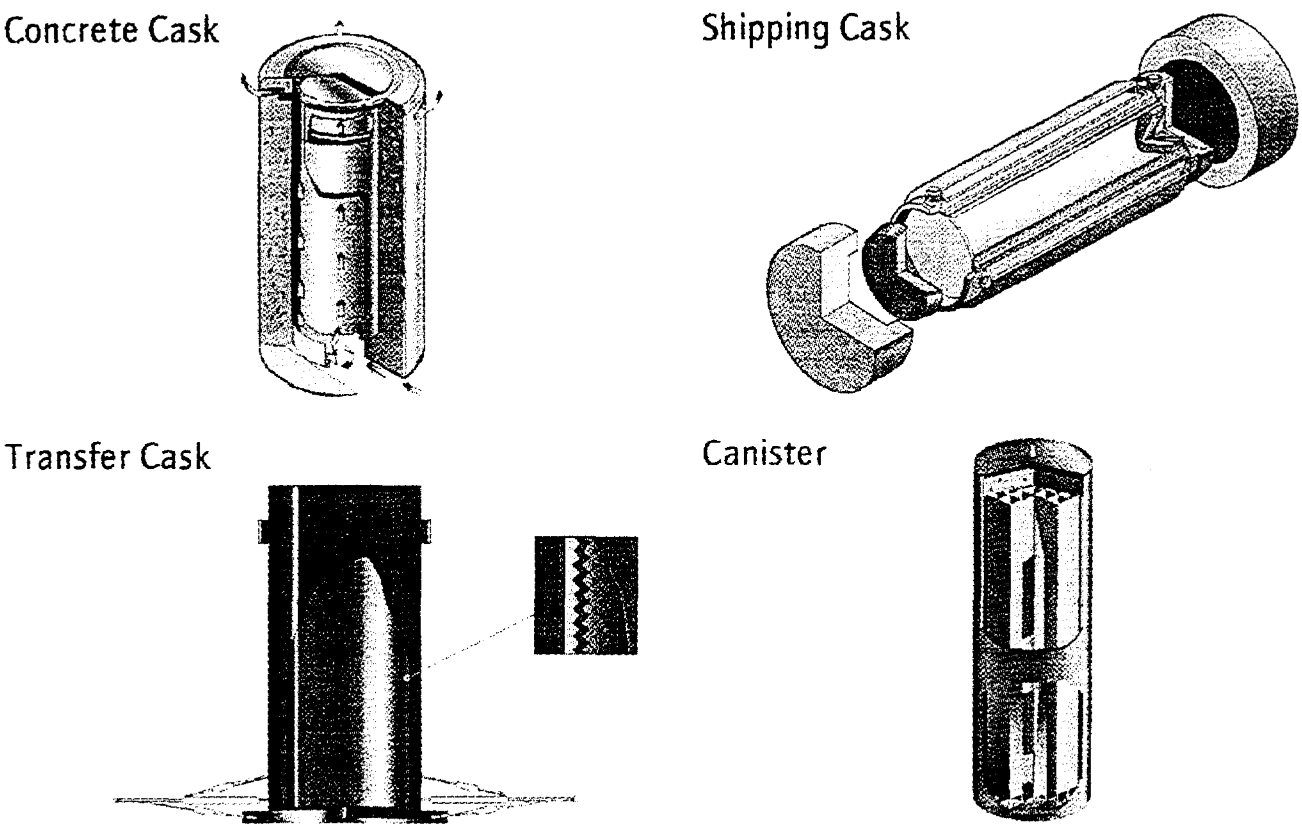


*FIG. 1. TranStor™ system components*

Loaded concrete casks are stored on a simple reinforced concrete pad with a compact layout, typical centre-to-centre spacing of five metres. No building or other cover is needed, although one can be constructed at minimum cost if necessary to enclose the casks and minimise visual impact. Vertical storage minimises the land area required, in comparison to horizontal storage. In addition to inlet/exhaust air temperature measurement and radiation monitors, the storage facility may also include security fencing and lighting depending on local requirements.

The TranStor™ system base design can store zircaloy or stainless steel clad fuel, and can accommodate heat loads of 26 kW (storage) or 24 kW (transport). These can be various combinations of initial enrichment, burnup and cooling time. The primary components in Fig. 1 can be tailored to some extent to suit older reactors which may have smaller fuel assemblies or reactor building restrictions such as crane capacity. As an example of such flexibility, the first-generation VSC system was modified to store highly enriched, high burnup fuel from the USDOE's Fast Flux Test Facility. TranStor™ is also sufficiently flexible to allow storage of a large variety of fuels, as the basket configurations in Fig. 2 illustrate. Photographs of operational casks are presented in Fig. 3.

## 4. LICENSING EXPERIENCE

This now covers two generations of storage systems, and Ukraine as well as the USA, as shown in Table II. The TranStor™ design has a carbon steel fuel basket (strength) and a stainless steel canister shell (corrosion resistance). The VSC canister is all carbon steel.

## 5. APPLICATION OF EXISTING BNFL TECHNOLOGY TO INTERIM STORAGE

BNFL experience in designing and operating transport casks proved very helpful from the beginning. In particular, the TranStor™ PWR fuel basket design drew heavily on tried and tested BNFL design principles. Other technology transfer examples are :

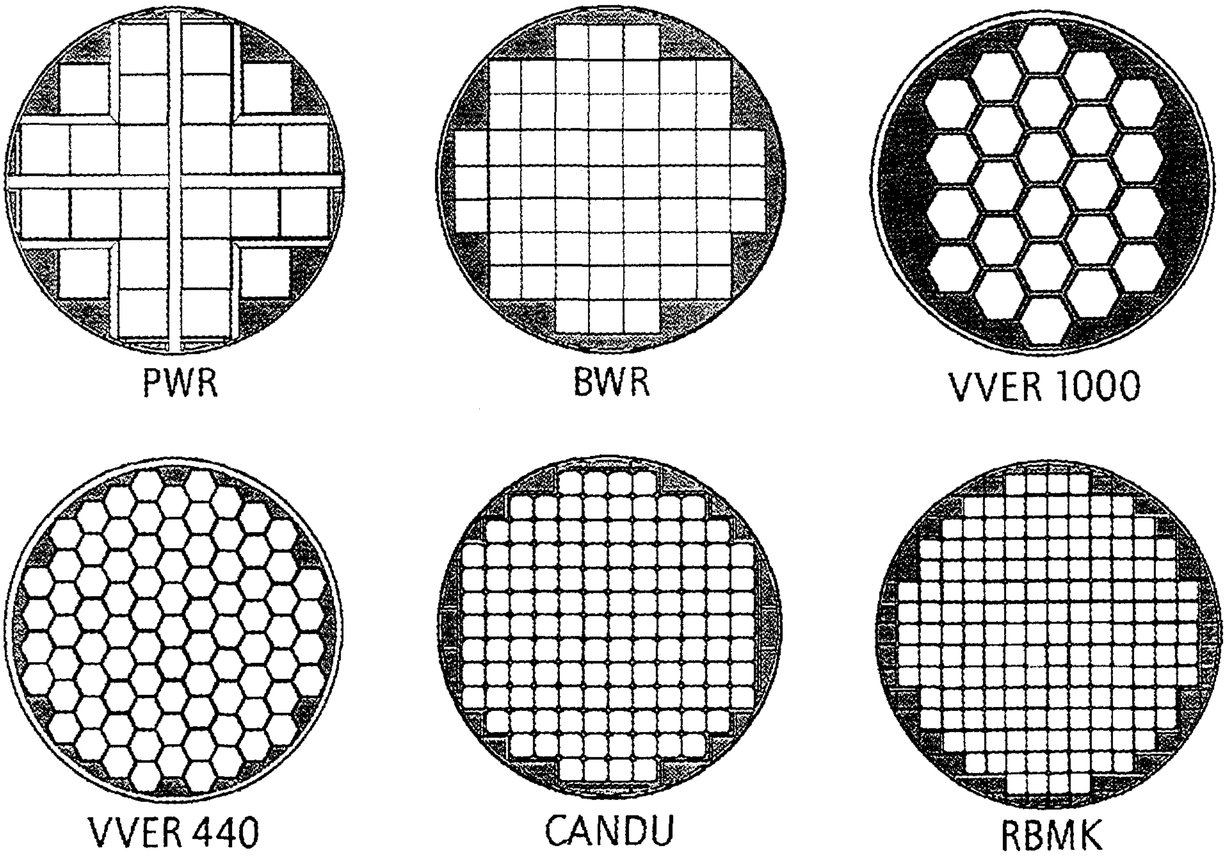


*FIG. 2. Fuel basket configuration*

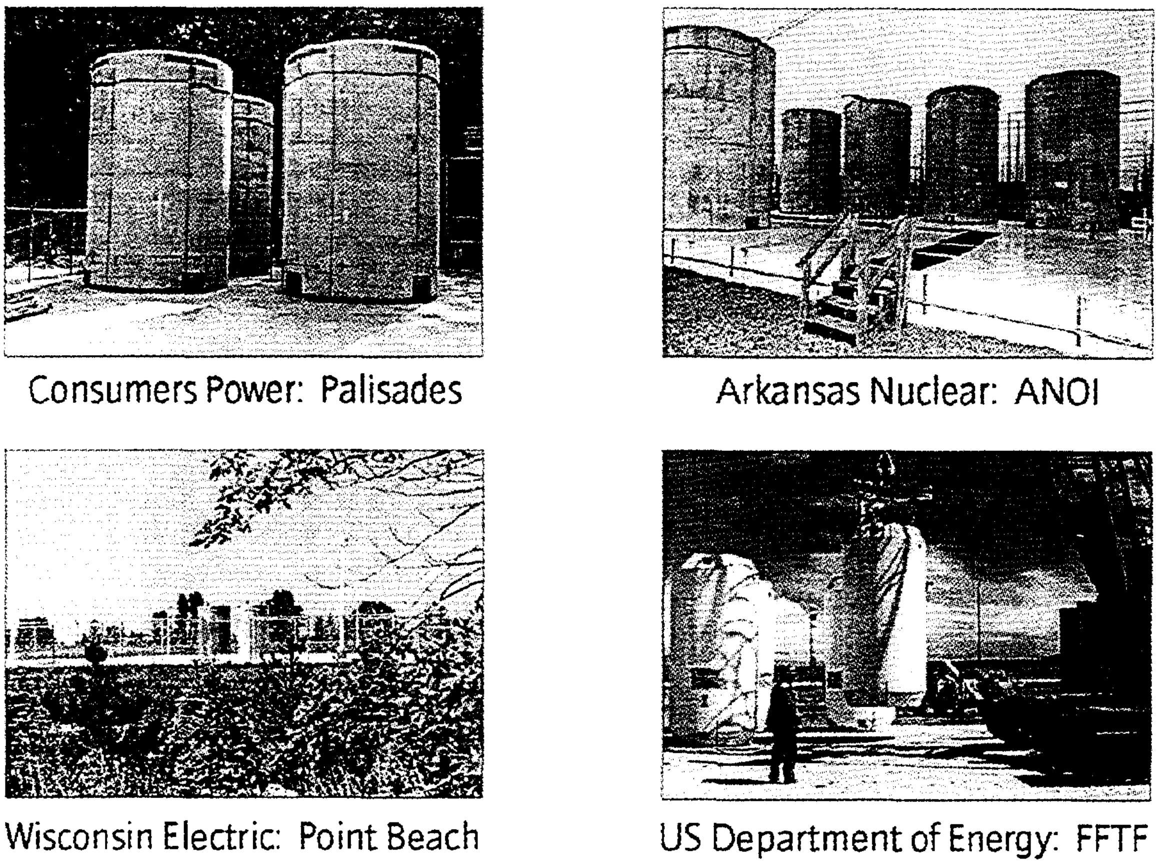


*FIG. 3. Loaded storage casks*

TABLE II. BNFL FUEL SOLUTIONS - STORAGE SYSTEMS LICENSING EXPERIENCE

| SYSTEM | LICENCE APPLICATION SUBMITTED | LICENCE APPROVAL (USNRC UNLESS INDICATED) |
|---|---|---|
| VSC - 24 Topical Report | February 1989 | March 1991 |
| VSC - 24 Certificate of Compliance | April 1991 | May 1993 |
| TranStor™ site-specific for Trojan nuclear plant | March 1996 | [Early 1999] |
| TranStor™ 10CFR71 for transportation | December 1996 | [Early 2000] |
| TranStor™ 10CFR72 Certificate of Compliance | June 1996 | [Late 2000] |
| VSC - 24 site-specific for Zaporozhye | 1996 | -------------- |

[ ] = anticipated dates

- use of ultrasonic know-how to ensure the USNRC-requested weld crack detection technique could be properly implemented. The ultrasonic testing equipment and process developed by BNFL/SNC and the VSC users group has been successfully implemented at the Point Beach, Palisades and Arkansas Nuclear storage projects;
- development of new coatings for the carbon steel canisters - BNFL already had a testing route for radiation-resistant coatings; and
- design of a "hot cell" for the Chernobyl Dry Storage bid. In order to reduce the RBMK fuel to manageable sections, a fuel dismantling cell was required. BNFL's considerable fuel dismantling and remote handling expertise, derived from around 40 years operations at Sellafield, proved particularly valuable in this task. Linking this know-how with US fuel storage technology enabled the design of a single building covering the whole process from fuel receipt to production of a fully sealed dry container holding a number of fuel sections.

BNFL radiometric instrumentation may also assist customers, for example benefits can be derived from the use of burn-up measurements to support the application of burn-up credit. Such credit is expected to offer considerable savings to storage, and to a greater extent transport, of spent fuel. These would be achieved through a reduction in storage cost per assembly from increased packing density or by reductions in the amount of neutron absorber required for the fuel basket. Increased packing density would also mean fewer transport journeys, again bringing lower cost per assembly. These examples illustrate how BNFL can exploit a wide range of know-how from other operations to assist TranStor™ and VSC users.

## 6. DEVELOPMENTS OF THE TRANSTOR™ SYSTEM

At present, technical resources are being focussed on meeting the licensing targets shown in Table II. Although development work is at a reduced level for now, it is still continuing; two recent patent applications may help to illustrate this:

- A double-containment canister design developed for the Chernobyl Dry Storage bid, Fig. 4. This concept is applicable in principle to any type of fuel, and could be used by any customer who wished to demonstrate an additional containment barrier (beyond the fuel cladding and the single-walled canister).

- A system to monitor the atmosphere inside the canister, but without breaking the containment barrier. Both BNFL and BNFL Fuel Solutions are fully convinced that the present double-lidded, multi-pass welding practice provides sufficient confidence in integrity of the canister seals. The recently developed ultrasonic weld testing technique only adds to an already satisfactory procedure. However, to provide the ultimate level of confidence in maintenance of 100% confinement integrity BNFL is developing a system for non-intrusive monitoring of the atmosphere inside the canister. Currently, the system has been demonstrated to work in the laboratory under non-radioactive conditions. The next stage will be tests on a larger scale, perhaps using the sort of arrangement shown in Fig. 5. The monitor is based on the ultrasonic detection of oxygen in helium, which would indicate air in-leakage.

## 7. CONCLUSION

The combination of US-developed storage systems with BNFL's existing technology affords complete flexibility to users in tailoring the systems to meet their own particular needs. Such flexibility already ranges from compact storage of non-LWR fuel to design of special fuel handling facilities and testing of radiation-resistant coatings. BNFL has, and will maintain, a policy of continuous enhancement to the current systems. Enhancements such as double-containment designs and non-intrusive monitoring are already emerging from the long-term development programme.

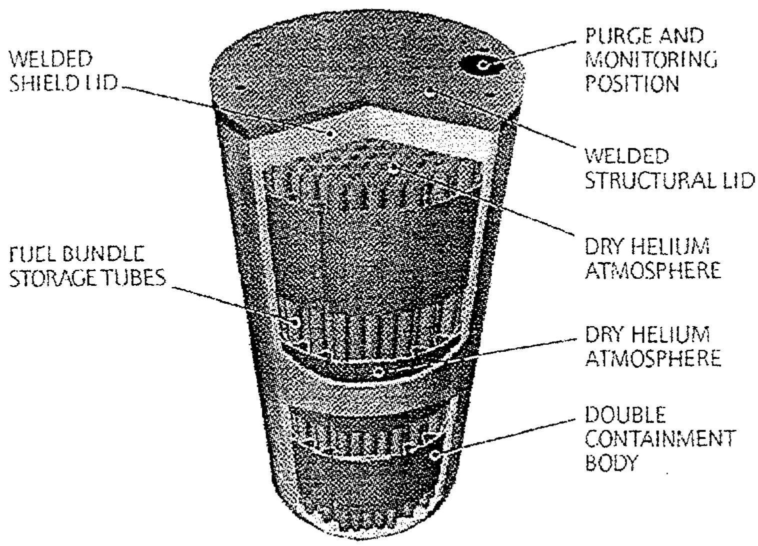


*FIG. 4. Fuel storage canister*

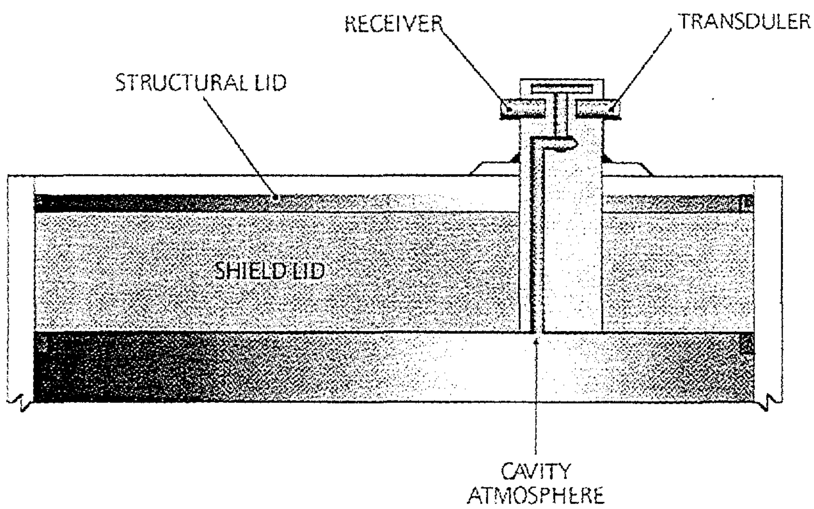


*FIG. 5. Non-intrusive canister atmosphere module*

IAEA-SM-352/26

# DUAL PURPOSE OR NOT? THE SIGNIFICANT FACTORS

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**Abstract**

The development of spent fuel storage systems requires consideration of many factors in making design decisions. A significant issue affecting the design is the need to incorporate transportability of the canister or cask system design, which results in major changes to the storage system design. This paper presents a review of the significant factors affecting storage system design to incorporate transportation requirements and looks at the trends in both the United States and Europe where Transnucléaire and its US affiliated companies Transnuclear Inc., Transnuclear West and PacTec are active. A discussion is also presented relative to the pros and cons of whether the spent fuel storage system vendor should anticipate these transportation needs in the design of their systems.

## 1. UNITED STATES EXPERIENCE

### 1.1. Factors affecting transportability decisions

The current spent fuel storage and transportation issues in the United States are driven by the need for nuclear power plants to free up space in their spent fuel pools. However, the design of these systems is influenced by many factors other than the simple requirement to store spent nuclear fuel outside the spent fuel pool in an independent spent fuel storage installation (ISFSI). One significant factor governing the design of today's systems is the ability to provide for transportation of the spent fuel as well as for storage. The transportability requirements are based on the current government position that all spent fuel will be taken by the DOE and stored at a central facility. This paper will examine the need for the design of spent fuel systems to anticipate these final DOE requirements and to anticipate the transportation requirements.

Dry spent fuel storage systems for use at commercial nuclear power plants are more and more required to allow for transportation of the spent nuclear fuel. This need is based on several factors:

- Compatibility with the DOE central repository requirements resulting in a desire to have a multi-purpose canister for on-site storage, transportation, and storage at the repository.
- Compatibility with the DOE central repository requirements so that the DOE will take the spent fuel canisters as-is with no need to repackage or re-handle the spent fuel.
- Public pressure wishing to make sure that the utility ISFSIs are not permanent storage facilities.

The need for compatibility with the DOE central repository requirements is governed by the previous DOE effort for the design of a multi-purpose canister system. The design requirements imposed by the DOE for this effort were that the system could be used for on-site-storage, for transportation to the repository, and finally for disposition at the repository. However, this effort was discontinued and the current DOE programme is resulting in a repository overpack that is significantly different from the canister systems in use today or that were designed as part of the original multi-purpose canister programme. Therefore, the design requirements for the system are now based on on-site storage and transportation to the repository. Compatibility with repository requirements is an unknown that is subject to change based on final DOE repository design.

The need for compatibility with the DOE system so that the DOE will take possession of the spent fuel without fuel handling or re-packaging is based on utility concerns that the DOE will take the fuel first from those utilities that have transportable systems. The utilities do not want to place spent fuel into storage and then be required to re-package the fuel prior to DOE acceptance. This philosophy is based on the demand that any fuel handling be performed by the DOE at the repository. Also, the potential for an interim storage facility would require a system that can be transported and then stored.

Finally, public pressure is driving a need to demonstrate that the utilities on-site ISFSIs are not permanent storage facilities. The use of dry spent fuel storage systems which are transportable shows that these ISFSIs are indeed temporary and that the spent fuel can be removed from the utility plant sites. The use of transportable systems provides the utility with the greatest perceived flexibility in dealing with the final DOE requirements for acceptance of the spent fuel.

These factors result in a market demand that the dry spent fuel storage systems include the ability for transport of the spent nuclear fuel. However, the transportability requirements come with a significant economic burden. The addition of transportability to the dry spent fuel storage system results in significant increases in system cost, licensing time, and system complexity. Also, the ultimate system design is based on "guesses" of the final repository needs. Fig. 1 shows the TN casks in use by the US utilities.

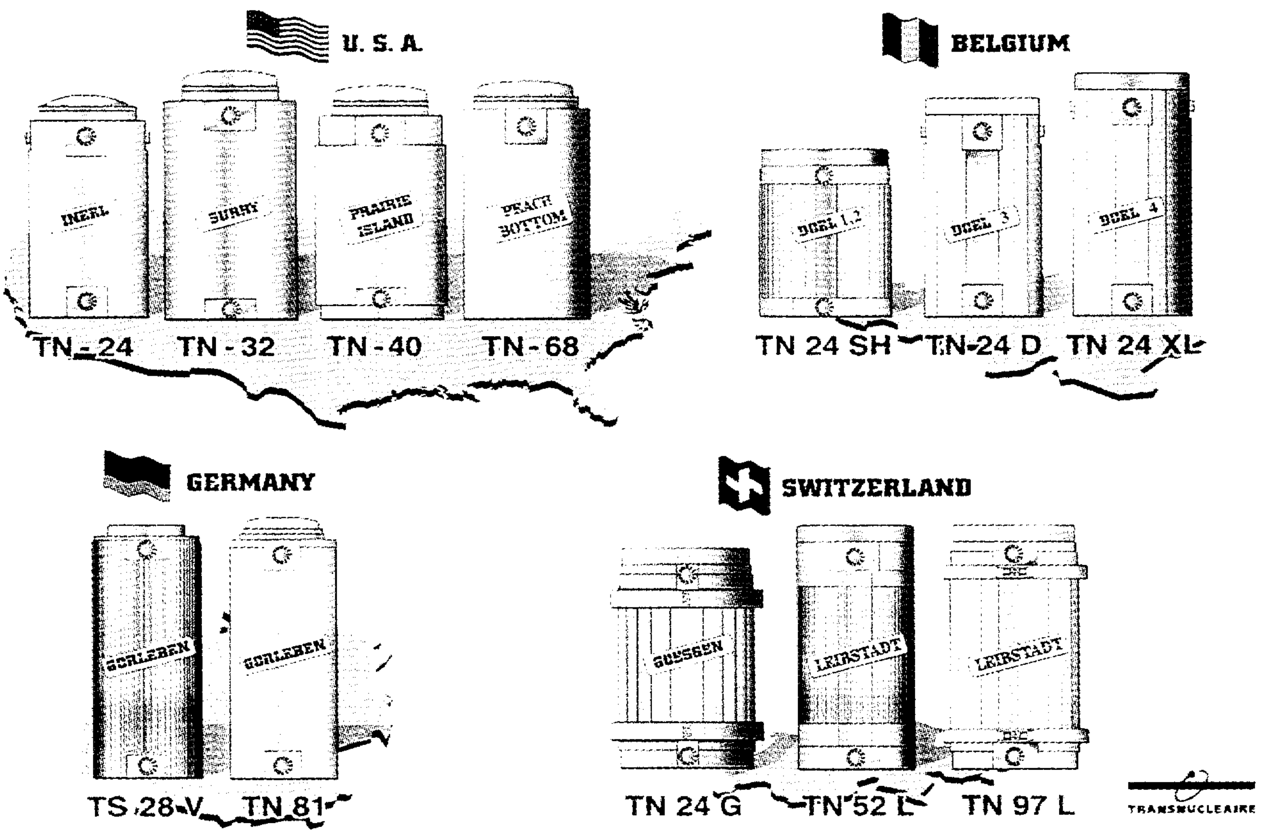


*FIG. 1. Transport and storage casks for many users*

## 1.2. Pros and cons of considering transportability effects

Based on the significant factors influencing the design of the spent fuel packages for transportation and the various unknowns with the final DOE requirements, should modular spent fuel storage systems anticipate transportation by themselves? There are pros and cons to this decision. The NUHOMS® MP187 system designed by Transnuclear West Inc. (TN West) provides an example of this approach an can be used to discuss the positive and negative aspects of developing a transportable system. The NUHOMS® MP187 system is a transportable system based on the NUHOMS® general licence storage system.

The positive aspects of developing a transportable system are:

- Compatibility with current utility and public demands to satisfy their requirements;
- Greatest flexibility for ultimately satisfying the DOE requirements whether the decision is for storage at a centralized interim facility or a repository;
- Opening the option to ship fuel to a reprocessing plant should this approach be reborn in the US;
- Added sturdiness to face the regulatory demands of 10CFR71;
- Added flexibility for a utility with multiple NPP units, so as to organize transhipments between pools or semi-centralized ISFSIs.

The negative aspects of developing a transportable system are:

- Licensing time is much longer since transportable systems require testing and obtaining separate licences for storage (10CFR72) and transportation (10CFR71);
- Design becomes more complex and costly.

The MP187 system provides a good example of the pros and cons stated above. The MP187 canister is similar in design to the general licence storage system in function, operation, and overall size. However, the MP187 canister requires additional hardware in that 26 spacer discs are required for structural support versus eight for the storage only system. Also, the MP187 canister requires the use of neutron poison in the design, which is not required by the storage only system. These changes result in a canister that requires additional fabrication work and which has a higher cost.

The MP187 system cask is required to meet transportation requirements of 10CFR 71 which results in additional hardware, testing, and design requirements. The transportation cask includes the need for impact limiter designs and testing to demonstrate the ability to handle the 9-meter drop requirement. Additionally, the cask must satisfy more stringent shielding and dose rate limits and must be capable of maintaining leak-tightness. These design features result in added fabrication complexity and cost.

Finally, the licensing period is increased due to the more complex nature of the transportation design. The MP187 system has been in licensing for over four years, which encompasses the initial submittal, testing, and final licensing submittals. It is currently the only fully licensed canister-based dual-purpose system in the world.

Similar considerations apply to the cask based interim storage systems such as those developed by Transnuclear Inc. Although the basic design features of storage only casks, like the TN 40 or the TN 32, are close to those of a dual purpose cask, the effort for licensing the dual purpose TN 68 for BWR fuel is greater (see Fig. 2 for its storage configuration). This effort will give to the customer more options in its fuel management policy.

Therefore, the design to provide transportability into the dry spent fuel storage system requires a review of the practical and economical costs/benefits of this approach.

### 1.3. Current trends

The current practice in the United States is moving toward dual certified systems – systems that are licensed for storage (10CFR72) and transportation (10CFR71). This practice is driven by the need for utilities that are decommissioning power plants to demonstrate transportability of the systems and by a general desire to have the flexibility that dual certified systems provide. Also, more dual-purpose systems are completing the licensing process and this will result in a further shift toward the use of the dual certified systems.

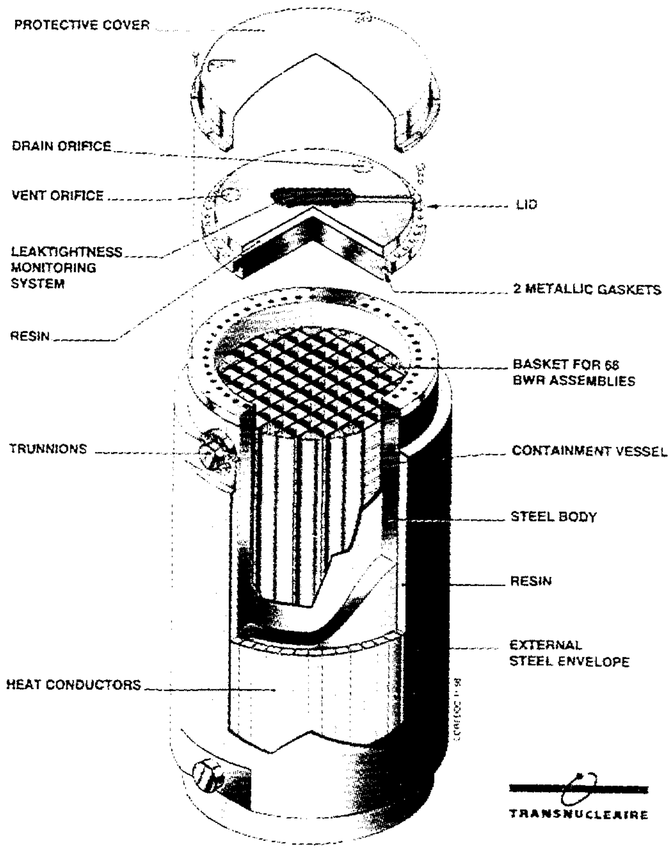


*FIG. 2 Storage configuration of the TN 68 cask*

## 2. EUROPEAN EXPERIENCE

### 2.1. Factors affecting transportability decisions

Europe, with the notable exception of Sweden, is characterized by several parameters that explain why interim storage has been implemented in transportable systems. Most European utilities have been contracting for reprocessing of their spent fuel. This has two major consequences:

- The power plants are equipped to handle transport casks and their operators are familiar with that type of equipment;
- Even though a utility may want to place part of its spent fuel in interim storage, it does not want to forego the possibility of sending that very fuel to reprocessing later in time. Therefore it will choose not only transportable interim storage systems, but also systems that can be unloaded at the reprocessing plants too.

Other reasons reinforce this choice. Some countries have built centralized interim storage facilities and it is easier to load the fuel into dual purpose casks at the NPP rather than transport it to such facility or organizing for transfer from a transport device into a storage system at the end of the road. Also, in most European countries, laws that govern interim storage are the same laws and regulations that apply to NPP and other nuclear industry facilities. Unlike the US 10 CFR 72, these laws and regulations give scant *specific* guidance to the ways one should regulate the interim storage facilities. Hence the strongly deterministic and conservative regulations for transport of radioactive materials provide a first and firm basis on which to license an interim storage system.

In addition, unlike in the USA, transportation of spent fuel has been ongoing for years in Europe. The licensing practice is well established and shared by Competent Authorities from the different countries. The licensing effort/cost gap that exists between storage only and dual purpose systems is thus narrower.

Companies like Transnucléaire, that have initially developed in the field of transport services and transport packaging engineering, have thus naturally become suppliers of dual purpose casks such as the casks from the TN 24 cask family. These dual-purpose casks have a versatile design that is readily adaptable to various spent fuel assembly characteristics and different power plants, as shown by Table I. Fig. 1 shows the various casks in use in Europe and Fig. 3 the storage configuration of the cask TN 52 L.

TABLE I. THE TN 24 DUAL PURPOSE CASK FAMILY

| Name of cask | Contents | Country |
|---|---|---|
| TN 24 D | 28 PWR 900 SFAs | Belgium |
| TN 24 XL | 24 PWR 1300 SFAs | Belgium |
| TN 24 G | 37 PWR SFAs | Switzerland |
| TN 52 L | 52 BWR SFAs | Switzerland |
| TN 97 L | 97 BWR SFAs | Switzerland |
| TN 24 DH | 28 PWR 900 high burnup SFAs | Belgium |
| TN 24 XLH | 24 PWR 1300 high burnup SFAs | Belgium |
| TN 24 SH | 37 PWR 14x14 high burnup SFAs | Belgium |

Another element in favour of dual-purpose system in the European view is that the international character of the transport regulations based on IAEA recommendations makes it easier to procure from several vendors based in different countries by holding them to a well identified set of specifications.

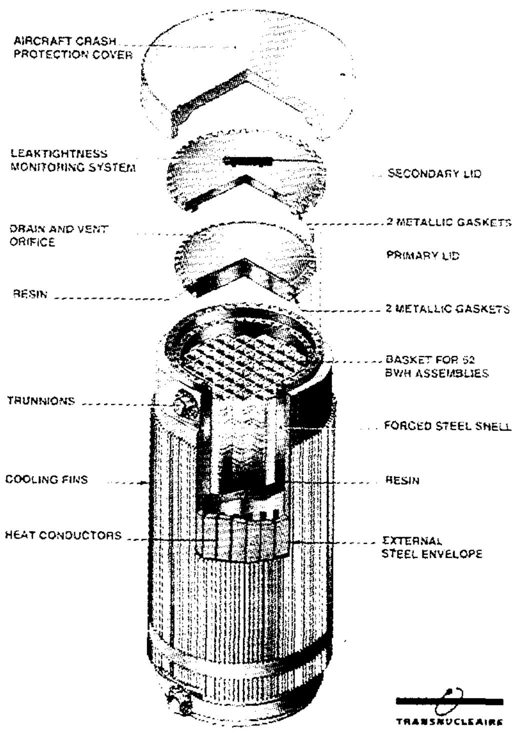


*FIG. 3. Storage configuration in the TN 52 L cask*

TABLE II. INTERIM STORAGE NEEDS IN EUROPE

| Country | Type of interim storage for spent fuel |
|---|---|
| Belgium | • Dry storage in **DP** casks at Doel NPP<br>• Pool storage in Tihange NPP |
| Bulgaria | • Pool storage<br>• Dry storage planned-undefined |
| Czech Republic | • Dry storage in **DP** casks |
| France | • Pool storage at reprocessing plants |
| Germany | • Centralized dry storage facilities with **DP** casks |
| Hungary | • Vault system + shipment of spent fuel to Russia |
| Italy | • Dry storage **DP** casks planned |
| Netherlands | • Vault system |
| Slovakia | • Pool storage<br>• Planned dry storage in **DP** casks |
| Spain | • Planned dry storage in **DP** casks |
| Sweden | • Centralized pool storage |
| Switzerland | • Dry storage in **DP** casks |
| UK | • Pool storage at reprocessing plants |

European countries are, by American standards, very small in area. So opposition against to and/or concern with the storage may well come from the public of a neighbouring country, as the Austrian example of pressures on the Czech and Slovak Republics has shown. Thus, transportability serves two purposes, showing that:

- the fuel will not be stored forever on the given spot, since it is transportable;
- a stringent safety approach based on internationally recognised rules is being implemented. Thus the experts of the neighbouring country may assert readily that the interim storage facility is up to valid standards.

### 2.2. Current trends

The situation of the current additional interim storage needs for spent fuel in Europe is summarized in Table II. Exempting the two countries that benefit from the large pools of their reprocessing plants, one can see that 70% of the listed countries implement or plan to implement dry storage in dual purpose systems.

## 3. CONCLUSION

While in Europe, transportability dominates the interim storage systems, in the US utilities still need to ponder on their choice. It may well be that as the safety justification and QA requirements for storage systems associated to higher performances and fuel burnup become ever more stringent, the gap between transportability and storage-only will narrow and lose significance. In Europe, the medium or long-term vision of multinational end-disposal facilities should add momentum to the transportability requirement.

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# ON-SITE STORAGE OF SPENT NUCLEAR FUEL ASSEMBLIES IN GERMAN NUCLEAR POWER PLANTS



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**Abstract**

The selection of back-end strategies for spent fuel assemblies is influenced by a number of different factors depending on the given situation in any specific country. In Germany, the back-end strategy implemented in the past was almost exclusively reprocessing. This strategy was required by the German Atomic Energy Act. Since 1994, when the Atomic Energy Act was amended, the option of direct final disposal has been granted the equivalent status by law to that afforded to reprocessing (and reuse of valuable materials). As a result, German utilities may now choose between these two alternatives. Another important condition for optimizing the back-end policy is the fact that fuel cycle costs in Germany are directly dependent on spent fuel volumes (in contrast to the US, for example, such costs are related to the amount of power generated). Another boundary condition for German utilities with respect to spent fuel management is posed by the problems with militant opponents of nuclear energy during transportation of spent fuel to interim storage sites. These facts have given rise to a reconsideration of the fuel cycle back-end, which has resulted in a change in strategy by most German utilities in favour of the following:

- Preference for long-term storage and maximized use of on-site storage capacity;
- Reduction in the amount of spent fuel by increasing burnup as much as possible.

These decisions have also been driven by the deregulation of energy markets in Europe, where utilities are now permitted to sell electric power to consumers beyond their original supply network and must therefore offer electric power on a very cost-competitive basis.

## 1. INTRODUCTION

Germany currently has 19 operational nuclear power plants with an installed generating capacity of 20,000 MWe, all designed and built by the Power Generation Group (KWU) of Siemens AG. Six of these plants are boiling water reactor (BWR) plants, while 13 are of pressurized water reactor (PWR) design. Together, these units produce some 35% of the total electric power generated in Germany. The performance of German nuclear power plants has been very satisfactory. In recent years, seven of these plants have ranked among the top ten world-wide in terms of power generated.

Commercial use of nuclear energy in Germany began in 1961 when Kahl Experimental Nuclear Power Station was handed over to the plant operator. All told, German nuclear power plants have produced 2.5 billion kW·h and thereby reduced $CO_2$ emissions to the atmosphere by 2 billion tons.

At present there is no demand for construction of new power plants in Germany. Since 1980, the amount of power generated has actually decreased by some 4% every five years, while over the same period the share of power generated by the nuclear power industry has increased by approximately 10% thanks to upgrading and better availability, etc.

From the beginning of commercial nuclear power generation in 1961 up to the end of last year, some 7,400 tU of spent fuel have been discharged from the reactor cores of German nuclear power plants. About 4,500 tU have been sent to COGEMA for reprocessing, 600 tU to BNFL, 100 tU have been reprocessed in Germany and 2,317 tU are stored on site in the plants' existing fuel pools. Germany has two interim spent fuel storage facilities, Ahaus and Gorleben, which together provide a total storage capacity of 8,400 tU of spent fuel, of which 100 tU is currently occupied. Dual-purpose transport and storage casks provide interim storage for these away-from-reactor storage facilities now in operation.

Prior to the recent revision of the German Atomic Energy Act, reprocessing of spent fuel and recycling of valuable reusable material constituted the mandatory back-end strategy. Long-term interim storage and subsequent direct disposal of spent fuel are now a legal option for back-end strategy. Some German utilities have therefore decided to terminate their reprocessing programmes.

According to a study conducted by of the Association of German Utilities (VdEW), a general shift in back-end strategy from reprocessing to direct disposal will result in annual savings of some DM 40 million each year per 1000-MWe nuclear power plant unit.

Moreover, political difficulties of a considerable scope have arisen in Germany over the shipping of spent fuel, in particular to the away-from-reactor storage sites, with significant public protest spear-headed by militant opponents of nuclear power generation. As a result, consideration has been given to expanding on-site storage capacity and decreasing spent fuel discharges by increasing enrichment and thereby raising discharge burnup.

## 2. DEVELOPMENT OF DEMAND FOR ON-SITE STORAGE CAPACITY IN NUCLEAR POWER PLANTS

Among designers of older-generation nuclear power plants (i.e. those units which began operation before 1974), it was originally assumed that spent nuclear fuel assemblies would be transported away from the nuclear power plant as quickly as possible, immediately reprocessed and the recyclable residual materials reused as quickly as possible in new fuel assemblies. Basis for the reasoning behind these beliefs was the assumption that prices for natural uranium and enrichment services would continually rise. However, with the exception of a brief period during and shortly after the 1st oil crisis in 1972 and 1973, this assumption proved to be false. The price for uranium in 1968, for example, was approximately US$ 6.50 per lb of $U_3O_8$. Today, in 1998, the price 30 years later lies between US$ 8 and 9 lb per $U_3O_8$. In 1968, it was forecasted that uranium prices in 1998 would reach some US$30/lb$U_3O_8$. At the time, the predicted developments in prices for uranium and enrichment processes justified a policy of quick recycling.

As a consequence of this philosophy, plant on-site storage was limited to a capacity which could accommodate the current core load and, at the very most, 2 to 3 reload batches. A non-availability of reprocessing capacity in the 1970s then led to the decision to increase on-site storage capacity. Those power plants which began operation beginning in the mid-1970s where then provided with storage capacity which allowed operation to continue for seven years until removal transport of spent fuel assemblies became necessary.

The German Atomic Energy Act stipulated at that time that - with but few exceptions - spent nuclear was to be reprocessed and the residual fissile materials recycled (uranium and plutonium). In supplement to this required reuse stipulated by the Atomic Energy Act, a regulation came into effect in February 1980 stating that nuclear power plants, from commissioning onwards, must continually provide proof on an annual basis, year for year, that storage of irradiated nuclear fuel assemblies was ensured in advance for an operating time period of six years either at the plant operator's own licensed facilities or through binding contracts for interim storage or reprocessing.

This ordinance led on the one hand to further increases in on-site storage capacity at German nuclear power plants through implementation of compact storage, while provisions for external, central spent fuel storage facilities were also initiated. The suitable concept at that point in time for external spent fuel storage facilities was spent fuel shipping cask storage of dual-purpose casks, i.e. for casks licensed for both shipping and storage.

While work on developing and licensing these dual-purpose casks advanced (the Castor design, for example), planning and licensing of the actual storage facility were also pursued. With the licensed cask concepts and a licensed storage facility, the call for storage provisions was able to be met in a very satisfactory manner.

The relatively simple storage building was licensed in the course of time and erected. As the casks could be manufactured in a short period of time, it was planned to order casks when and as needed. From then on, investigations required for the more long-term dry storage concept and experimental testing of spent fuel assemblies to determine storability were addressed with increased vigour.

At the same time, it was also decided to direct efforts to research and development as well as prototype testing in preparation for direct final disposal. One result of this decision is the pilot conditioning plant at Gorleben, designed for conditioning of spent fuel for direct final disposal. Studies were begun simultaneously in the immediate vicinity of the interim storage facility and the pilot conditioning plant at Gorleben to investigate a final repository site in the salt dome located there.

Beginning in the mid-1980s, companies in neighbouring countries such as COGEMA began to offer nuclear fuel reprocessing services once the UP-2 plant had been converted for reprocessing of light water reactor fuel assemblies. German utilities played a significant role in financing the UP-3 plant, which has a capacity of 800 tU/a and has been in operation since 1989. BNFL's nuclear fuel reprocessing facility, THORP in Sellafield, England, then became available as of the mid-1990s, with a reprocessing capacity of 1,000 tU/a. All told, German utilities reached agreements within the framework of the so-called baseload contracts with COGEMA and BNFL covering a reprocessing volume of 8,500 tU.

A new legal situation with respect to the disposal of spent nuclear fuel assemblies came into being at the end of 1994. At that time, new legislation placed direct final disposal on a legal par with reprocessing. From the end of 1994 on, German utilities could chose between these two options. Some utilities subsequently cancelled their existing contracts (new contracts) as of December 31, 1994, and chose the path of interim storage and direct final disposal.

This situation and the increasing difficulties encountered in particular since 1996 when shipping spent fuel assemblies to interim storage, led by a militant wing of protesters, have led to new considerations to increase yet again the on-site storage capacity at nuclear power plants.

## 3. FURTHER EXPANSION OF ON-SITE STORAGE CAPACITY IN GERMAN NUCLEAR POWER PLANTS

In addition to compact storage, other more advanced techniques are available of expanding storage capacity. The most important of these are the following:

- Two region storage;
- Consolidation of spent fuel.

### 3.1. Two region storage

An irradiated fuel assembly contains far less than the half the original fissile material contained in a fresh fuel assembly. The originally high radioactivity decreases considerably within the course of core residence time. In addition, build-up of fission products and actinides in the fuel further reduces reactivity [1].

It was assumed for earlier storage techniques that fuel assemblies of the highest reactivity could be stored in all storage positions. This requires not only neutron-absorbing material between the fuel assemblies, but a sufficiently large water gap as well. Contrary to this, spent fuel assemblies require less absorber material owing to the lower radioactivity, and can additionally be stored in a considerably tighter arrangement while remaining safely and reliably subcritical.

Advantage can be taken of these conditions by dividing the fuel pool of nuclear power plants into two regions. Region 1 serves to store high radioactivity fuel assemblies. In this region, the fuel assemblies of the current reactor core load and new fuel assemblies are stored for reload actions. The remaining available storage space is filled by the storage positions of Region 2. The rack positions in the Region 2 store are arranged in a tighter configuration, with smaller center-to-center spacing of the fuel assemblies, and have less absorber material between the fuel assemblies (Fig. 1). This configuration allows storage capacity to be increased considerably. Typical scopes of such increases amount to a doubling of storage volume of spent fuel assemblies.

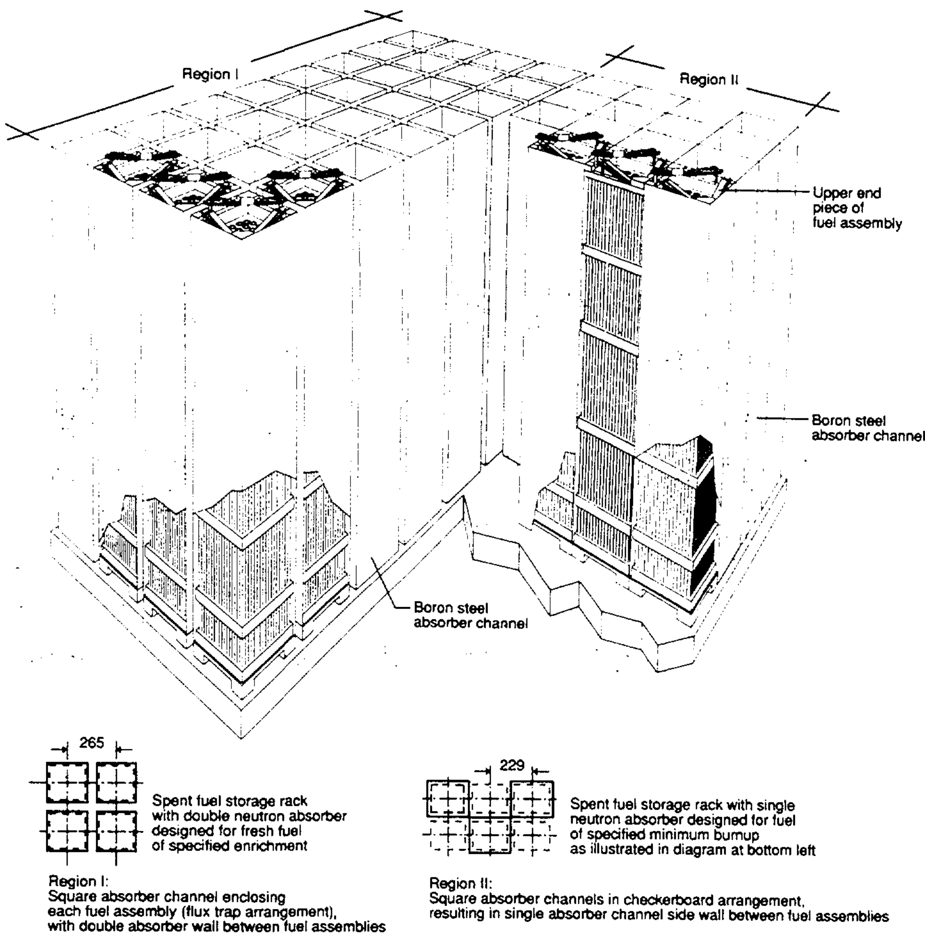


*FIG. 1. Two region high density storage racks with boron SS neutron absorbers*

In almost all nuclear power plants, the higher fuel pool bottom loads and the increased fuel assembly cooling requirements owing to higher decay heat are covered by existing design reserves. Having implemented this storage concept already at eleven nuclear power plants outside Germany, licensing procedures have recently begun in Germany for introducing this method of increasing storage capacity to several German nuclear power plants as well.

A portion of the existing storage racks are used for Region 1. The remaining older racks are dismantled and replaced with Region 2 storage racks. This concept of two-region storage will be implemented at PWR plants. Given a postulated erroneously storage of high-radioactivity fuel assemblies in Region 2, the boric acid already provided in the fuel pool would ensure adequate subcriticality in any situation, in line with the double contingency principle.

Backfitting of the fuel pool is performed during reactor operation according to an exact, predetermined erection sequence. The existing storage racks and the new ones can be mounted under water by means of remote-controlled hoisting gear and handling tools. Newly developed and verified computer codes allow very precise determination of the dynamic behaviour of these storage racks, for example under earthquake conditions. This also permits better utilization of existing storage space, among other factors by reducing the gap between the storage racks and the fuel pool walls.

### 3.2. Consolidation of spent fuel

Fuel rods make up less than half the total volume of a fuel assembly. As an additional measure for increasing storage capacity in the fuel pool and optimizing fuel cycles, spent fuel can be consolidated in nuclear power plant fuel pools. In this method, the spent fuel rods are removed from the fuel assembly and packed together at the highest possible density. In this way, the fuel rods of two fuel assemblies are consolidated to the volume of one fuel assembly, thereby doubling the storage capacity of existing facilities.

The techniques used for implementation of consolidation are already used extensively for nuclear fuel services (for re-installing irradiated fuel rods in new fuel assembly skeletons, for example). The fuel rods, packed together at the highest density possible, are inserted into canisters which can then be handled and stored similarly to normal fuel assemblies (Fig. 2).

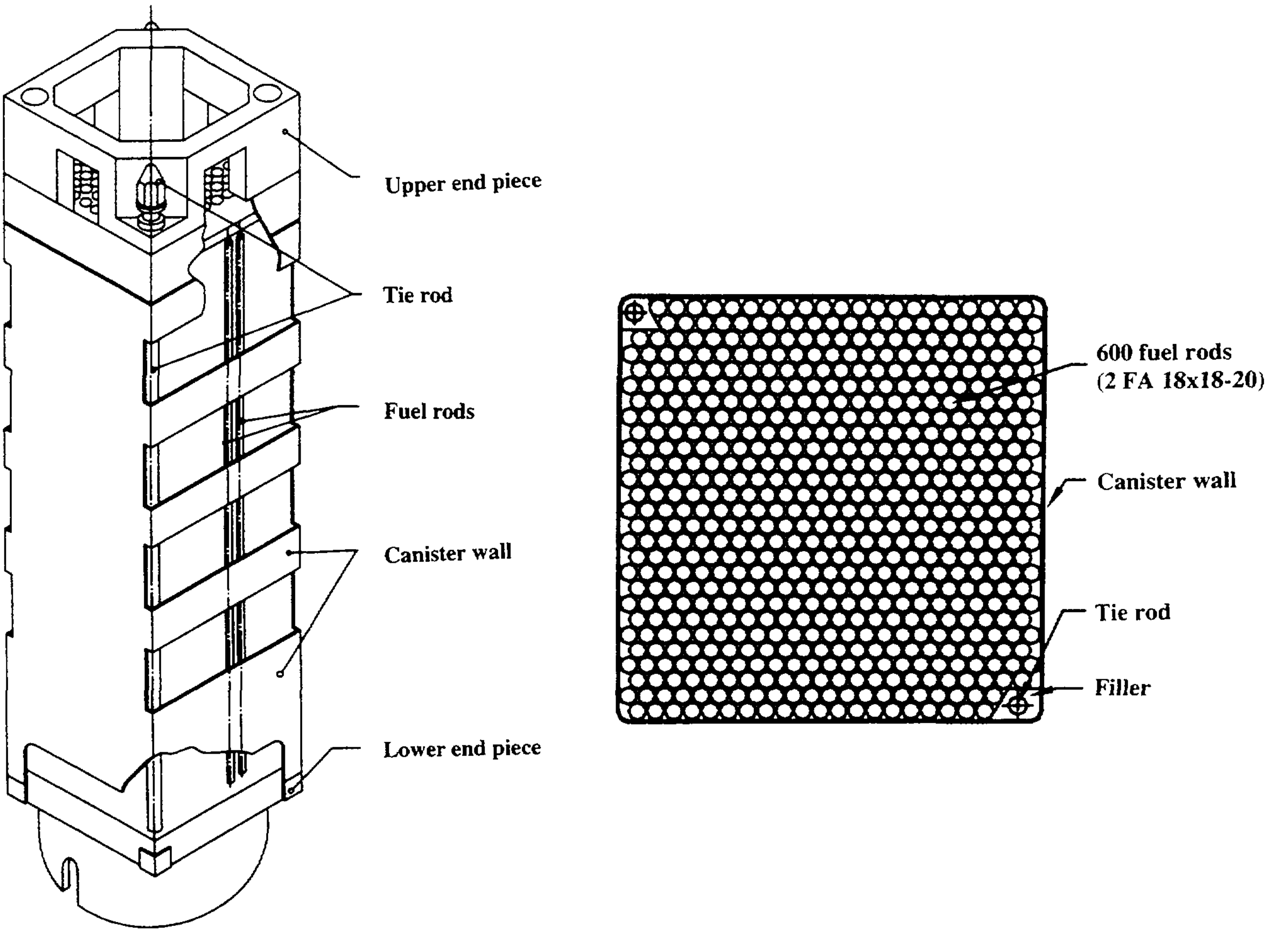


*FIG. 2. Canister with consolidated fuel rods and cross-section*

These canisters can be designed with openings for cooling, but there also exist concepts in which the canisters, subsequent to drying the inside volume, are seal-welded closed. The resultant capsules, designed for long-term interim storage, thus fulfil additional barrier functions, in certain cases right up to final storage.

The fuel assembly skeletons remaining subsequent to fuel rod consolidation can be shredded and consolidated in the fuel pool and stored in appropriate canisters for disposal. New fuel assembly racks, for example those intended for two-region storage, are designed from the start to accommodate the higher loads arising due to consolidated fuel assemblies.

## 4. INCREASING DISCHARGE BURNUP

Virtually all nuclear power plants in Germany have made efforts in recent years to substantially increase discharge burnup. The reason for this are the considerable savings achieved thereby in fuel cycle costs (Fig. 3). Not all countries place such importance on increasing discharge burnup. This is due, among other reasons, to the fact that back-end costs in different countries are calculated in different ways. In the United States, utilities pay fees for the back-end of the nuclear fuel cycle to the Department of Energy, the figure paid depending on the amount of power generated (cents per kW·h).

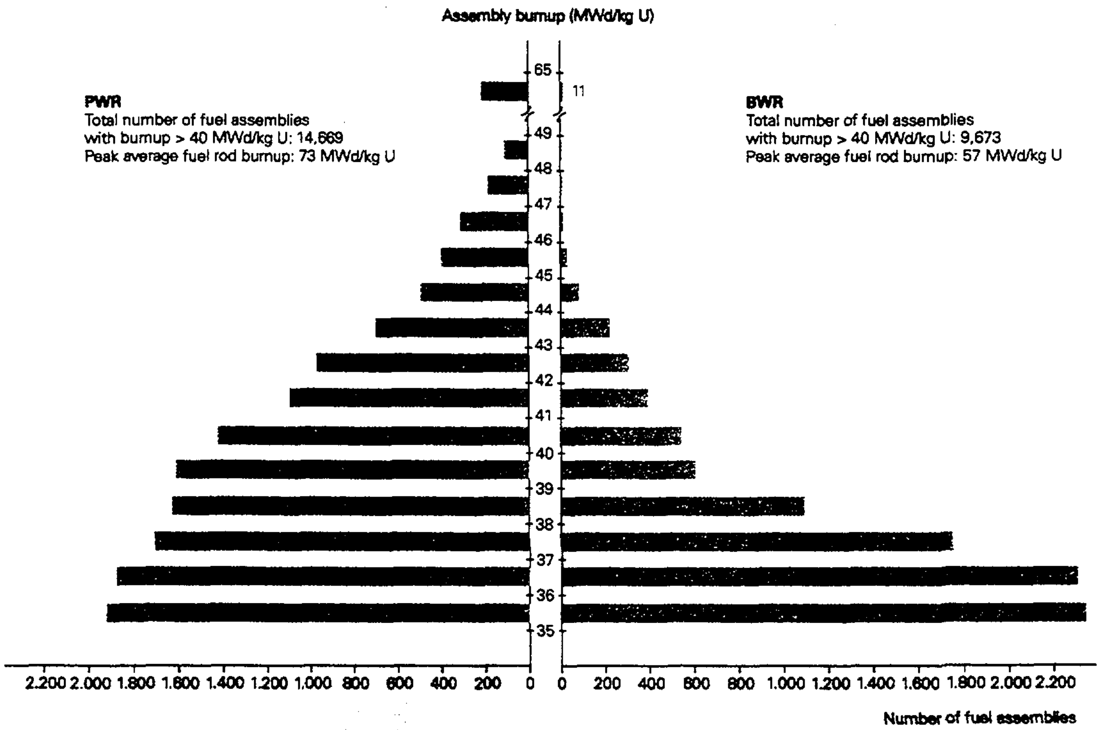


*FIG. 3. Burnup distribution of SIEMENS fuel assemblies with higher burnup (Status: December 1997).*

By contrast, the costs incurred by utilities in Germany are proportional to the quantity of discharged spent fuel (DM per kgU). As a result, US operators - for reasons of cost optimizing - devote greater attention to reducing operating costs (thus the 18-month cycle, for example), while discharge burnup is not of primary concern. In Germany the situation is the opposite, which translates into shorter operating cycles (1 year, and sometimes shorter). Disharge burnups are correspondingly higher and thus discharge volumes lower [2].

In Germany, therefore, a discharge burnup increase by some 10 GW·d/tU in a Convoy-series plant results in cost savings of approximately DM 30 million annually (Fig. 4). Siemens, as a major supplier of fuel assemblies, has therefore been devoting considerable effort for some time to developing fuel assemblies suitable for high burnups. Pathfinder fuel assemblies have been already been in operation for some years now, providing verification of operating reliability (Fig. 3). Parallel to this, nuclear power plant operators are implementing the upgrades required of plant systems in order to achieve these higher burnup.

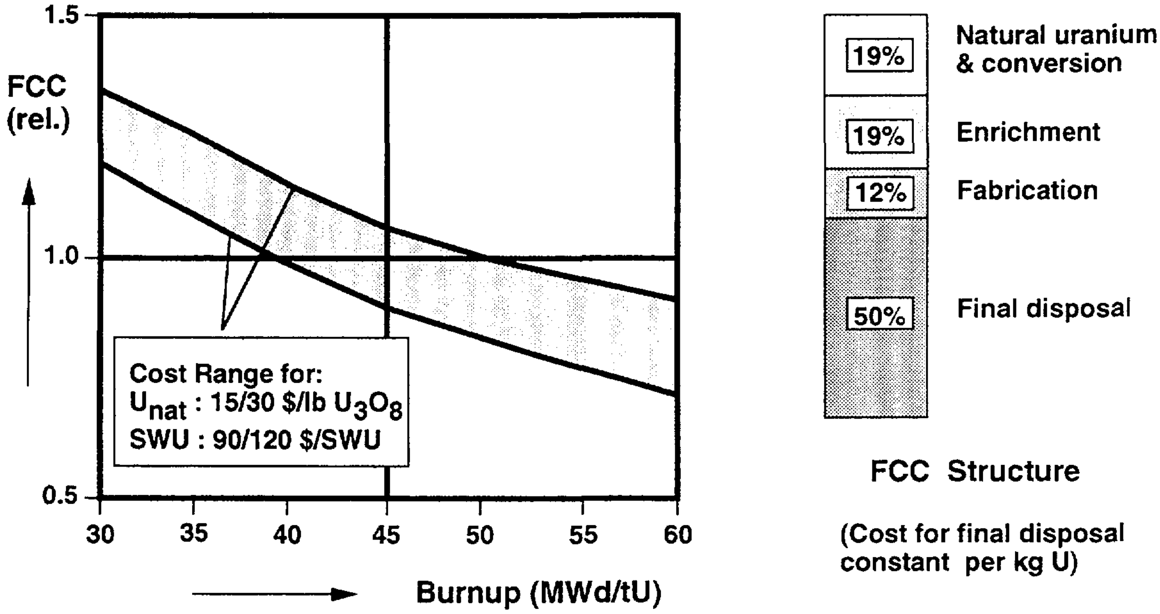


*FIG. 4. Fuel cycle cost as function of burnup*

Since increasing burnup is also directly related to increase in fuel enrichment, certain measures are required to allow handling and storage of these more highly enriched fuel assemblies. Depending on the specific nuclear power plant such measures may also affect, for example, the storage of new fuel assemblies and the fuel pool. The reactor coolant system may also be affected, owing for example to the required increase in boric acid efficiency.

The handling and storage of spent fuel assemblies with higher burnup in nuclear power plant fuel pools poses no particular problem. Adaptation measures must be implemented in the case of burnup of 60 GW·d/tU and higher for external shipping and away-from-reactor storage facilities.

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# THE LONG TERM STORAGE OF ADVANCED GAS-COOLED REACTOR (AGR) FUEL

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**Abstract**

The approach being taken by BNFL in managing the AGR lifetime spent fuel arisings from British Energy reactors is given. Interim storage for up to 80 years is envisaged for fuel delivered beyond the life of the Thorp reprocessing plant. Adopting a policy of using existing facilities, to comply with the principles of waste minimisation, has defined the development requirements to demonstrate that this approach can be undertaken safely and business issues can be addressed. The major safety issues are the long term integrity of both the fuel being stored and structure it is being stored in. Business related issues reflect long term interactions with the rest of the Sellafield site and storage optimisation. Examples of the developement programme in each of these areas is given.

## 1. INTRODUCTION

British Nuclear Fuels (BNFL) has been contracted to manage the lifetime irradiated AGR fuel arisings from British Energy reactors[1]. The agreement formulated is a mixture of reprocessing (covering the planned life of the Thorp reprocessing plant) and interim storage for the remainder of the fuel arisings. Interim storage is projected to be up to 80 years to comply with direct disposal acceptance criteria and projected repository availability. Eighty years represents a significant increase in storage times compared to current operational experience; of around 18 years. Confidence that AGR fuel can be stored safely for extended periods has been provided by our experience of storing AGR fuel to date and the supporting research and development programmes initiated in the late 1970's for wet storage and 1990's in the case of Scottish Nuclear (SNL) dry storage project.

AGR fuel elements comprise 36 stainless steel clad fuel pins, containing uranium dioxide fuel pellets, are held together by stainless steel braces enclosed in an open ended graphite sleeve which acts as part of the neutron moderator. Normally 8 fuel elements (7 in the case of Dungeness NPP) are held together by a tie rod running through the central tube of each fuel element to make up what is referred to as an AGR stringer. After irradiation the stringer is dismantled into individual fuel elements before being wet stored in fuel skips.

The main differences between AGR long term storage as proposed as part of the SNL dry storage project and BNFL (Sellafield) taking on such a contract are:- SNL was limited by reactor site operating licences (which do not allow the transfer of fuel between sites, and therefore the need for a store at each reactor site), ARTICLE 37[2] effectively does not allow the removal of graphite sleeve, (and simplification of the dry store technology) as this would be viewed as waste, and existing storage facilities are limited. In comparison BNFL (Sellafield) already manages fuel from national and international reactors prior to reprocessing, operates fuel dismantling and associated waste store facilities and has three existing large pools (AGR Storage Pond, Fuel Handling Plant and Receipt & Storage) which are utilised for AGR fuel storage.

## 2. PROPOSED STRATEGY

Figure 1 outlines the available storage options and the main Pros and Cons of each. Based on compliance with the principles of waste minimisation and the avoidance of high initial capital

[1] British Energy reactors comprise of the former generating companies Nuclear Electric and Scottish Nuclear Limited.
[2] ARTICLE 37 of the EURATOM Treaty is plant specific, it relates to the impact of operations on neighbouring member European states. Changes to ARTICLE 37 would require reapplication.

expenditure the approach to be taken in the first instance is to use existing facilities and storage techniques; i.e. wet storage. This approach dictates development requirements to demonstrate that long term storage can be carried-out safely and to resolve business related issues. These can basically be divided into two main development areas:- Fuel Integrity and Pool Storage Management.

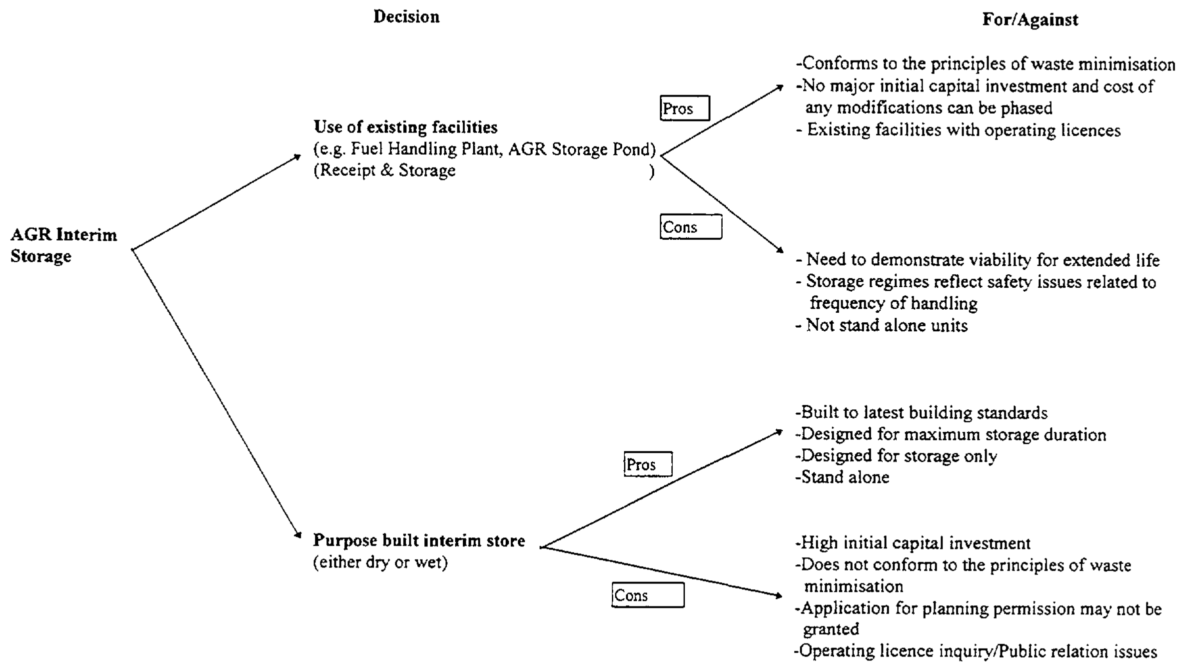


*FIG. 1. Available storage options*

If the initial option becomes untenable on safety grounds the alternative of constructing a purpose built facility will be taken. The decision then will either be to go down the existing wet technology route or to develop a variation of the SNL dry store technology.

## 3. FUEL INTEGRITY

Spent AGR fuel cladding performs two functions; as primary containment barrier, and for mechanical handling of individual fuel pins for rod consolidation/reprocessing purposes. The final recovery and conditioning of the fuel after 80 years storage in principle is the more restrictive of the functions, based on a minimum requirement of half original wall thickness, compared to localised cladding perforations which can be resolved by encapsulation.

Figure 2 outlines the bounding cladding thickness, the margin between post irradiation and minimum clad thickness to allow for mechanical handling, and provides a very simplistic comparison with the long term wet storage of BWR fuel. Whilst wet storage of zircaloy clad fuel can be considered to be unlimited, in the absence of any other failure mechanism except general corrosion, AGR wet fuel storage is limited to a maximum predicted to be ~152 years or even less (see below).

It has been well reported [1-3], that irradiated AGR fuel elements 1-5 of the original irradiated stringer are known to be susceptible to irradiation induced intergrannular stress corrosion cracking of the stainless steel fuel cladding and structural components. To inhibit this failure mechanism AGR fuel is stored in pool water dosed with sodium hydroxide to pH 11.5. Sodium hydroxide was chosen as a result of a corrosion inhibitor development programme undertaken early 1980s and has been used since 1986 for the interim storage of all AGR fuel at Sellafield. Operational experience to date indicates that fuel cladding perforation has been totally prevented.

The technical case for the storage of AGR fuel for up to 80 years is reliant upon the continued use of corrosion inhibitors. With changes to the front end of the fuel cycle, such as increasing fuel burnup, combined with a significant increase in proposed storage duration there is a need to revisit the original corrosion development work. One aspect of AGR fuel corrosion to be investigated as part of the development programme is the underlying corrosion rate of stainless steel in sodium hydroxide to pH 11.5.

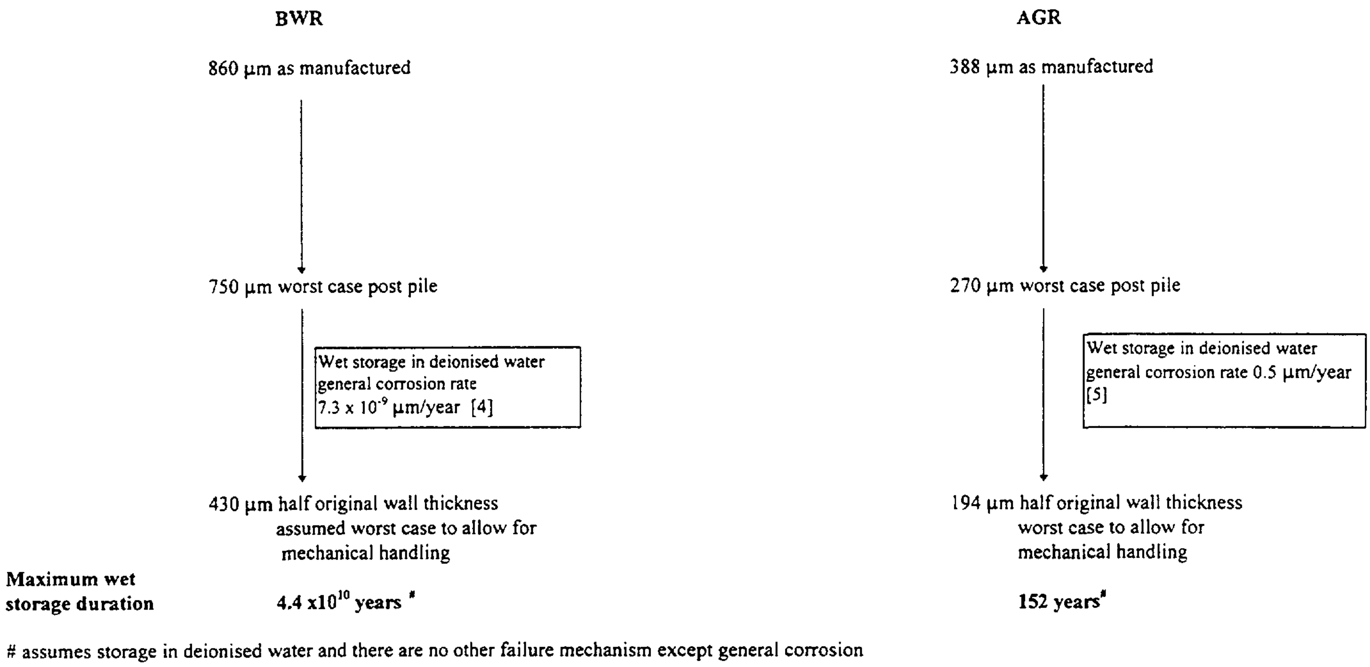


*FIG. 2. Simplistic comparison of maximum wet storage duration*

During the original corrosion inhibitor development programme it was noted that there was an additional anodic (corrosion) reaction occurring in caustic solutions. Limited work was undertaken which identified that this reaction was associated with enhanced dissolution of the surface oxide layer. As this reaction was not associated with localised corrosion, and because pool storage duration prior to reprocessing was typically ~ 10 years it was not considered to constitute a threat to the integrity of either the cladding or braces and was therefore never quantified.

Given the current storage duration being proposed, such low corrosion rates could ultimately result in significant reduction in the cladding thickness. A Direct Current Potential Drop technique Field Signature Method (FSM) is currently being used to establish the general corrosion rate in sodium hydroxide; the principles of this technique are given in Appendix 1. Given a minimum resolution of 0.05% of metal thickness, a monitoring period of around a year (~ 0.19 μm limit of detection over measuring period) will be enough to establish whether general corrosion in sodium hydroxide is an issue. In the event of general corrosion in sodium hydroxide not meeting the safety criteria then alternative corrosion inhibitors will be investigated in the first instance.

In the longer term some form of condition monitoring during the long term storage of AGR fuel is required; both the condition of the fuel pins and the fuel element braces needing to be assured. Currently the integrity of dismantled AGR fuel stored in the AGR Storage Pond is monitored by means of an activity release model. The technique is retrospective and the application of non destructive techniques such as Electrochemical Noise (see Appendix 1) and FSM to give predictive information will be investigated as alternative methods.

## 4. POOL STORAGE MANAGEMENT

The use of existing facilities primarily raises two issues:- Firstly the need to provide confidence in the integrity of existing facilities for such extended periods of time. Secondly the fuel storage facilities were designed as buffer facilities for reprocessing activities; i.e. high throughputs and reliance on the site infrastructure. The following development programmes have been initiated to address these issues.

### 4.1. Structural integrity

Structural integrity of the existing facilities is crucial to a re-use policy. All ancillary plant items such as cranes, electrical wiring, roofing etc. can all be either refurbished, up-graded or replaced. Reinforced concrete is prone to the following deterioration mechanisms:

- **Direct corrosion of the steel reinforcement**
  Via cracking and spalling concrete, allowing the moisture in the atmosphere to attack the steel;
- **Carbonation of concrete**
  Carbon dioxide in the atmosphere diffuses into the concrete and chemically reacts with the products of cement hydration, increasing the porosity of the structure, opening the way to steel corrosion;
- **Chloride ingress**
  Chloride is the most common material which can destroy the protective passivation of he steel in concrete and is generally provided from the atmosphere; this is especially relevant in Sellafield's case due to its coastal location. The loss of passivation again open the way to steel corrosion.

To establish the current status of oxide pool facilities at Sellafield structural surveys have been undertaken specifically looking for any signs of either physical or chemical /environment effects which would impact on their long term integrity. All pools have been found to be in good condition.

The next stage in the investigative process is to extract shallow small diameter test cores from candidate facilities to determine:- characteristic cube strength, depth of carbonation, chloride ingress/penetration, and chemical/petrological examinations to establish cement contents/ water:cement ratios. The use of Fick's Law of Diffusion will then be used to give an indication of the time it will take for the carbonation front to arrive at the reinforcement.

The right choice of facility, however, is also dependent on not only its general condition, but on the evidence of how it was built. Facilities built post 1980 (Fuel Handling Plant and Receipt & Storage) benefit from improved quality assurance, standard of build e.g. seismic qualification, a change in emphasis to durability and better building techniques. The Receipt & Storage facility represents the peak of these achievements and has collected three awards including a special award from the Institution of Structural Engineers (1990) in recognition of "construction to the highest modern quality and safety standards".

Condition monitoring is part and parcel of both the site and facility operating licences. To ensure the long term integrity of the chosen facility is maintained, improvements to the current routine visual inspections are being investigated. These include non destructive techniques such as dynamic monitoring systems, for example Modal Testing Techniques, to initially establish fingerprints of the pool structures which can then be monitored, ideally remotely on line, to provide an early indication of potential system failure.

### 4.2. Operational

#### *4.2.1. Current process*

Irradiated AGR fuel elements are received at Sellafield into Fuel Handling Plant in 15 compartment (A2) skips which are placed into lidded storage containers. Each skip compartment contains a single fuel element. After an initial period of cooling to reduce decay heat the fuel is dismantled and the resulting fuel pins are packed into stainless steel slotted canisters. Each canister contains the equivalent of three elements (108 rods) and they are placed in 20 compartment skips (A1) which are also stored in locked lidded containers. This achieves an overall four fold increase in storage capacity. The dismantled fuel is then transferred to the AGR Storage Pond until it reaches a minimum of 3 years cooled. When required, the fuel is transferred to Receipt & Storage for reprocessing.

The function of skips being housed in locked lidded containers is to prevent a criticality under the fault scenario of a three high stack being knocked over and the dismantled fuel contents spilling out. This arrangement also facilitates the stacking of containers and has the ability to control container water chemistry. All storage facilities are reliant on up stream deionised water plants as the source of pool purge water and down stream sentencing or conditioning plants for pool purge effluents.

#### *4.2.2. Move to long-term storage*

In comparison with current operations the longer term requirements would be to make the chosen facility independent of others and incur low operating costs. For example the pond purge from Fuel Handling Plant is discharged to sea via the Site Ion Exchange Plant (current availability to around 2019), and to reduce resource levels to a minimum by the incorporation of remote techniques to monitor plant status. In terms of storage layout there are significant areas for improvement as the need to be able to immediately access any particular container of fuel is no longer required. Additionally there is no longer a requirement to dismantle the AGR fuel, i.e. removal of the graphite sleeve, unless there is a net benefit through fuel consolidation activities.

Preliminary engineering studies have looked at current storage regimes and the potential for optimisation to deal with the projected business needs. For example if the fuel is stored undismantled in (A2) skip/container combinations there would be a three fold reduction in potential storage capacity. A review of criticality for undismantled fuel allows the requirement for containerisation to be relaxed to just a fixed lid on a skip. Under these conditions the skips can touch both horizontally and vertically without compromising the criticality safety case.

Given the above conditions, i.e. removal of containers and close packing of skips, the standard grid positions in one bay of Fuel handling Plant can be almost doubled. In addition stack height can be increased from three high for containerised storage to a potential four high for skip only storage; with no change to the shielding requirements. The net impact of these changes would mean that the fuel could be stored undismantled without any loss in overall capacity of the facility (in weight of uranium terms).

The question now remains whether storage capacity of the facility could be further increased if Burnup Credit was approved by applying the same arrangements to dismantled fuel.

## 5. SUMMARY

The initial approach and fall back positions being taken by BNFL in managing the AGR lifetime spent fuel arisings from British Energy reactors has been out-lined. Interim storage for up to 80 years is envisaged for fuel delivered beyond the life of the Thorp reprocessing plant. Adopting a policy of using existing facilities has defined the development requirements to demonstrate that this approach can be undertaken safely and business issues can be addressed. The major safety issues are the long term integrity of both the fuel being stored and structure in which it is being stored. Business related issues are reflected in the need to address long term interactions with the rest of site and storage optimisation.

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## APPENDIX 1

### The principles of Field Signature Method (CorrOcean$^{TM}$) experimental technique

The Field Signature Method (FSM) is based upon feeding a direct current through the object to be monitored and measuring the resultant electric field via an array of electrical contacts on the external surface of the object. Changes to the magnitude of these electric fields result from any metal loss due to corrosion. The voltage $A_s$ (pin pair on structure being monitored) and $B_s$ (reference pair) are measured with the structure in its initial condition. This is the "signature" reading. Subsequently voltages $A_i$ and $B_i$ are measured for the same pairs of electrodes and a Fingerprint coefficient is calculated using the equation:-

$$Fc_{Ai} = \left\{\frac{B_s}{A_s} \times \frac{A_i}{B_i} - 1\right\} \times 1000 \qquad \text{(parts per thousand)}$$

where

$Fc_{Ai}$ = Fingerprint coefficient for pair A at time ,
$A_s$ = Voltage across pair A at start,
$B_s$ = Voltage across reference pair B at start,
$A_i$ = Voltage across p[air A at time i,
$B_i$ = Voltage across reference pair B at time i.

By using voltage ratios any variation in the excitation current is automatically compensated. The potential drop between electrode pairs is typically 100-200 μV with a minimum of 50 μV being recommended by CorrOcean. The equipment has a measuring resolution of 4.77 nV. A resolution of 0.05% of the wall thickness is claimed for general corrosion. With filtering to remove noise a resolution of 0.01% is expected to be achievable.

### National Nuclear Corporation (NNC) Electrochemical Noise Probe

NNC Electrochemical Noise Probes were used as part of fuel integrity development work to support the SNL dry store project. The technique basically measures fluctuations in the corrosion current and potential (Electrochemical Noise) on a suitable corrosion probe as a result of changes in environmental conditions. The technique is reliant on producing probes that are representative of the material to be long term stored.

The potential application of this technique would be in the form of an on-line monitor to provide a quick response to loss of pond water chemistry causing corrosion to initiate. It is not expect that this technique would measure the extent of attack, but would compliment the FSM technique.

## SESSION 3

## EXPERIENCE AND LICENSING

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# LICENSING OF SPENT NUCLEAR FUEL DRY STORAGE IN RUSSIA



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Paper withdrawn

**Abstract**

The Federal nuclear and radiation safety authority of Russia (Gosatomnadzor) being the state regulation body, organizes and carries out the state regulation and supervision for safety at handling, transport and storage of spent nuclear fuel. In Russia, the use of dry storage in casks will be the primary spent nuclear fuel storage option for the next twenty years. The cask for spent nuclear fuel must be applied for licensing by Gosatomnadzor for both storage and transportation. There are a number of regulations for transportation and storage of spent nuclear fuel in Russia. Up to now, there are no special regulations for dry storage of spent nuclear fuel. Such regulations will be prepared up to the end of 1998. Principally, it will be required that only type B(U)F, packages can be used for interim storage of spent nuclear fuel. Recently, there are two dual-purpose cask designs under consideration in Russia. One of them is the CONSTOR steel concrete cask, developed in Russia (NPO CKTI) under the leadership of GNB, Germany. The other cask design is the TUK-104 cask of KBSM, Russia. Both cask types were designed for spent nuclear RBMK fuel. The CONSTOR steel concrete cask was designed to be in full compliance with both Russian and IAEA regulations for transport of packages for radioactive material. The evaluation of the design criteria by Russian experts for the CONSTOR steel concrete cask project was performed at a first stage of licensing (1995 – 1997). The CONSTOR cask design has been assessed (strength analysis, thermal physics, nuclear physics and others) by different Russian experts. To show finally the compliance of the CONSTOR steel concrete cask with Russian and IAEA regulations, six drop tests have been performed with a 1:2 scale model manufactured in Russia. A test report was prepared. The test results have shown that the CONSTOR cask integrity is guaranteed under both transport and storage accident conditions. The final stage of the certification procedure for the CONSTOR steel concrete cask in Russia has been started in 1997. On behalf of GOSATOMNADZOR, the Russian assessment institutions VNIPIET and FEI have prepared the safety evaluation report. In 1998, Gosatomnadzor issued the certificate for type B(U)F approval of the CONSTOR cask design. Thus, GOSATOMNADZOR approved the compliance of CONSTOR steel concrete cask design with Russian regulations. The evaluation of the design criteria by Russian experts for the TUK-104 steel concrete cask project will start in June of 1998.

# EXPERIENCE WITH THE LICENSING OF THE INTERIM SPENT FUEL STORAGE FACILITY MODIFICATION

XA9951798

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**Abstract**

After political and economical changes in the end of eighties, the utility operating the nuclear power plants in the Slovak Republic (SE, a.s.) decided to change the original scheme of the back-end of the nuclear fuel cycle; instead of reprocessing in the USSR/Russian Federation spent fuel will be stored in an interim spent fuel storage facility until the time of the final decision. As the best solution, a modification of the existing interim spent fuel storage facility has been proposed. Due to lack of legal documents for this area, the Regulatory Authority of the Slovak Republic (ÚJD SR) performed licensing procedures of the modification on the basis of recommendations by the IAEA, the US NRC and the relevant parts of the US CFR Title 10.

## 1. INTRODUCTION

Spent nuclear fuel is produced at nuclear power plants (NPPs) concentrated at two localities in the Slovak Republic – Jaslovske Bohunice and Mochovce. The two nuclear power plants are equipped with the Russian VVER-440 reactors. The plant at the Bohunice site consists of two V-230 reactors and two V-213 reactors which were put into operation between 1978 and 1985. The Mochovce NPP is under construction and consists of V-213 type reactors. The first of four units was put into operation in August 1998. It is assumed, that operation of the second unit will start in 1999. The destiny of the other two units is not clear.

All VVER 440 units use identical fuel assemblies (FAs) with 126 fuel rods of uranium dioxide pellets in zirconium-niobium cladding. The FA has a hexagonal shape containing 120 kg of uranium with an enrichment of 3.6 or 2.4 weight % U-235. Each reactor contains 349 FAs (313 for the V-230 type) and an average burnup of about 32 GW·d/tU is being achieved. The annual production of spent fuel in Slovakia is approximately 45 tHM. An amount of 500 tHM has already been produced and this figure will increase to about 750 tHM by the year 2000.

The original Slovak spent fuel management was based on the assumption that all spent fuel from VVER reactors would be transported to the former USSR, but only about 700 FAs have been shipped. After the radical political changes in the USSR, no transport of spent fuel from Slovak reactors has taken place. Currently, the spent fuel is stored in at-reactor pools and in the interim spent fuel storage facility (ISFSF) located at the Bohunice NPPs site.

The initial idea to build an ISFSF for the Bohunice plants originated at the beginning of eighties. Shortly after the decision had been made, preparation and implementation of the construction started (the design was supplied from the former USSR) and the ISFSF was commissioned in 1986. The original intention was to store spent fuel in the ISFSF for a period of 10 years, after 3 years cooling in the at–reactor pool. Following this period, the fuel should have been transported for reprocessing or permanent disposal to the USSR. From this assumption resulted the required maximum storage capacity of the ISFSF. The facility is a wet type facility using water as cooling and shielding medium with a total capacity of 5,040 spent fuel assemblies.

After the decision to store spent fuel on the territory of the Slovak Republic until its final disposal, the utility realized that the ISFSF capacity will be exhausted very soon and a new storage should be ready to receive spent fuel already in 1999. Analyzing the situation, the utility decided to extend the capacity of the existing ISFSF at Bohunice, as the most effective solution.

2. MODIFICATION OF THE INTERIM SPENT FUEL STORAGE FACILITY

## 2.1. Current status

The facility consists of four storage pools interconnected by one transport corridor (see Fig. 1). One storage pool is kept empty for emergency cases. Spent fuel is stored under water in open cylindrical baskets placed in steel lined concrete pools. The storage pools are double lined - upper liner and all piping and auxiliary systems are made of stainless steel. The pool concrete pit liner is made of carbon steel. The pitch of spent fuel in the basket is 225 mm. The capacity of one basket is 30 FAs. The water layer above the spent fuel is 3 m.

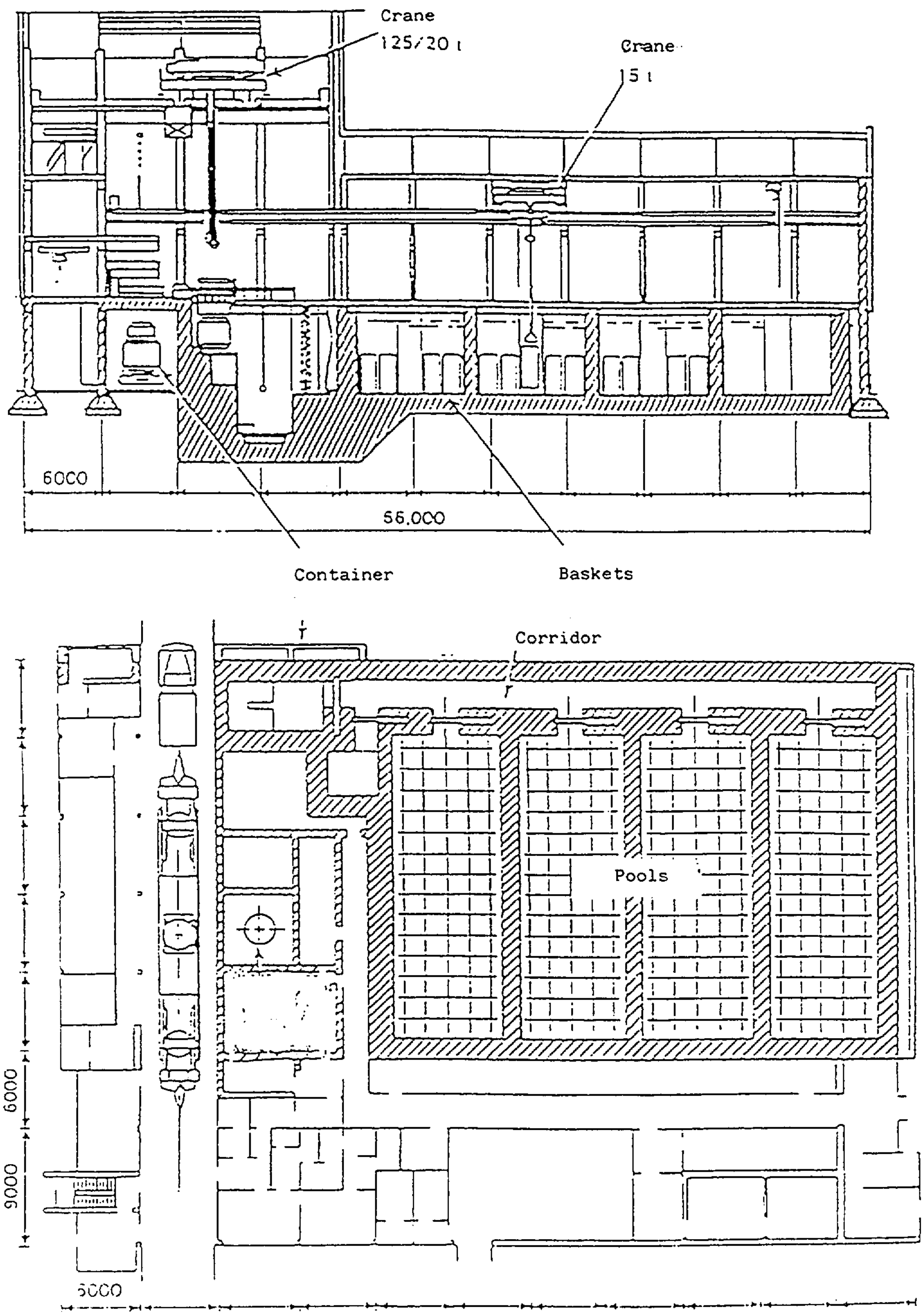


*FIG. 1. The ISFSF arrangement*

Up to now no significant problems have appeared during operation of the facility. Neither there are any symptoms of fuel cladding deterioration or storage component corrosion. Visual observation of the outer surface of selected fuel assemblies using an underwater camera has been carried out. This inspection did not reveal any significant difference between the newest and oldest stored FAs and there was no corrosion identified. Up to now, 4,752 FAs are stored in the Bohunice ISFSF which represents more than 94 % of its storage capacity.

## 2.2. Main goals of the modification

The operator proposed to modify the ISFSF through the following steps:

- compact arrangement of spent fuel up to the level of creating sufficient capacity for the operation of all four units of the Bohunice NPP during the whole period of their lifetime;
- seismic upgrading of the ISFSF building and its technology to 8 degrees of MSK scale;
- extension of the lifetime of the ISFSF for the period of about 50 years.

To extend the capacity the operator designed a new basket type for spent fuel (see Fig. 2) with capacity of 48 FAs and proposed also a new arrangements of the baskets in existing pools (see Fig. 3) so that the total capacity of the facility would increase up to 14,112 FAs.

Analyses of the requirements for the modification showed that:

- the increased number of FAs will require structural modification of the cooling system for residual heat removal from spent fuel during the whole ISFSF operating lifetime;
- the seismic upgrading could be achieved by strengthening of the building supporting structure and by exchange of its wall-panels. The water cooling and make-up system as well as selected power supply and I&C systems should be also upgraded. The storage pools were adequately seismic resistant;
- elimination of the non-uniform subsidence of the ISFSF building during the whole period of its operation of about 50 years should be assured by appropriate arrangement of baskets with spent fuel.

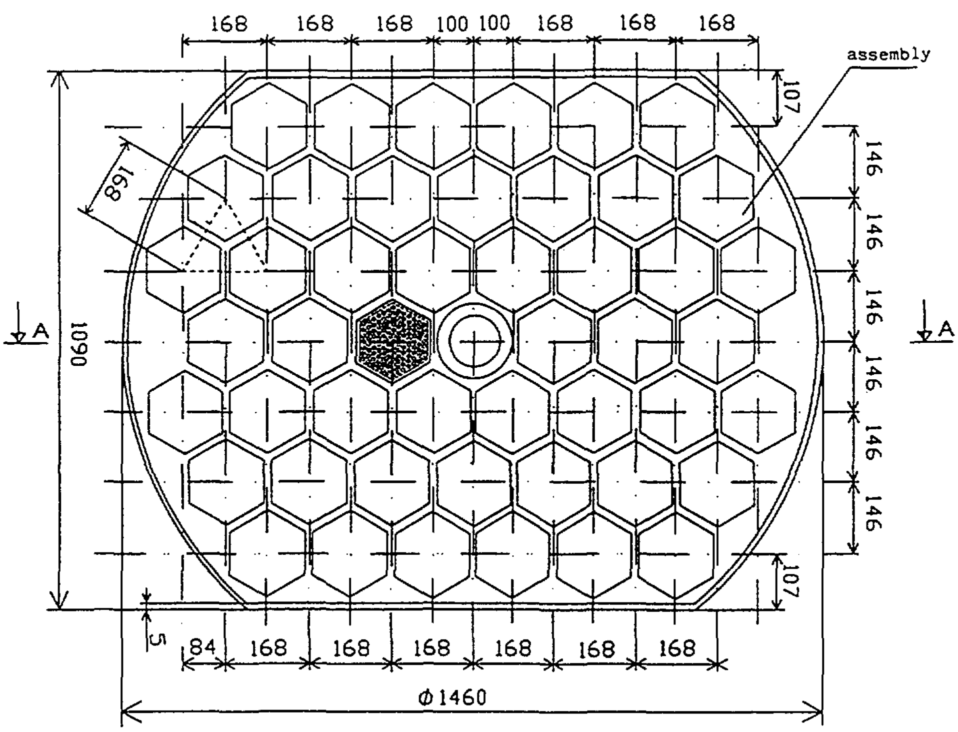


*FIG. 2. Compact basket*

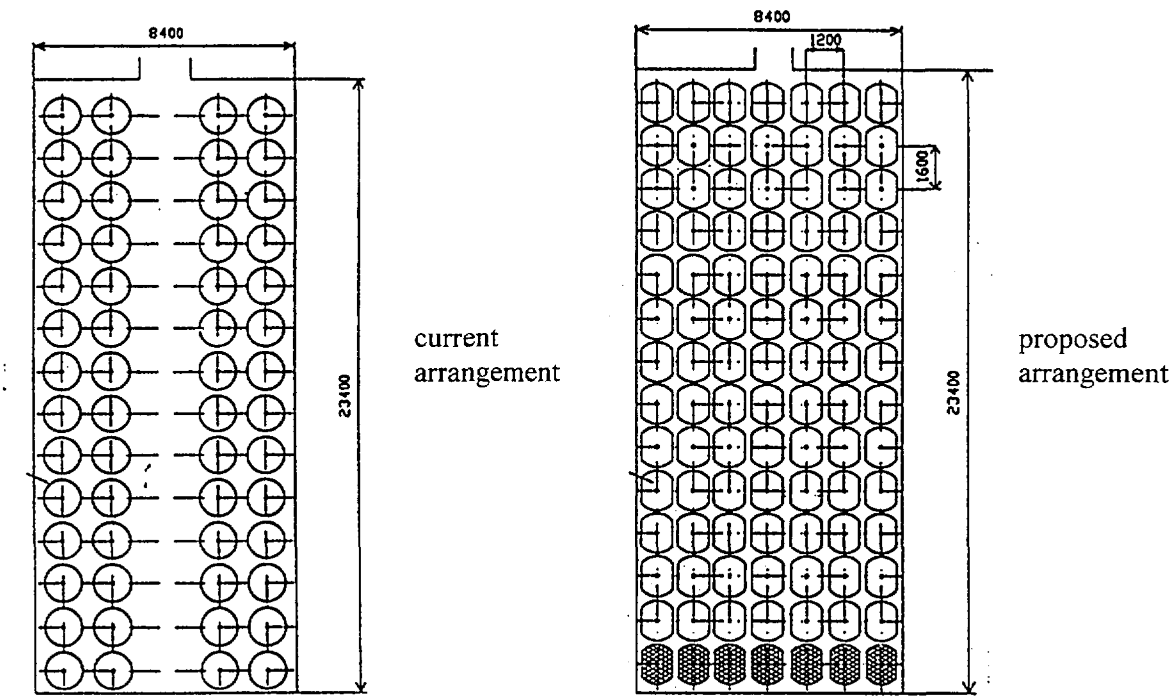


*FIG. 3. Arrangement of baskets*

### 2.3. Licensing process

The modification of any nuclear facility in the Slovak Republic may be licensed in two ways:

- the licence is issued by the Construction Office in the case of substantial changes of the facility. In this case the ÚJD SR issues a statement which is obligatory if it is negative;
- in other cases the licence is issued by the ÚJD SR.

After a long and complicated discussion with the Construction Office and the ÚJD SR, the operator of the ISFSF admitted that its proposal for modification would significantly influence the original purpose of the storage and hence, the licence of the Construction Office would be necessary. In this case, the licensing process should have three steps: (1) siting, (2) construction and (3) commissioning. For each step, the operator should apply for a licence at the Construction Office. The ÚJD SR issues its statement – approval if positive - for each step on the basis of assessment of safety documentation submitted by the operator.

#### *2.3.1. Siting*

Early in 1996, the ISFSF operator applied at the ÚJD SR to issue an approval required by the legislation procedure according to the Construction Act No. 50/1976. As part of this application, the Final Safety Analysis Report was submitted to the ÚJD SR. Even the ÚJD SR accepted this report as a document for environmental impact assessment according to Act No. 127/1994. Careful study of the Environmental Impact Assessment report resulted in the regulatory body's position that the submitted report failed to address sufficiently the issue of the building subsidence and the assurance of its seismic resistance up to 8 degrees of MSK scale. The ÚJD SR expressed a request to supplement the report with detailed analysis of the mentioned issues supported by calculations taking into account the whole expected period of its operation. The Construction Office issued its siting licence with a condition that requirements of the ÚJD SR will be fulfilled until the construction licence will be issued. The form of this document followed a pre-operational safety report for the existing ISFSF and was modified according to requirements of the Act No. 50/1976.

*2.3.2. Construction*

Based on the results of the discussions with the ÚJD SR, the operator started preparation of a preliminary safety analysis report according to the requirements shown in the appropriate documents of the IAEA, US NRC and § 72 of the US CFR Title 10. The format of the report was based on the US NRC Guide No. 3.44 Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (water-basin type) except of chapter No. 9 Conduct of Operations and No. 10 Operating Controls and Limits, which would be included into the pre-operational safety analysis report. This report was submitted to the ÚJD SR for review in the middle of 1997 together with an application for ÚJD's approval required by the Construction Office.

The ÚJD SR reviewed the submitted report from the point of view of compliance with the provisions of the contents according to which the report was prepared, and from the point of view of conformance with the basic safety functions of the ISFSF, namely:

- assurance of subcriticality of the spent fuel stored;
- residual heat removal;
- prevention of releases of radioactive material into the environment.

The major portion of the safety report, predominantly that dealing with the parts not affected by the modification, was taken over or was supplemented by new evaluations based on the safety analysis report developed for the existing ISFSF. This part consists basically of geographical, demographical, meteorological, hydrological evaluations of the site and of the possibility to use the ISFSF auxiliary systems interconnected with auxiliary systems of the NPP in operation.

With regard to the compliance with the safety functions for the modified ISFSF, the following items had to be demonstrated:

- assurance of subcriticality of the spent fuel stored in compacted form in baskets and of baskets in pools under all nominal conditions and operational events of the ISFSF;
- reliable residual heat removal from the spent fuel during the whole period of the ISFSF operation;
- reliable confinement to avoid release of fission products into environment.

The results confirmed the following safety functions under normal conditions:

**(1) Subcriticality**

By means of the SCALE 4.3 code, it was confirmed that with spent fuel located in basket tubes made of boron steel with a boron content in the range of 1-1.1 percent , basket wall width in the range of 3-3.3 mm, inside tube diameter of 151-153 mm, mesh of absorber tubs of 167-169 mm, stored fuel with an enrichment up to 4.4 weight % of U-235, storing water temperature of 20°C and infinite assembly grid, accounting for their most adverse value, the value of $k_{eff} + 2\ \sigma$ does not exceed 0.946 which is less than 0.95 as required.

In an analogous way for a real basket structure of a storage facility, the value of $k_{eff} + 2\ \sigma =$ 0.875. In the case of average values of the above mentioned basket parameters, the estimate value of $k_{eff} + 2\ \sigma = 0.91$ and for an infinite basket grid (with the step of their arrangement of 1,600 times 1,200 mm) $k_{eff} + 2\ \sigma = 0.849$.

**(2) Residual heat removal**

For demonstrating sufficient heat removal, the RELAP/MOD3 code was used. It was calculated that for natural water circulation through the spent fuel assemblies stored in a pool, the coolant heat-up will not exceed 7°C. For basket inlet water temperature less than 30°C the pool water temperature does not exceed the required 40°C. Under nominal operating conditions, a pool water cooling system is automatically activated when the pool water temperature reaches about 40°C. Under normal operating conditions, the fuel cladding temperature in the assembly with peak residual heat does not exceed the value of 46°C.

Calculations also demonstrated the maximum value of the integral residual heat of the completely filled storage facility which should reach about 2,000 kW after approximately 25 years since the storage would be completely loaded by spent fuel. To remove such residual heat, the pool water cooling system should be modified. It was proposed to replace the existing heat exchanger by a new one of the plate type with a cooling output of 2,000 kW.

**(3) Confinement**

It was demonstrated that the existing ISFSF filtration system has sufficient capacity for capture and disposal of fission products that were released into the pool from small cladding defects of spent fuel. It was also demonstrated that the filtration capacity of the existing purification system is sufficient for the capture of the whole inventory of fission products of all fuel elements in one basket, if released through a possible total cladding falaire. The purification capacity for corrosion products is sufficient for their absorption as well. It is thus not necessary to increase the current purification capacity of the facility.

The other systems such as for dosimetry control, electric power supply, fire protection, monitoring of fuel cladding and structural parts degradation are sufficient for reliable operation of the reconstructed ISFSF and their concepts and capacity need not to be modified. Certain modifications of certain elements may occur due to their physical or moral wear out. However, these would not affect the basic idea or the way in which the modification will be implemented.

In the submitted preliminary safety analysis report, the main safety functions were analyzed under incident and accident conditions as well. With regard to abnormal operation, the following items were analyzed:

- failure of transport means during transportation of a full basket;
- interruption of water supply to the heat exchanger secondary side for a period of 8 hours;
- interruption of pool water cooling due to pump or heat exchanger failure for a period of 8 hours;
- failures in pool water make-up for a period of 8 hours;
- failures in ventilation systems for venting the space above the pool water level for a period of 8 hours;
- water leakage from the pools due to shut – off gate leaks;
- drop of a full basket;
- drop of a fuel assembly during handling;
- failures in pool water purification system;
- loss of power supply.

and with regard to accident conditions:

- changes in pitch of baskets in the pool;
- changes in pitch of assemblies in the basket;
- insertion of fuel with higher enrichment than expected;
- total black-out of electric power supply for two days;
- accident related to spent fuel transport.

The analysis of the above mentioned cases demonstrated a sufficient time margin for their elimination (accidents related to residual heat removal), or a very low probability of occurrence of such events (e.g. for a failure to maintain subcriticality), so that from this point of view nuclear and radiation safety of the modified ISFSF is ensured sufficiently. However, the ÚJD SR requested to amend the submitted safety report by:

- determination of the ISFSF operating life time and justification whether changes of characteristics of structural and technological parts during the ISFSF planned lifetime were accounted for in the design;
- specification of the way in which the ISFSF will be operated after shut down of the NPP V-1, as the current ISFSF uses some auxiliary systems of this plant;
- preliminary schedule for decommissioning of the ISFSF following the termination of its operation;
- the procedure for assurance of nuclear and radiation safety of the ISFSF operation during the modification work.

As all above mentioned issues had been answered sufficiently, the ÚJD SR has issued a positive statement as a recommendation to the Construction Office which has subsequently issued a construction licence.

The current statues of the work on the ISFSF modification gives a real expectation that the modified ISFSF will be brought into normal operation at the beginning of 1999.

*2.3.3. Operation*

Together with the application for the operational licence the operator must submit a safety document including mainly a pre-operational safety analysis report. The report must contain all changes in the project of modification important from the point of view of nuclear safety, chapters dealing with conduction of operation and operating controls and limits and results of tests of modified systems and components. According to the Act No.130/1998 on Peaceful Use of Nuclear Energy, which came into force on 1 July 1998, the safety document should also contain:

- plan of physical protection;
- conceptual plan for decommissioning;
- selected operating procedures;
- evidence of special qualification of employees;
- evidence of readiness of nuclear installation for start-up;
- evidence of insurance or other financial cover.

The ÚJD SR is going to issue a permission for operation only for limited period (approximately 5 to 10 years). The most important condition for prolonging the operational licence will be the positive results of monitoring structural material and fuel cladding degradation.

**2.4. Regulatory activity during the modification**

The ISFSF will be modified during its normal operation. Therefore it is necessary to minimize the influence of the modification on the nuclear safety. The ÚJD SR made an agreement with the operator that for each activity influencing nuclear safety, the operator would prepare a detailed programme which would be submitted to the ÚJD SR for approval. So far, several dozen of such programmes have been evaluated. The most important were programmes for replacement of cooling system (pumps and heat exchangers), seismic upgrading of water make-up system, power supply switchboards, replacement of the storage building outer wall and roof panels etc. The ÚJD SR conducts inspection to check compliance of real procedures with the approved programmes.

Due to time consuming procedures connected with the cooling system modification, the operator applied for modification of the operating limits mainly in the part dealing with the inoperability of the system. Based on analysis submitted with the application, the ÚJD SR approved the inoperability of the cooling system for 4.5 days under the condition that the current spent fuel arrangement (input data for the analysis) would not be changed.

## 3. CONCLUSION

The modification of the ISFSF at Bohunice is quite a new experience for the ÚJD SR. Due to lack of legislation for a regulatory activity in this area, the ÚJD SR utilized recommendations of the group of international experts concentrated in the IAEA Safety Series No. 116, 117 and 118 as well as regulatory guides and legal documents used in the USA. Based on these documents, the content of safety analysis reports has been defined. The elaborated preliminary safety analysis report has been evaluated with assistance of experts from universities and the Academy of Science. The pre-operational safety analysis report is under preparation. The most effective way how to assure nuclear safety during the modification are operational programmes for each activity with potential influence on nuclear safety combined with ÚJD's inspections.

The ÚJD SR is going to use the gained experience in the preparation of decrees and regulations implementing the requirements of the Act No. 130/1998 in the area of safety documentation for nuclear facilities and spent fuel management as well as in the preparation of the ÚJD's guide for interim spent fuel storage facilities.

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# QUALITY ASSURANCE AND DESIGN CONTROL PROBLEMS ASSOCIATED WITH THE FABRICATION AND USE OF SPENT FUEL DRY STORAGE COMPONENTS

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Presented by M.W. HODGES, U.S. NRC

**Abstract**

This paper presents the concerns of the staff of the U.S. Nuclear Regulatory Commission (NRC) regarding vendor and utility quality assurance (QA) oversight during the design and fabrication of spent fuel dry storage cask (DSC) systems. Deficient QA and design control programmes have resulted in significant enforcement actions against both vendors and utilities. In addition, the utilities, vendors, and NRC, have expended a considerable amount of resources on resolving these problems. As a result, some utilities have been forced to explore other options for long-term storage of spent fuel, including reracking the spent fuel pool and switching DSC vendors. Some vendors stopped fabricating DSCs until appropriate corrective actions were implemented. This resulted in significant financial and operational burdens on both utilities and vendors. In fiscal years 1996 and 1997, NRC reallocated resources from licensing activities to increased inspection and enforcement activities, thus causing delays in the licensing of new DSC designs. It is imperative that vendors and utilities learn from these mistakes and implement effective QA and DC programmes.

## 1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) began to identify declining trends in quality assurance (QA) and design control (DC) at several dry storage cask (DSC) vendors in the early 1990s. Although, NRC inspections found that in most cases the vendors' QA programmes met NRC requirements, inspectors noted that the vendors had not adequately implemented their QA programmes. At that time, NRC issued generic information notices, specific letters to vendors, and conducted meetings with the vendors and utilities in an effort to prevent further declines in performance. However, problems continued.

Problems became so widespread that, in 1997, only two utilities could load spent fuel into dry storage systems. Numerous QA and DC deficiencies that NRC identified during routine inspections of DSC vendors and fabricators resulted in significant NRC enforcement actions which prohibited some vendors from fabricating DSCs and some utilities from loading DSCs. In other instances, utilities voluntarily stopped loading DSCs until the vendors and fabricators corrected the problems. It was not until late Summer 1998 that NRC allowed loading of DSCs to resume.

A number of problems contributed to vendors' unacceptable QA and DC performance. The problems can be grouped into four areas: 1) vendor failure to provide adequate QA during the design process; 2) inadequate vendor oversight of fabrication contractors; 3) vendor failure to follow regulatory processes to resolve both non-conformance and control design changes during fabrication; and 4) failure of vendors to review non-conformance for generic implications and to promptly notify system users of their findings. This paper presents NRC staff views on the root causes that led to the decline in vendor QA and DC performance and the correcting actions taken by the vendors.

## 2. VENDOR FAILURE TO PROVIDE ADEQUATE QA DURING THE DESIGN PROCESS

The most safety-significant issues arose because vendors failed to implement effective QA during the design process. In May 1996, an ignition of hydrogen gas occurred while a worker was welding a closure lid on a DSC. The hydrogen was generated when the coating, used to protect the carbon steel cask from the borated water in the spent fuel pool (SFP), reacted with the boric acid in the pool. NRC later discovered that during the design process the vendor had not adequately assessed the coating for this application. In another instance, as the result of NRC enforcement actions, a vendor performed a design review and identified that the design calculations performed to support the safety analysis report (SAR) design basis, did not support the conclusions in the SAR. No DSCs of this design were loaded until the vendorreanalyzed the design and made necessary modifications.

In December 1996, shortly after the welds were completed on a DSC, cracking was identified in the cask closure welds. Further investigation revealed that over 20 percent of this type of DSC had experienced some type of cracking during the welding of closure lids. The vendor found that the SAR lacked sufficient detail on weld parameters to ensure the completed welds met the design basis. This issue took over 18 months to resolve during which time no utilities were allowed to load DSCs of this design.

## 3. INADEQUATE VENDOR OVERSIGHT OF FABRICATION CONTRACTORS

NRC identified a significant number of instances in which vendors did not perform appropriate QA reviews of contractors during the fabrication of dry storage systems. As a result, it was later discovered that some fabrication contractors had not fully complied with all design documents and applicable industry codes during the fabrication process. For example:

- In 1994, a utility identified defects in the seam weld of a DSC containment boundary. The utility discovered the defects while reviewing contractor radiographs, after the DSC was loaded with spent fuel and placed in service. Had the vendor properly implemented its QA programme, the defects would have been discovered and corrected before the DSC was placed in service.
- In 1995, a DSC containing a seam weld that did not meet the minimum wall thickness was placed in service. Had the vendor provided adequate oversight of the contractor's fabrication process, this defect would have been discovered and corrected before the DSC was placed in service.
- In 1996, NRC identified that concrete aggregate and paint used to fabricate DSCs did not meet the design basis approved by NRC. In this case, design requirements set forth in the licensing basis documents were not accurately transferred to fabrication documents by fabrication contractors, and the vendor did not review the fabrication documents. NRC discovered this problem before theDSC was placed in service.

Although none of these problems involved significant public health and safety concerns, these and numerous other problems indicated an overall weak effort by the DSC industry to provide adequate QA during the fabrication of DSCs. NRC determined it was prudent to take enforcement action to preclude the recurrence of these problems. Subsequently, NRC issued two of the vendors Demands for Information (Demand). A Demand is an enforcement tool NRC uses to request a vendor to show cause as to why, in light of the facts set forth in the Demand, NRC should not Order the vendor to suspend fabrication and/or design activities untilNRC's concerns are resolved.

## 4. INADEQUATE RESOLUTION OF NON CONFORMANCE AND DESIGN CHANGES DURING FABRICATION

Another problem that NRC identified, involved inadequate vendor and utility evaluations of non conformance identified during the review of design changes. All utilities have the authority to make changes to the SAR, provided the changes do not affect license conditions or constitute safety questions that NRC has not reviewed. Utilities and vendors routinely make changes to the DSC designs for various reasons, including utility-specific needs and fabrication considerations. However, NRC has identified numerous instances in which the proper regulatory processes for making changes were not followed.

NRC regulation 10 CFR 72.48 provides a requirement and an evaluation process for utilities to determine if design changes require NRC approval before implementation. Notwithstanding the regulatory requirement, in many instances vendors and utilities would dispose of minor, one-time, changes to drawings referenced in the SAR, without adhering to 10 CFR 72.48 requirements. During review and approval of a DSC design, NRC relies not only on the text of an SAR, but also on the drawings referenced in the SAR. Material or dimensional changes to drawings that do not involve changes to the text of the SAR may seem minor, but may have significant impacts on the design basis and behaviour of the DSCs under both normal and accident conditions.

This problem became so widespread that NRC issued Information Notice 97-39, "Inadequate 10 CFR 72.48 Safety Evaluations of Independent Spent Fuel Storage Installations", dated 26 June 1997, to alert the industry to the breadth of this concern. The Information Notice stated that a DSC:

> *..."is a passive system in which its structural, criticality-control, thermal, and shielding performances depend on the detailed drawings and descriptions provided as the design bases in the SAR. It is important that the licensee perform an adequate 10 CFR 72.48 safety evaluation of any modification in the independent spent fuel storage installation SAR or DSC SAR, including any changes in dimensions, materials, and procedures. In addition, one-time changes to the independent spent fuel storage installation (ISFSI) SAR or the DSC SAR constitute a modification that also requires an adequate 10CFR 72.48 safety evaluation".*

In 1998, NRC identified two instances where, because of inadequate 10 CFR 72.48 safety evaluations, utilities did not recognize they were required to seek license amendments from NRC. The first instance involved a design deficiency that represented a malfunction of equipment important to safety that had not previously been evaluated in the SAR. The second instance involved the loading of core components, other than spent fuel, into a DSC. The loading of core components was not permitted under the utility's license to operate its ISFSI. These issues concern NRC, because they represent licensee failure to properly implement the requirements of 10 CFR 72.48 and therefore, to identify safety-significant deviations from their licensing bases.

## 5. FAILURE TO REVIEW NON-CONFORMANCE FOR GENERIC IMPLICATIONS

NRC is concerned that vendors have not adequately reviewed non-conformance for generic implications and notified all users of deficiencies in a timely manner. NRC has identified cases in which non-conformance appear to be addressed on a case-by-case basis. Some vendors have not notified NRC, nor utilities using the systems, of generic design and operational concerns. Examples of generic issues that vendors did not properly evaluate nor communicate to the utilities are:

- A vendor did not communicate to user utilities discrepancies in the SAR associated with the selection of concrete aggregate used to fabricate DSC concrete storage components;
- A vendor did not communicate that certain conditions contained in a DSC Certificate of Compliance (COC) did not adequately bound SFP operating conditions at utilities using the DSC. Further, the vendor failed to request NRC for an amendment to the COC;
- A vendor did not identify the generic implications of welding problems associated with DSC closure welds;
- A vendor did not address the generic implications of fabrication errors that resulted in a utility using a DSC that did not meet the minimum wall-thickness requirements.

NRC is very concerned that vendors have failed to identify generic issues and communicate them to DSC users. NRC expects vendors and utilities to identify problems and requires them to implement appropriate corrective actions to resolve problems. If vendors are finding problems but not notifying utilities of those problems, a utility cannot implement corrective actions. The result is that DSCs that do not meet their design bases may inadvertently be placed in service.

## 6. ROOT CAUSE AND CORRECTIVE ACTIONS

Both vendors that received Demands from NRC contracted independent entities to perform assessment of the effectiveness of their QA and corrective action (CA) programmes. The assessments identified numerous problems throughout the vendors' QA and CA programmes. Although some of the problems were specific to just one vendor, several problems were common to both. The following is a sample of the most significant findings common to both vendors:

- Inexperience in implementing 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level RadioactiveWaste";
- Inadequate management attention to emerging issues and problems;
- Inadequate management attention to theQA and CA programmes;
- Inadequate staffing and training;
- Inadequate root cause analysis programme, including analysis of problems for generic implications.

In response to these findings, both vendors receiving Demands initiated extensive recovery programmes. The following are examples of corrective actions common to both programmes:

- Reorganizing and also replacing several senior managers;
- Hiring additional staff;
- Augmenting corrective action programmes to clearly define staff roles and responsibilities;
- Establishing mechanisms to allow better vertical flow of information up to senior managers and flow of management expectations to staff;
- Performing detailed design reviews to identify issues that may have been previously missed.

## 7. CONCLUSION

Since 1992, the nuclear DSC industry has made progress in improving QA and CA programmes. Utilities have become more involved in assuring the quality of the DSCs they are purchasing. Most utilities have dedicated teams to oversee QA during DSC design and fabrication. Recently, utilities have instituted the use of the Nuclear Utilities Procurement Issues Committee to perform independent audits of the various DSC designers. Overall, NRC has seen improvements in the design and fabrication ofDSCs.

However, these improvements have not been without a cost. Some vendors have declared bankruptcy and been purchased by competitors. Utilities have had to delay loading of DSCs until design and fabrication deficiencies were corrected. In some cases, as a result of not being able to load DSCs, certain plant maintenance activities had to be delayed because there was insufficient room in the SFP to completely unload fuel from the reactor core. Delays have occurred in licensing new DSC and transportation applications because NRC staff had to temporarily be reallocated to the inspection of DSC design and fabrication and the corresponding enforcement action that ensued.

It now appears that the industry has turned the corner in QA. However, vendors and utilities need to be ever vigilant for problems to avoid further delays in the licensing, fabrication, and use oDSCs.

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# WELDING ISSUES ASSOCIATED WITH DESIGN, FABRICATION AND LOADING OF SPENT FUEL STORAGE CASKS

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**Abstract**

The U.S. Nuclear Regulatory Commission (NRC) has observed a number of welding issues associated with design, fabrication, and loading of spent fuel storage casks. These emerging welding-related issues involving a certain dry cask storage system have challenged the safety basis for which NRC approved the casks for storage of spent nuclear fuel. During closure welding, problems have been encountered with cracking. Although the cracks have been attributed to several causes including material suitability, joint restraint and residual stresses, NRC believes hydrogen-induced cracking is the most likely explanation. In light of these cracking events and the potential for flaws in any welding process, NRC sought verification of the corrective actions and the integrity of the lid closure welds before allowing additional casks to be loaded. As a result, the affected utility companies modified the closure welding procedures and developed an acceptable ultrasonic inspection (UT) method. In addition, the casks already loaded at three power reactor sites will require additional non-destructive examinations (NDE) to determine their suitability for continued use. NRC plans to evaluate the generic implications of this issue for current designs and for those in the licensing process.

## 1. BACKGROUND

Because most United States (U.S.) nuclear power plant spent fuel pools were not originally designed for all the spent fuel generated over the licensed period, many plants have reached or are approaching their storage capacity. Consequently, U.S. nuclear power plants have turned to alternative means and are storing spent reactor fuel in dry casks at 11 sites (see Fig. 1). Currently, 21 new locations are planned or under construction (Fig. 2).

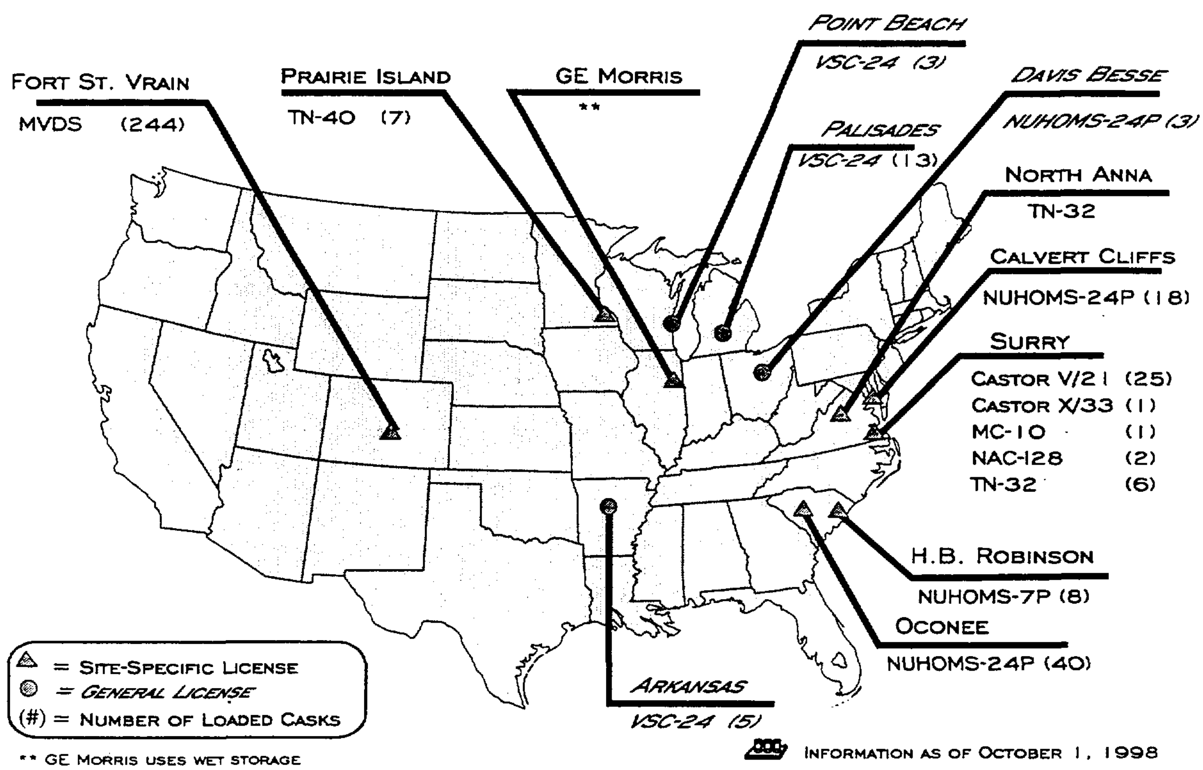


*FIG. 1. Operating spent fuel storage sites*

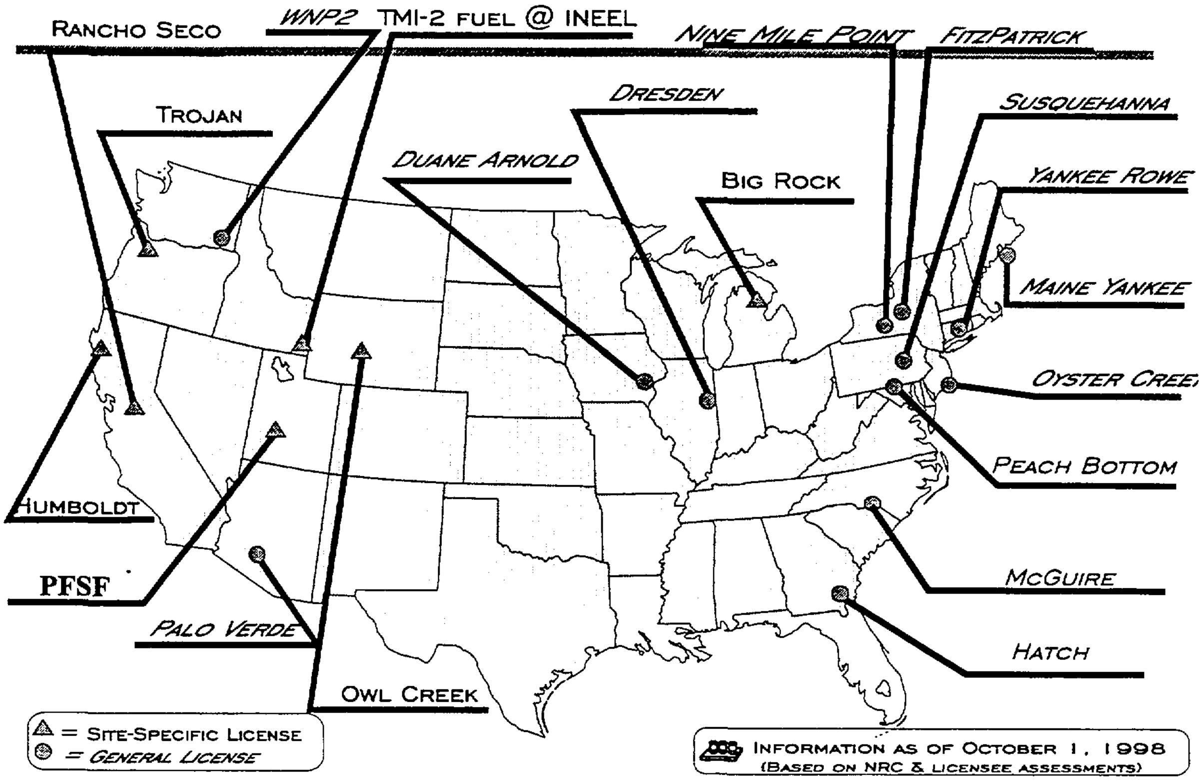


*FIG. 2. Potential near-term spent fuel storage sites*

During their licensed lifetimes, approximately 90 percent of the 105 operating U.S. nuclear plants will need storage for spent fuel beyond the current capacities in their spent fuel pools. The increased demand for dry cask storage has led to new designs, vendors, and fabricators. Three power plants [Palisades, Point Beach, and Arkansas Nuclear One (ANO)] use the ventilated storage cask system (VSC-24) design [1] to store their spent fuel in an independent spent fuel storage installation licensed under 10 CFR Part 72 [2].

The VSC-24 consists of a ventilated concrete cask (VCC) that stores 24 pressurized water reactor (PWR) spent fuel assemblies (SFAs) vertically in a steel multi-assembly sealed basket (MSB) (Fig. 3). An MSB transfer cask (MTC) is used to transfer the filled MSB from the reactor spent fuel pool to the VCC. The VCC is a 3.4 m (11.2 ft) diameter hollow right cylinder with steel lined, 74 cm (29 in) thick reinforced-concrete walls. The VCC varies from 5.0 to 5.4 m (16.4 to 17.7 ft) high and has four air inlet penetrations near the bottom and four air outlets near the top. A 1.9 cm (0.75 in) steel plate is bolted on top. The MSB consists of a 2.5 cm (1 in) thick SA-516, Grade 70 steel 1.6 m (5.3 ft) diameter cylindrical shell with a 24 cm (9.5 in) thick shield plug at the top and steel cover plates welded at each end. The shell length is fuel specific and varies from 4.2 to 4.6 m (13.8 to 15.1 ft). The internal steel basket is a welded steel structure consisting of 24 square storage locations.

Between March 1995 and March 1997, the utilities using the VSC-24 experienced four incidents in which cracks occurred in either the weld between the shield lid and the MSB shell or the weld between the structural lid and the MSB shell (Fig. 4). This cracking was identified by either helium leak test or dye penetrant examination required to be performed during cask loading. Table I summarizes the weld crack events. The MSB shell, shield lid, structural lid, and their closure welds form part of the confinement boundary for the VSC-24 dry spent fuel storage system and are classified as important to safety.

After learning of the cracking events at Palisades (March 1995) and ANO (December 1996) and because these weld joints form part of the confinement boundary, NRC became concerned about the difficulties encountered with welding and performed an inspection at SNC in March 1997. As part of the inspection, NRC evaluated possible root causes but did not positively confirm a root cause. However, NRC concluded that neither SNC nor utilities using the VSC-24 had performed a comprehensive analysis to identify the root cause(s) of the welding problems nor implemented sufficient corrective actions to preclude recurrence of the problems. On March 26, 1997, shortly after the inspection, another crack occurred during loading at ANO.

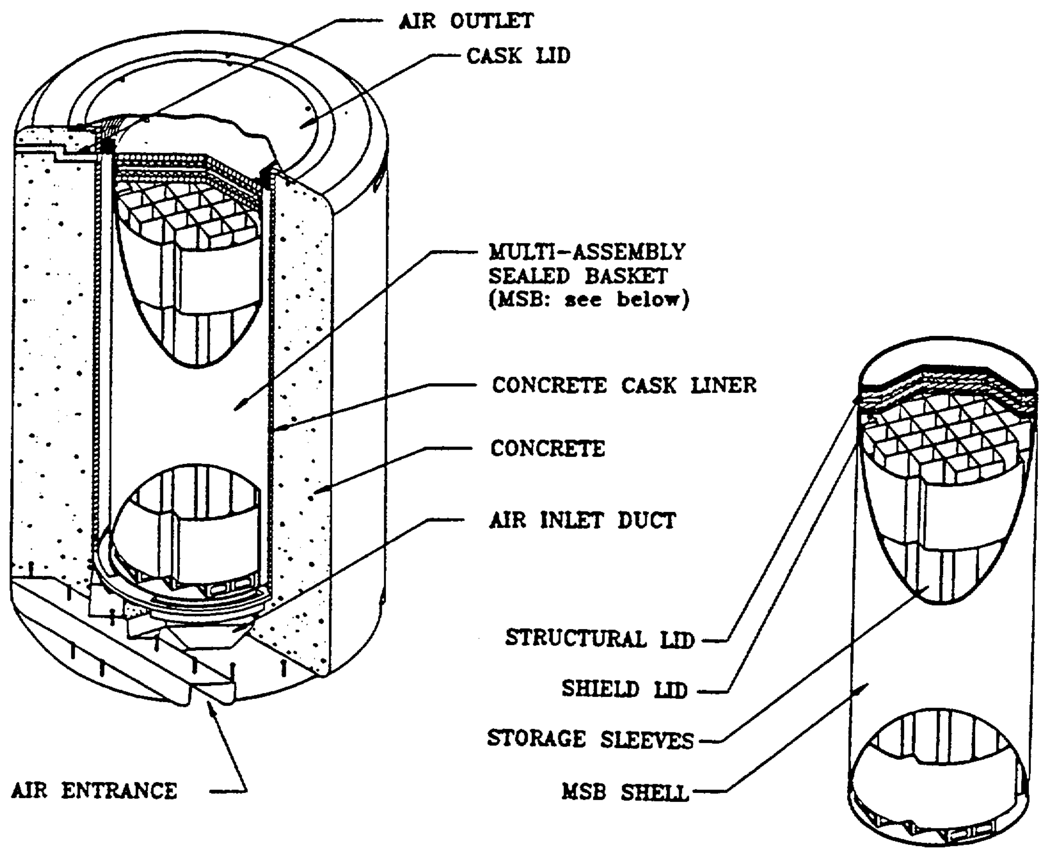


*FIG. 3. Ventilated storage cask (VSC-24) system component*

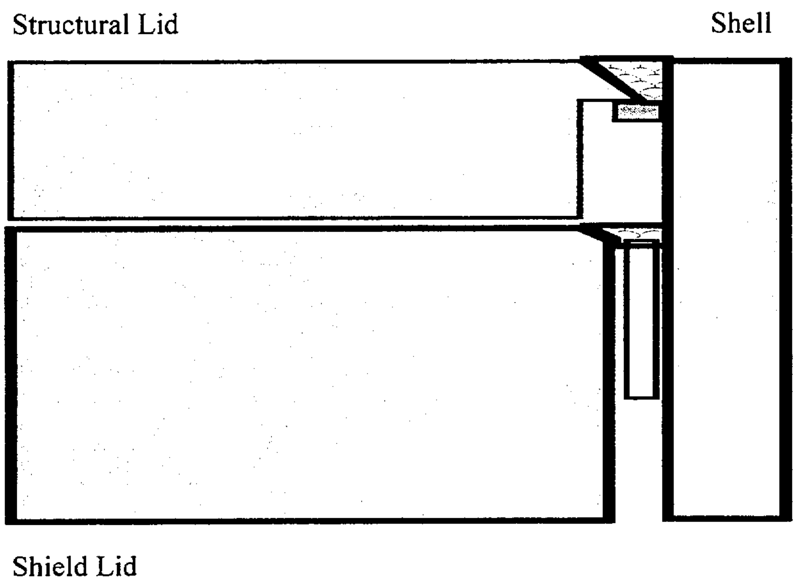


*FIG. 4. MSB structural lid-to-shell weld*

TABLE I. WELD CRACKING EVENTS SUMMARY

| Facility | Date | Detection | Location | Description |
|---|---|---|---|---|
| Palisades | 3/95 | helium leak test | shield lid-to-shell weld | about 15 cm (6 in) long by 3.2 mm (1/8 in). deep that extended from about 3.2 mm (1/8 in) above the shield lid-to-shell weld fusion line into the shell base metal |
| Point Beach | 5/96 | liquid penetrant test (PT) | structural lid-to-shell weld | three cracks, each less than 1 inch long, located along the center of the root pass at locations where the fit-up gap between the lid and the backing ring was widest |
| | | | structural lid-to-shield | in addition, cracking and weld porosity were found in the structural lid-to-shield lid seal weld (fillet weld associated with the vent port covers) |
| ANO | 12/96 | helium leak test | shield lid-to-shell weld | about 10 cm (4 in) long located along the weld fusion line |
| ANO | 3/97 | liquid penetrant test (PT) | shield lid-to-shell weld | about 18 inches long located along the weld fusion line of the root pass |

On 16 May 1997, NRC issued confirmatory action letters (CALs), to SNC and the three utilities [3,4,5,6] using the VSC-24, that documented their commitments to resolve the welding problems. In response to the CALs, the utilities using the VSC-24 and SNC formed the Owners Group to collectively identify the root cause(s) and develop corrective actions.

The Owners Group assembled a team to evaluate the welding problems. The team consisted of industry experts in metallurgy, welding and NDE. This evaluation included an assessment of the information gathered during the initial evaluations performed by the individual licensees and additional testing and evaluation to determine the root causes of the four observed cracking incidents. To ensure that the Owners Group's proposed corrective actions received the appropriate technical and regulatory attention, NRC also assembled a team of staff experts in the areas of metallurgy, welding and NDE.

Subsequently, in June 1997, during its investigation into the welding problems, the Owners Group identified weld repairs that had been made to numerous MSB shells during fabrication that were not documented in accordance with VSC-24 licensing commitments and regulatory requirements. As a result, on 5 September 1997, NRC supplemented the CALs [7,8,9] issued to the utilities. The supplements documented each utility's commitment that, in addition to resolving the welding problems, they would certify that all unloaded casks intended for use were fabricated in accordance with the VSC-24 licensing basis and regulatory requirements.

On July 30, and September 18, 1997, the Owners Group submitted to NRC the results of its review and proposed corrective actions to the welding process to inhibit the cracking [10,11].

## 2. ROOT CAUSES OF THE WELD PROBLEMS

The Owners Group weld review team evaluated each of the four weld cracking events to identify the root cause(s). Their evaluation included a reassessment of the initial evaluations performed by the individual users, additional evaluation by the team and testing. The team identified separate root causes for each of the four cracking events.

The team concluded that the weld crack in the Palisades cask was caused by an existing condition in the rolling plane of the shell material which was opened up by the process of making the shield lid weld. Metallographic analysis revealed a crack that propagated along prior austenitic grain boundaries of a pre-existing weld of unknown origin, possibly an undocumented repair weld.

The cracks on the root pass of the structural lid-to-shell weld, associated with the Point Beach cask, were determined to be caused by wide fit-up gaps that were not properly filled by the welding technique. This resulted in a lack of fusion in the weld metal. The cracking and weld porosity found in the structural lid-to-shield lid seal weld were caused by moisture contamination of the weld. The moisture came from water forced out of the drain line during cask loading. The team concluded that the causes of the weld cracks at Point Beach were associated with the welding technique and were not related to the causes of the cracking observed at Palisades or ANO.

The crack in the shield lid-to-shell weld for the first cask loaded at ANO was initially considered to have been caused by lamellar tearing based on visual observations of the crack by the welders before the crack was repaired. No other data were available other than the observation that this crack was similar in appearance to the second crack observed during welding of the shield lid-to-shell weld for the third cask loaded at ANO.

The team (1) observed differences between the welding conditions at ANO and those at Point Beach and Palisades that were judged to have a possible influence on the cause of the cracks observed at ANO; (2) evaluated parameters which affect the risk of hydrogen induced cracking (HIC), including hydrogen level, microstructure, and stress; (3) tested samples of the ANO shell material and determined that they had excellent resistance to lamellar tearing; and (4) re-examined a replica of the second weld crack using light microscopy and scanning electron microscopy and concluded that the crack had the appearance of a HIC.

The team concluded that the second crack at ANO appeared to have been HIC. The team also concluded that there was no evidence to indicate that other potential causes of cracking, including lamellar tearing, undocumented welds discovered as a result of the team investigation, or small sulfur inclusions found in MSB shell material at ANO, contributed to the cause of the cracking observed at ANO.

The staff reviewed information submitted by the Owners Group regarding the MSB weld cracking and additional information gathered during inspections and site visits. The existence of undocumented welds observed on several MSB shells was confirmed during inspections conducted at SNC's fabricator site and at ANO. The staff accepted the Owners Group's conclusions as reasonable explanation for the cracks [12].

## 3. POTENTIAL FOR DELAYED CRACKING

The Owners Group determined that HIC of the MSB closure welds was possible because a sufficient combined severity of the following three conditions may have existed during the welding of previously loaded casks and, therefore, previously loaded casks may have been susceptible to HIC. The conclusions were based on the following: (1) the hydrogen content of the welding consumables, particularly for ANO and Palisades, was high (15.0 - 15.9 ml $H_2$ STP/100g); (2) the microstructure of the selected steels welded for previously loaded VSC-24s were judged to have been in the susceptible range for embrittlement by hydrogen and likely to contain martensite; and (3) the joint configuration for these welds is recognized as being highly constrained so that residual stresses are expected to be at, or near, the yield level.

The Owners Group estimated the maximum expected delay times for HIC of SA-516, Grade 70 steel, at 3 hours. The delay time is the time between completion of the weld and the onset of cracking. Their computed estimates considered: (1) data gathered from a literature review of delay times associated with HIC; (2) an understanding of factors that affect the onset and continuation of cracking such as the influence of temperature and alloy content on hydrogen diffusion rates; (3) the material/weld strength; and (4) volume of the weld -- the time for escape (loss of a given fraction of the hydrogen to the environs) increases as the square of the hydrogen diffusion path.

The Owners Group compared the estimated maximum expected delay times with the elapsed times between completion of a weld pass and inspection of that pass and concluded that it was unlikely that delay times would have exceeded the actual inspection time intervals. The Owners Group further supported its conclusion with the observation that cracking was detected at ANO within an elapsed time of 30 minutes after welding.

The staff reviewed information provided by the Owners Group and independently reviewed additional literature regarding HIC and agreed with the three principal factors that promote HIC as discussed by the Owners Group. The NRC staff also identified additional factors that may affect the potential for HIC of the MSB closure weld including: (1) use of low-sulfur steels for better fracture toughness; (2) strict controls on selected weld parameters; (3) notch effects and higher local stresses caused by poor fit up; (4) differences in heat affected zone (HAZ) microstructure due to differences in cooling rates; and (5) initial temperatures of the welds [12].

While many of the factors that affect delay times described by the Owners Group were accepted, the staff did not accept 3 hours as the maximum delay time for cracking. While the initiation of cracks of significant size would most probably have been within the 3-hour delay time, the staff determined that cracking may have occurred over longer time periods.

On previously loaded casks, the staff concluded that (1) conditions that promote HIC may have existed during welding, (2) welds of both the structural lid and the shield lid may have been susceptible to HIC, and therefore, (3) that HIC may have occurred in those welds. In addition, the staff concluded that delayed HIC behaviour could have occurred both before and after completion of weld inspections performed on previously loaded MSBs.

## 4. CORRECTIVE ACTIONS

The Owners Group developed corrective actions (CA) to address each root cause and prevent recurrence. In general, those actions involved modifications to weld processes, non-destructive examinations, and other quality or operational changes. Table II is a composite summary of the corrective actions that address welding for unloaded MSBs.

The staff reviewed the information provided by the Owners Group and independently evaluated the proposed corrective actions related to the weld cracking. The corrective actions described in Table II are based on the root causes identified for the known instances of cracking. The four corrective actions for the first root cause, defects in shell material and undocumented welds, are based on identifying defects present in the MSB wall or preventing undocumented welds (CAs 1-4). The staff concluded that the proposed corrective actions for the first root cause provide adequate measures to detect and properly address potential base metal defects or undocumented welds in existing unloaded MSBs. For MSBs fabricated in the future, the proposed corrective actions should minimize the potential for base metal defects and undocumented welds.

The two corrective actions for the second root cause, improper fit-up of lid and backing ring, are intended to check proper fit-up (CA 5), and if required, repair by welding to fill in unacceptable gaps (CA 6). These measures serve to reduce the residual stresses generated during cooling of the weld that can distort the structure; exacerbate an existing welding flaw, e.g., propagate a crack; and make a weld more susceptible to HIC. The staff concluded that the proposed corrective actions for the second root cause should adequately prevent poor fit-up conditions during future closure lid welding.

The corrective actions for the third root cause, moisture contamination of weld, should preclude and remove water in the area surrounding the weld joint to keep it from becoming entrained into the closure welds and causing welding flaws, including porosity. The staff concluded the proposed corrective actions for the third root cause are adequate to prevent future moisture contamination of welds. The corrective actions for the fourth root cause, HIC, are preventive measures. The staff evaluated the changes to the welding procedures proposed by the Owners Group to address HIC and concluded that these changes will (1) allow hydrogen to diffuse out of base metal before and after welding to reduce susceptibility to HIC (CAs 10 and 11); (2) reduce the cooling rate and thus hardness, which reduces susceptibility to HIC (CAs 10 and 11); (3) improve fracture and notch toughness (CAs

10 and 11); (4) reduce hydrogen introduction into weld metal from electrodes to reduce susceptibility to HIC (CA 12); and (5) reduce residual stresses to reduce susceptibility to HIC (CAs 10, 11, and 13). If, despite the above corrective actions, HIC still occurs, a hold period of 2 hours between the completion of the weld and inspection (CA 14) will allow time for delayed HIC to develop such that it will be detectable by the various NDE methods used on the VSC-24 closure welds. For future casks, steels with a lower carbon equivalent, which are less susceptible to HIC, will be used (CA 15).

TABLE II. OWNERS GROUP CORRECTIVE ACTIONS FOR UNLOADED MSBs AND MSBs TO BE MANUFACTURED IN THE FUTURE

| **Root Cause** | **Corrective Action (CA)** | **Applicability** |
|---|---|---|
| Defect in shell material; undocumented welds | 1. Acid etch top 4 inches of cask<br>2. UT[a] per ASTM A435[b]<br>3. Certification that MSBs meet the design and terms and conditions of the CoC[c] and are in conformance with the Safety Analysis Report and Safety Evaluation Report including any referenced standards, criteria, or requirements[d]<br>4. Use of low sulfur, calcium-treated, vacuum-degassed steel | 1 - 2. Unloaded MSBs already manufactured<br>3. Unloaded MSBs<br>4. MSBs manufactured in the future |
| Improper fit-up of lid and backing ring | 5. Proper fit-up of assembly<br>6. Manual welding to fill-in unacceptable gaps before automated welding | 5 - 6. All future lid welding |
| Moisture contamination of weld | 7. Ensure water level in MSB is adequately below the shield lid via partial drain of MSB<br>8. Vent or inert airspace beneath shield lid<br>9. Preheat weld area to 93°C (200°F) | 7 - 9. All future shield and structural lid welding, including lid-to-lid and valve cover fillet welds |
| Hydrogen induced | 10. Addition of 93°C (200°F)[e,f] preheat<br>11. Addition of 93°C (200°F) postheat, 1h<br>12. Low hydrogen welding electrodes (<10 ml $H_2$ /100g deposited weld metal)<br>13. Large tack welds/balanced weld sequence to prevent movement of lids and better distribute shrinkage forces from cooling of weld<br>14. 2-hour delay before inspection<br>15. Use of materials with lower carbon equivalent values | 10 - 14. All future shield and cracking structural lid welding, including lid-to-lid and valve cover fillet welds<br>15. MSBs manufactured in the future |

a. UT = ultrasonic examination or ultrasonic test.
b. Palisades indicated an alternate, American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel (B&PV)Code, Section III, NB-2532.1 [13].
c. CoC = certificate of compliance.
d. NRC CAL Supplements 97-7-002A, 97-7-003A, and 97-7-004A.
e. SNC evaluation indicated the effect on time to drain down (VSC-24 Technical Specification 1.2.10) is negligible.
f. Meets requirements of ASME B&PV Code, Section III, Sub-Section NC [13].

The staff found that the 2-hour hold period prior to performing the final weld inspection for future MSB loading is an acceptable practice. The revised welding procedures ensure that temperatures are maintained at or above 93°C (200°F) during welding. At these temperatures, the diffusivity of hydrogen is increased, leading to the escape of hydrogen from the weld to the surrounding metal and to the ambient air. This decreases the available hydrogen and thereby decreases the likelihood of cracking. However, it also leads to a decrease In the delay time for any cracks that might form. Nevertheless, the detection of any significant cracks that may have formed is ensured by the fact that the ultrasonic test (UT) of the structural lid closure weld is not to be conducted until 2 hours after the completion of the final pass. As the root pass is regarded to be the most likely pass to crack, the beneficial effects of elevated temperatures will have been present for many hours from the time that this pass is completed to the time of the inspection. This gives high confidence that sufficient time has been allowed for the initiation of cracks (that may form) prior to conducting the UT inspection. The staff concluded that CAs 10 through 15 for the fourth root cause, HIC, should effectively reduce the susceptibility of future welds toHIC and allow for detection, should they occur.

On the basis of concerns with the weld cracking events and experience that has shown that welding processes are not always reliable, NRC staff sought reasonable assurance to confirm that the MSB closure has sufficient integrity to meet the regulatory requirements. In response to NRC staff's request for testing, surveillance, or monitoring to confirm that the corrective actions were effective, the Owners Group committed to volumetrically examine the structural lid-to-shell closure weld during future cask loadings and for previously loaded casks. Because volumetric examination has the capability to examine the entire volume of the weld, it would also verify the presence of a structurally sound weld. The ASME B& PV Code recognizes two techniques for volumetric inspection: radiography (RT) and UT. Because of physical limitations in accessing the examination area, the Owners Group determined RT of the weld was impractical. Therefore, the Owners Group pursued UT of the structural lid weld.

Development of the UT process involved two major tasks: (1) demonstration of a UT technique on a full diameter MSB mock-up containing imbedded flaws; and (2) development of a methodology to disposition flaw indications located by UT. The process included the following major steps: (1) development of flaw size screening criteria; (2) performance of material fracture toughness testing of weld metal, HAZ, and base metal at -18°C (0°F) to determine the physical properties for the welding techniques used at each site; (3) submission of weld coupons to NRC for independent material testing; (4) construction of an MSB mock-up with implanted flaws of various known sizes, orientations, and locations within the structural lid-to-shell weld; (5) development of procedures for the UT of the lid-to-shell weld; and (6) demonstration of theUT technique duringNRC inspection.

The Owners Group successfully demonstrated its UT technique and the NRC staff's findings were documented in an inspection report [14]. The staff concluded that the time-of-flight diffraction UT technique developed by the Owners Group [15] will confirm the integrity of the welded joint through detection of subsurface flaws such as cracking and weld process-induced flaws, e.g., slag inclusions, incomplete fusion, incomplete joint penetration.

The staff evaluated the methodology for dispositioning flaw indications found by UT [16,17]. Flaw indications will be characterized, evaluated for flaw proximity per the criteria of ASME B&PV Code, Section XI, IWA-3300 [18], and compared to screening criteria. The staff conducted independent confirmatory material testing on weld samples provided by the Owners Group and independently assessed the fracture toughness. The allowable flaw sizes screening criteria were calculated in accordance with ASME B&PV Code, Section XI. IWB-3611 or IWB-3612 and based on lower bound fracture toughness testing data and upper bound residual stress value at the yield stress level 262 MPa (38 ksi) for the material. For a flaw larger than these screening limits, a flaw specific evaluation, using the methodology of ASME Section XI, IWB-3600, may be performed to assess acceptability. Flaws that are unacceptable will be removed or reduced to acceptable limits, the area repaired, and the area re-examined to determine acceptability. The staff concluded that this approach provided reasonable assurance to ensure the adequacy of the structural lid closure weld. The screening criteria are applicable to both currently loadedMSBs and those that will be loaded in the future.

## 5. REGULATORY ACTIONS

Based upon the staff's evaluation [12] of the Owners Group root causes and corrective actions reports, the CALs for Point Beach and ANO [19,20] were closed, and additional storage casks were successfully loaded at each site. While closing of the CAL for Palisades is pending the submittal of additional information, the licensee is proceeding with its UT examinations and evaluations of previously loaded casks.

As a result of lessons learned from these cracking events, the NRC is evaluating the generic implications of this issue in the licensing process for current cask designs. At question is whether PT (liquid penetrant surface examination or liquid penetrant test) examination of weld root and final pass layers, along with helium leak test, is sufficient to provide reasonable assurance of the containment boundary closure weld joint integrity or whether UT examination is necessary for detection of subsurface flaws such as cracking and weld process-induced flaws (e.g., slag inclusions, incomplete fusion, incomplete joint penetration). Because austenitic stainless steel does not have a nil ductility transition temperature, the weld can sustain "large" flaws without a concern for flaw growth. This allows the use of either UT or PT, although both would have limitations on detectable flaw size and both would accept less than critical flaws. Therefore, for austenitic stainless steels, NRC is considering multiple pass dye penetrant techniques such that, along with root and final layer PT, there would be sufficient intermediate layer PT examinations to detect critical flaws. The maximum undetectable flaw size would have to be less than the critical flaw size. Also, a stress-reduction factor of 0.8 would be applied to the weld design to account for any uncertainties. ASME B&PV Code, Section XI methods would be used to calculate the critical flaw size. Other materials would be considered on a case-by case-basis.

## 6. CONCLUSIONS

The staff concluded that the VSC-24 Owners Group proposed corrective actions, (modifications to the welding procedures, NDEs, and other material, quality, and operational changes) were adequate to prevent recurrence of the root causes of the identified welding defects. In addition, volumetric examination of the structural lid-to-shell weld joint will provide reasonable assurance to confirm the presence of a structurally sound weld for both future MSB loadings and previously loaded MSBs. The methodology to address flaws identified by UT requires that if unacceptable conditions are identified, the flaws are required to be evaluated and/or repaired in accordance with the licensee's quality assurance program and Owners Group commitments.

The importance of identifying necessary welding procedures and examination techniques early in the design and approval process has been clearly demonstrated. The need for UT examination of the final containment boundary closure welds on new storage cask design approvals is receiving closer scrutiny at NRC.

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# CHARACTERIZATION OF SPENT FUEL ASSEMBLIES FOR STORAGE FACILITIES USING NON DESTRUCTIVE ASSAY

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**Abstract**

Many non destructive assay (NDA) techniques have been developed by the French Atomic Energy Commission (CEA) for spent fuel characterization and management. Passive and active neutron methods as well as gamma spectrometric methods have been carried out and applied to industrial devices like PYTHON™ and NAJA. Many existing NDA methods can be successfully applied to storage, but the most promising are the neutron methods combined with on line evolution codes. For dry storage applications, active neutron measurements require further R&D to achieve accurate results. Characterization data given by NDA instruments can now be linked to automatic fuel recognition. Both information can feed the storage management software in order to meet the storage operation requirements like: fissile mass inventory, operators declaration consistency or automatic selection of proper storage conditions.

## 1. INTRODUCTION

France, which is a major actor in nuclear industry, owns plants and facilities that cover the whole cycle of nuclear materials. Part of this cycle is: handling, reloading, transporting, storing and reprocessing of spent fuels. So, for a long time, the French CEA has studied technologies for non destructive measurement and characterization of spent fuel assemblies. Many R&D programmes have been undertaken in partnership with major French nuclear industry companies like EdF or COGEMA, the industrialization being performed by EURISYS MESURES[1]. Various characterization techniques have been developed for: safety criticality controls, core loading checking or Safeguards concerns. They have been implemented in various NDA instruments. In a first step, the physical methods involved in these NDA instruments are briefly described before their potential application for storage purposes is discussed. All known NDA techniques for spent fuel characterization use spontaneous or induced nuclear emissions. Gamma rays as well as neutron emissions are used [1].

## 2. PHYSICAL PRINCIPLES OF SPENT FUEL CHARACTERIZATION

### 2.1. Passive neutronic emission

Since the spontaneous neutron emission is linked to the burnup of the fuel with a power law, the burnup determination with passive neutron measurement is extremely accurate. The correlation law is:

$$BU = aNE^{b} , \qquad (1)$$

in which $a$ is a constant slightly dependant on initial enrichment and $b$ another constant close to 0.02 for usual irradiation histories. In addition, the constants $(a,b)$ can be calculated using an evolution code taking into account initial enrichment and actual irradiation history of the assembly to

[1] The nuclear measurement company of the COGEMA group

be measured. As a result, relative error on neutron emission $^{\Delta NE}/_{NE}$ from the measurement leads to small burnup relative error:

$$\frac{\Delta BU}{BU} \cong 0.02 \frac{\Delta NE}{NE} , \tag{2}$$

## 2.2. Induced neutron emission

In nuclear fuel that contains fissile material, neutron emission can be induced by an external neutron flux. It could also be created by a high energy gamma intense flux (photo-fission) but, today, this technique has never been applied to spent fuel because of its cost and complexity. Neutron induced emission *INE* is linked to the multiplying factor $k_{eff}$ by the relation:

$$INE = a \frac{Keff}{(1 - Keff)} , \tag{3}$$

Using pulsed neutron irradiation, the neutron detector measures on one hand the passive emission and on the other hand both induced and passive contributions. This NDA method is used for initial $^{235}U$ enrichment or for MOX characterization.

## 2.3. Total gamma emission

Gamma emission comes from fission products and activation products. After discharge of spent fuel assemblies from the reactor core, short lived fission and activation products are responsible for most of the total gamma emission which decreases very fast in the spent fuel. After several months, the total gamma emission *TGE* can be linked to the cooling time with the following correlation law:

$$CT = a(BU) \cdot \left( \frac{TGE}{BU} \right)^{b(BU)} , \tag{7}$$

$a_{(BU)}$ and $b_{(BU)}$ are quadratic functions of *BU*, determined by fitting of parametrical calculation results. Since the total gamma emission can be measured with simple and inexpensive detectors (e.g. ionisation chambers), the cooling time can be simply estimated.

Contrary to neutronic emission, gamma emissions can be collimated. This property is used to determine the relative irradiation profile along the assembly, in order to measure the burnup of the fuel extremities for safety criticality purpose. The average *BU* being known (using a neutron method), the total gamma profile is measured along the assembly with a collimated detector. The result of the scanning measurement is an array *TGE(z)*. The extremity burnup $EBU_{(0,z0)}$ is defined by the relation:

$$EBU_{(0,z0)} = BU \cdot \frac{z}{z_0} \cdot \int_0^{z0} TGE(z) \cdot dz \Big/ \int_0^{z} TGE(z) \cdot dz \cdot F(CT) , \tag{4}$$

This relation assumes that *TGE* is mainly composed of burnup proportionally produced gamma emitters. Actually, a correction factor *F(CT)* has to be used for short cooling times *CT*, because short lived fission products induce non proportionality. The following relation is convenient for PWR reactors:

$$F(CT) = 0.5 \cdot CT^{0.1} , \tag{6}$$

## 2.4. Gamma spectrometry

Many gamma emitters have interesting properties in order to characterize the spent fuel. Tables I and II, give an overview of several relevant isotope abundance and isotopic ratios to determine burnup and cooling time. Convenience of the different isotopes is displayed, with regard to the range of cooling times. Convenience of the gamma spectrometric methods is very dependent on the characteristics of the spent fuel. They have to be carefully selected to produce proper results. Anyway, because the slope of the correlation laws is always smaller for gamma emitters than for neutron emission, burnup determination using passive neutron counting leads to more accurate results than gamma measurements.

TABLE I. OVERVIEW OF GAMMA SPECTROMETRIC BURNUP DETERMINATION

| Isotope | Correlation law[2] : | Cooling time[3] | | |
|---|---|---|---|---|
| | (example PWR 17x17 IE 3% CT 3 years) | 0 to 90 days | 90 to 5,000 days | Over 5,000 days |
| $^{134}Cs$ | $a{\cdot}BU^2$ (a : 0) | + | - | 0 |
| $^{137}Cs$ | $a{\cdot}BU$ (a : 3,000) | + | + | + |
| $^{154}Eu$ | $a{\cdot}BU^b$ (a : 5, b : 2) | 0 | - | + |
| $^{134}Cs/^{137}Cs$ | $a{\cdot}BU^b$ (a : $10^{-2}$, b : 1) | ++ | + | 0 |
| $^{154}Eu/^{137}Cs$ | $a{\cdot}BU^b$ (a : $10^{-3}$, b : 1) | 0 | + | ++ |

TABLE II. OVERVIEW OF GAMMA SPECTROMETRIC COOLING TIME DETERMINATION

| Isotope | Correlation law : (example PWR 17x17 IE 3% ) | 0 to 90 days | 90 to 5000 days | Over 5000 days |
|---|---|---|---|---|
| $^{144}Ce/^{137}Cs$ | $^{144}Ce/^{137}Cs=a.exp^{b.CT}$ (a : 10, b : -0.002) | + | ++ | - |
| $^{106}Ru/^{137}Cs$ | $(^{106}Ru/^{137}Cs)/TC^{0.5}=a.exp^{b.CT}$ (a : 1 , b : -0.02) | ++ | + | - |

## 3. EVOLUTION OF THE FUEL UNDER AND AFTER IRRADIATION

The composition of the isotopes in the fuel changes during irradiation and cooling. Different nuclear reactions and decay lead to production and destruction. The evolution of the fuel components results from neutron fission captures, (n,2n) reactions and ($\alpha$, $\beta$) radioactive decay. The following differential equation describes this complex process:

$$
\begin{aligned}
&\frac{dN(t)}{dt}[A,Z] = \\
&\left[\Phi \quad \sigma_{c(El,BU)} \quad N(t)\right]_{[A-1,Z]} + \left[\Phi \quad \sigma_{n,2n(El,BU)} \quad N(t)\right]_{[A+1,Z]} \\
&+\left[\lambda_{\beta+} \quad N(t)\right]_{[A,Z+1]} + \left[\lambda_{\beta-} \quad N(t)\right]_{[A,Z-1]} \\
&+\left[\lambda_{\alpha} \quad N(t)\right]_{[A+4,Z+2]} + \left[\lambda_{TI} \quad N(t)\right]_{[A_{\text{métastable}},Z]} \qquad , \qquad (8) \\
&-\Phi\left[\left(\sigma_{c(El,BU)} + \sigma_{f(El,BU)} + \sigma_{n,2n(El,BU)}\right) \quad N(t)\right]_{[A,Z]} \\
&-\left[\left(\lambda_{\beta+} + \lambda_{\beta-} + \lambda_{\alpha} + \lambda_{TI} + \lambda_{SF}\right)N(t)\right]_{[A,Z]}
\end{aligned}
$$

2 In the correlation examples, activities are given in Curie/g, burnup in GW·d/t and cooling times in days

3 0 means « really not appropriate », - means « not appropriate »,
+ means « can be successfully used », ++ means « recommended »

In this relation, $N(t)_{[A,Z]}$ is the atomic abundance of the isotope with atomic number Z and mass number A. $\Phi$ is the neutron flux, $\sigma_c$, $\sigma_f$, $\sigma_{(n,2n)}$ are the cross sections respectively for capture, fission and (n,2n) reactions, $\lambda_i$ the decay constants for $\beta^+$, $\beta^-$, $\alpha$, spontaneous fission and isomeric transition reactions.

Contrary to decay constants, which are intrinsic values, neutron cross sections depend on irradiation conditions. In the relation (8), neutron cross sections are condensed into one energy group representative of the neutron spectrum and auto-protection effects. During fuel life these parameters change. As a result, for every kind of fuel, neutron cross sections have to be tabulated with regard to initial enrichment and burnup.

This important work has been carried out to provide the French reprocessing company COGEMA with a calculation tool CESAR .

[2]. This code is able to calculate the components and emissions of fuel taking into account their particular geometry, initial enrichment and irradiation histories. An on line version of this code has been developed for spent fuel NDA characterization purposes.

## 4. APPLICATIONS OF NDA METHODS

NDA methods have been applied by CEA in several plants and equipment. They have to be shared in two families depending whether they are applied under water or in air.

### 4.1. NDA in air

In air, gamma spectrometric methods are easier to carry out and give better results than under water. As an example, we can describe the spectrometric analysis of the spent fuel in the head end of the COGEMA La Hague reprocessing plant [1]. This device is composed of two HPGe detector with collimators. Burnup and cooling time are determined using respectively $^{134}Cs/^{137}Cs$ and $^{144}Ce/^{137}Cs$ ratio. Scanning the fuel between the detectors leads to a very accurate burnup profile measurement. Neutron passive counting combined with an on line evolution code, is also used and gives very accurate burnup determination (within 2% considering one single assembly).

It has also been shown [3], that neutron measurement in air combined with a correlation law can be used to determine the plutonium amounts of spent fuel using a relation like:

$$M_{Pu} = aNE^b \ , \tag{9}$$

where $M_{Pu}$ is the plutonium mass, $NE$ the neutron emission rate and $a$ is a function of the initial enrichment $IE$ and cooling time $CT$ in the following relation:

$$a = a_0 + (a_1 + a_2 CT)IE \ , \tag{10}$$

The constants $a_n$ and $b$ depend only on the assembly type and since, $a_2 << a_1 << a_0$, the influence of the cooling time and initial enrichment remains very small. This very simple method gives accurate results and it has been shown that, considering several assemblies, the total plutonium amounts can be determined within 1% for PWR assemblies. No information on irradiation histories neither than on burnup are necessary. In addition, no on line code has to be used.

### 4.2. Underwater measurements

Various techniques have been used for under water measurements but neutron measurements are particularly convenient. Two major R&D projects have been carried out by CEA giving birth to the PYTHON ™ device [4] and NAJA device [5].

*4.2.1 The PYTHON™ device*

The PYTHON device has been developed in collaboration between EDF and CEA. Its main objectives are to measure the average and extremity burnup for safety criticality purposes. The PYTHON device is a combination of:

- a passive neutron measurement;
- a collimated total gamma measurement;
- an on line evolution code.

Figure 1 shows a schematic view of the two measurement heads that operate on top of the storage racks in the NPP ponds. The figure is a graphic output of a MCNP model of the measurement heads [6]. It is used to optimize the head's design, to precalculate the measurements yields and to parametrically calculate the multiplication factor $k_{eff}$ of the fuel taking into account boron concentration in the water and burnup.

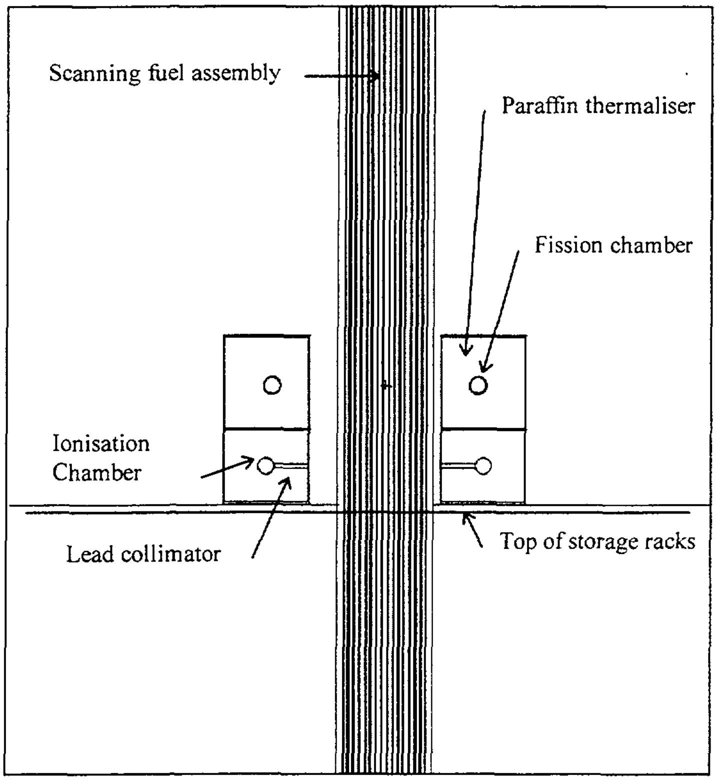


*FIG. 1. The two detection heads of the PYTHON™ system*

The neutronic yield is about $10^{-5}$ c/n·assembly$^{-1}$, so in order to achieve good statistical accuracy with low burnup, the detection heads are equipped with high efficiency fission chambers (1 c/n.cm$^{-2}$). The gamma detectors are simple ionization chambers with $10^{-9}$ A/Gy.h$^{-1}$ efficiency.

Since the PYTHON device is intended to measure the average burnup, the neutron signal has to be representative of the entire fuel assembly. This means that contributions to the average signal have to originate from the whole fuel. FIG. 2 shows an example of the radial importance function of the fission chamber in the fuel section measured on a fuel mock-up in borated water [7]. Contrary to the gamma emissions that are absorbed when crossing fuel pins, the neutrons detected by the fission chamber originate from almost all the fuel section. In addition, since the two head signals are averaged, the sensitiveness to radial gradient for the burnup measurement is very low.

To take into account the neutron axial profile, fuel is scanned between the two heads and signals are averaged. However, despite neutron signals are acquired along the fuel, it is not possible to get from them a burnup profile. FIG. 3 shows the axial importance function of the detectors. It is clear that almost a length of several tens of centimetres contributes significantly to the signal. As a result, the burnup profile has to be measured with the collimated total gamma detector and extremity burnup calculated using both gamma profile and average burnup using the relation 8.

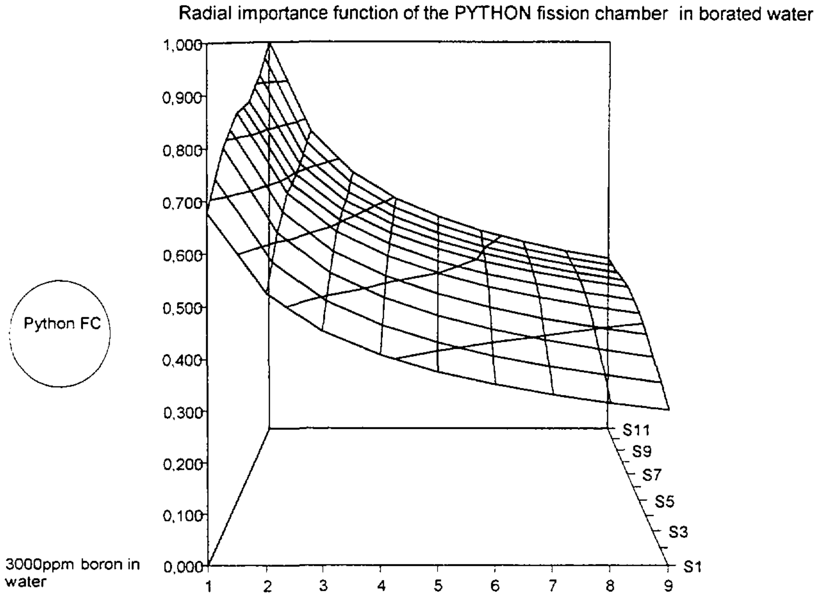


*FIG. 2. Radial importance of Python fission chamber (9x11 pins mock-up)*

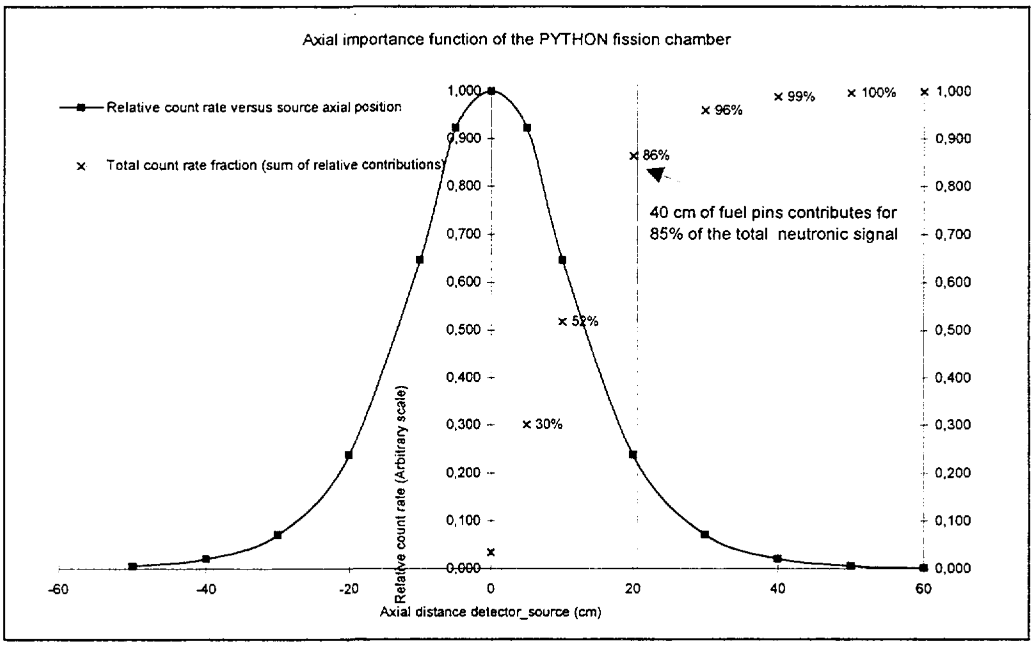


*FIG. 3. Axial importance of Python fission chamber*

For safety criticality purpose, the irradiation histories are supposed to be known as well as the initial components of the fuel. They are used as input data by the on-line evolution code that determines a correlation law (BU = f(NE)) for each fuel assembly. As a result, no standardization is required to determine the relevant correlation law to apply. Only a calibration is required to measure the detector yields. In order to avoid mistakes, the measured yields values are confirmed with MCNP calculations.

The PYTHON system has intensively been qualified using a prototype device with active mode capability at the Tricastin NPP and with industrial systems manufactured by EURISYS MESURES at the Gröhnde and Brökdorf NPPs. Table III summarizes the qualification range.

TABLE 3. PYTHON QUALIFICATION RANGE

| | **Tricastin** | **Gröhnde** | **Brökdorf** |
|---|---|---|---|
| Number of assemblies | 35 | 50 | 35 |
| Fuel types | UOX PWR 17 x 17 | UOX PWR 16 x 16 | UOX PWR 16 x 16 |
| Initial enrichments | 1.8% to 3.7% | 2.1% to 4% | 1.8% to 3.7% |
| Cooling Time | 80 days to 7.5 years | 60 days to 6 years | 500 days to 3.5 years |
| Reactivity ($k_{eff}$) | 0.4 to 0.7 | Passive mode only | |

The accuracy evaluated by a comparison with declared values for burnup and cooling time and with calculations for $k_{eff}$ are as follows: on average burnup within 2%, on cooling time within 15% and on the multiplying factor $k_{eff}$ within 3%.

At the moment another industrial system is delivered by EURISYS MESURES for the Gösgen NPP in Switzerland. Further R&D projects have been focused on extensions of the PYTHON capabilities in order to use burnup and reactivity measurements for core loading checks.

### *4.2.2. The NAJA device*

The main objectives of the NAJA device consist in developing a measurement device which combines nuclear methods and video control in order to evaluate the physical characteristics of each fuel assembly (burnup, reactivity, initial enrichment, etc.) and to automatically validate the final core loading. Such a device would be placed on the passage of the fuel assembly between the storage pond and the reactor building. It should be useful for core loading conformity control and on-line core monitoring.

The NAJA device is able to automatically determine (with non-destructive measurements) for each assembly:

- the nature of the fuel element (fresh or irradiated, UOX or MOX);
- the presence and the kind of neutron absorber;
- the initial enrichment in $^{235}U$ for fresh UOX assembly;
- the identification number.

This information allows to characterize the fuel assemblies accurately and to be sure, without human factor hazard, of the core loading conformity.

The NAJA device has been optimized in order to take into account severe constraints (no influence on the loading or unloading schedule, no need for human interface, no impact on operation). Such optimization has led to the following conclusions:

- without human interaction, the device controls each assembly which goes to the reactor building (loading) or which goes to the storage building (unloading);
- the device is located on the passage of fuel elements on the pond building near the transfer tube;
- three nuclear methods are applied to cover the whole fuel assembly panel (active neutron interrogation, passive neutron counting, gamma spectrometry);
- an ultra-sonic probe is used to monitor the different parts of the fuel element (foot, beginning of the fissile length);
- a video system linked to Optical Character Recognition (OCR) software, leads to an automatic reading of the fuel assembly number.

In fact, two video systems are planned to be used: the first one is linked to the NAJA device in the storage pool and the second one is linked to the loading machine in the reactor building. These two video systems are necessary in order to be sure of the good appropriateness between the position X, Y of the fuel assembly in the core and its physical characteristics (UOX or MOX fuel, burnup, initial enrichment in $^{235}U$, kind of absorbent etc.). Successful tests of the OCR software have been performed even for "black" (e.g. corroded) fuel assembly numbers.

One of the big interests of the NAJA device consists of the combination between the nuclear measurements, the ultra-sonic probe and the video system which allows to associate each fuel element placed in the core to a fuel identification number and its physical characteristics without any information coming from the operator. The combination between the ultra-sonic probe and the nuclear device leads us to assert the reproducibility and the reliability of the fissile column measurement.

The feasibility study of the device has been made using experimental results from the PYTHON device and theoretical calculations for its optimization. The panel of the fuel assembly characteristics which have been taken into account is large and representative of the French fuel cycle:

- average burnup of the spent fuel from 6,000 MW·d/tU to 48,000 MW·d/tU;
- cooling-time varying from 1 to 90 days;
- initial enrichment in $^{235}U$ for UOX assembly varying between 3 and 4 %;
- nature of neutron absorbent: pins containing silver - indium - cadmium and/or pins containing silver - indium - cadmium and boron carbide ($B_4C$).

The study indicates the good performances which should be obtained with the NAJA device (for example +/- 2 % as a global uncertainty at 2 standard deviations on the absolute average burnup evaluation or +/- 1 % (in relative) as uncertainty at 2 standard deviations on the initial or residual enrichment evaluation) without any influence on the loading or unloading timing. Detailed results of this feasibility study are given in reference [5].

Potential uses of the NAJA device derive directly from its main functions:

1. The core conformity control which allows us to increase the safety level of the plant significantly;
2. The absolute and accurate burnup measurements of the irradiated fuel assemblies which allow us to improve the global availability of the power plant and to gain some investment benefits.

## 5. APPLICATION OF NDA TO STORAGE FACILITIES

The devices described below have been developed to provide NDA capabilities applied to fuel management in NPP ponds. Nevertheless, the same or similar NDA techniques could be successfully applied to fuel storage. For storage purposes the concerns are:

- establish an inventory of entering materials (direct disposal for example);
- control the consistency between the entering fuels and operators declarations (e.g. partial defect test, burnup, plutonium amounts);
- select the proper storage conditions regards to the residual reactivity of the fuels;
- prepare input data for fuel evolution during storage;
- long-term checking of the fuel integrity.

Storage application of existing NDA techniques have to be divided in two cases whether the storage would be dry or wet. The wet storage conditions are similar to reactor ponds, so, all existing NDA techniques (including active methods) can be successfully applied.

In dry conditions, all the passive measurements are easier to perform. Passive neutron counting as well as spectrometric gamma scanning would give accurate results if associated with a convenient evolution code. But, at the moment no active method have been yet applied to spent fuels in dry conditions (Cf. SAGOR Report [8]). If full characterization is necessary (including initial enrichment determination) no active technique is already available. In addition spent fuel could be placed in containers (steel bottles) that make the measurement more difficult.

The major challenge for storage application is to develop an active NDA in air. For active NDA, isotopic neutron sources, e.g. $^{252}Cf$, are used for under water measurements because of technological difficulties to operate a neutron generator under water. In dry conditions, such a generator delivering activities over $10^{10}$n/s could be easily used.

For storage operating, the OCR technique could be used to associate characterization of the fuel and its management in the storage. Using such a capability, storage can be designed to take into account the burnup credit in order to save space and costs.

## 6. CONCLUSION

Many NDA techniques have been developed for spent fuels characterization and management. The most promising for storage application are the neutron methods combined with on line evolution codes. All existing NDA methods but active one can be successfully applied to storage.

For dry storage applications, active neutron measurements require further R&D to achieve accurate results. Characterization data given by NDA instruments can now be linked to automatic fuel recognition. Both information can be fed into the storage management software.

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# CRITICALITY ANALYSIS OF PWR SPENT FUEL STORAGE FACILITIES INSIDE NUCLEAR POWER PLANTS



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**Abstract**

This paper describes some of the main features of the actinide plus fission product burnup credit methodology used by Siemens for criticality safety design analysis of wet PWR storage pools with soluble boron in the pool water. Application of burnup credit requires knowledge of the isotopic inventory of the irradiated fuel for which burnup credit is taken. This knowledge is gained by using depletion codes. The results of the depletion analysis are a necessary input to the criticality analysis. Siemens performs depletion calculations for PWR fuel burnup credit applications with the aid of the Siemens standard design procedure SAV90. The quality of this procedure relies on statistics on the differences between calculation and measurement extracted from in-core measurement data and chemical assay data. Siemens performs criticality safety calculations with the aid of the criticality calculation modules of the SCALE code package. These modules are verified many times with the aid of various kinds of critical experiments and configurations: Application of these modules to spent LWR fuel assembly storage pools was verified by analyzing critical experiments simulating such storage pools. Actinide plus fission product burnup credit applications of these modules were verified by analyzing PWR reactor critical configurations. The result of performing a burnup credit analysis is the determination of a burnup credit loading curve for the spent fuel storage racks designed for burnup credit. This curve specifies the loading criterion by indicating the minimum burnup necessary for the fuel assembly with a specific initial enrichment to be placed in the storage racks designed for burnup credit. The loading of the spent fuel storage racks designed for burnup credit requires the implementation of controls to ensure that the loading curve is met. The controls include the determination of fuel assembly burnup based on reactor records.

## 1. INTRODUCTION

Application of burnup credit methodologies to design calculations for spent fuel storage facilities inside nuclear power plants becomes more and more a must due to economic reasons.

If credit is taken for burnup space has to be left for at least one full core of maximum reactivity fuel [1-2]. The spent fuel storage pool is usually divided into at least two storage regions, therefore. One of these regions named as "Region I" in the following is designed for accommodating unirradiated and unpoisoned (i.e. without absorber material) fuel having the initial U-235 enrichment maximum allowable for the nuclear power plant of interest. The other region (or regions) named as "Region II" type storage region(s) in the following is (are) based on a minimum fuel burnup depending on the initial enrichment of the fuel to be stored. So therefore, the design of the storage racks of such a region is based on a decision criterion which provides for any initial fuel enrichment being allowed the minimum average discharge burnup that the fuel must have reached in order to be acceptable for storage in the Region II storage racks of interest. This decision criterion is given by a loading curve like that one shown in Figure 1: The region above the loading curve and this curve itself are the acceptance region for storage of fuel in the Region II racks. Fuel assemblies with average discharge burnup beneath the loading curve are not allowed to be loaded into these racks.

By definition, the loading curve takes credit for the reduction in fuel reactivity due to the burnup of the fuel. This "burnup credit" makes use of the changes in the isotopic composition of the fuel and accounts for the net reduction of fissile material and the build up of neutron absorbers in the fuel as it is irradiated. These neutron absorbers include actinides as well as other isotopes generated as a result of the fission process.

### 1.1. Criticality safety analysis procedure for region I

The criticality safety analysis procedure applied to the Region I racks consists of six major steps as summarized below (cp. Figure 2):

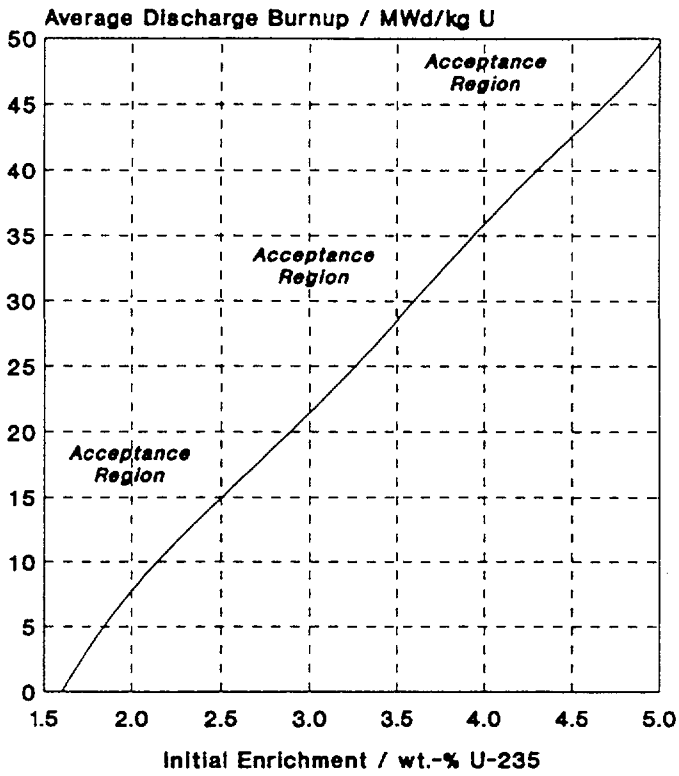


*FIG. 1. Spent Fuel Pool at Nuclear Power Plant Vandellos II: Storage Region II Loading Curve*

*(1)* Description of the fresh fuel characteristics and the fuel assembly types to be stored;

*(2)* Sensitivity analyses of the reactivity effects of the different fuel types, manufacturing tolerances and storage rack design features;

(3) Determination of a criticality safety acceptance criterion establishing sufficient subcriticality taking account of:
- the regulatory requirements;
- the bias in the applied calculation procedure as obtained from comparisons with appropriate benchmark experiments; and
- the results of the sensitivity analyses;

(4) Determination of the initial enrichment limit;

(5) Determination of the minimum boron content of the pool water required for meeting the criticality safety acceptance criterion in case of plant condition IV or V events, as defined in [1];

(6) Procedures necessary to verify that:
- the initial enrichment limit is met;
- the right fuel assembly types are stored;
- the minimum required boron content of the pool water is maintained.

So therefore, the criticality safety analysis procedure applied to Region I racks is a fresh fuel assumption procedure. In the paper on hand the attention is therefore focused upon the burnup credit procedure applied to Region II storage racks.

## 1.2. Criticality Safety Analysis Procedure for Region II

The criticality safety analysis procedure applied to Region II racks consists of nine major steps as summarized below (cp. Figure 3):

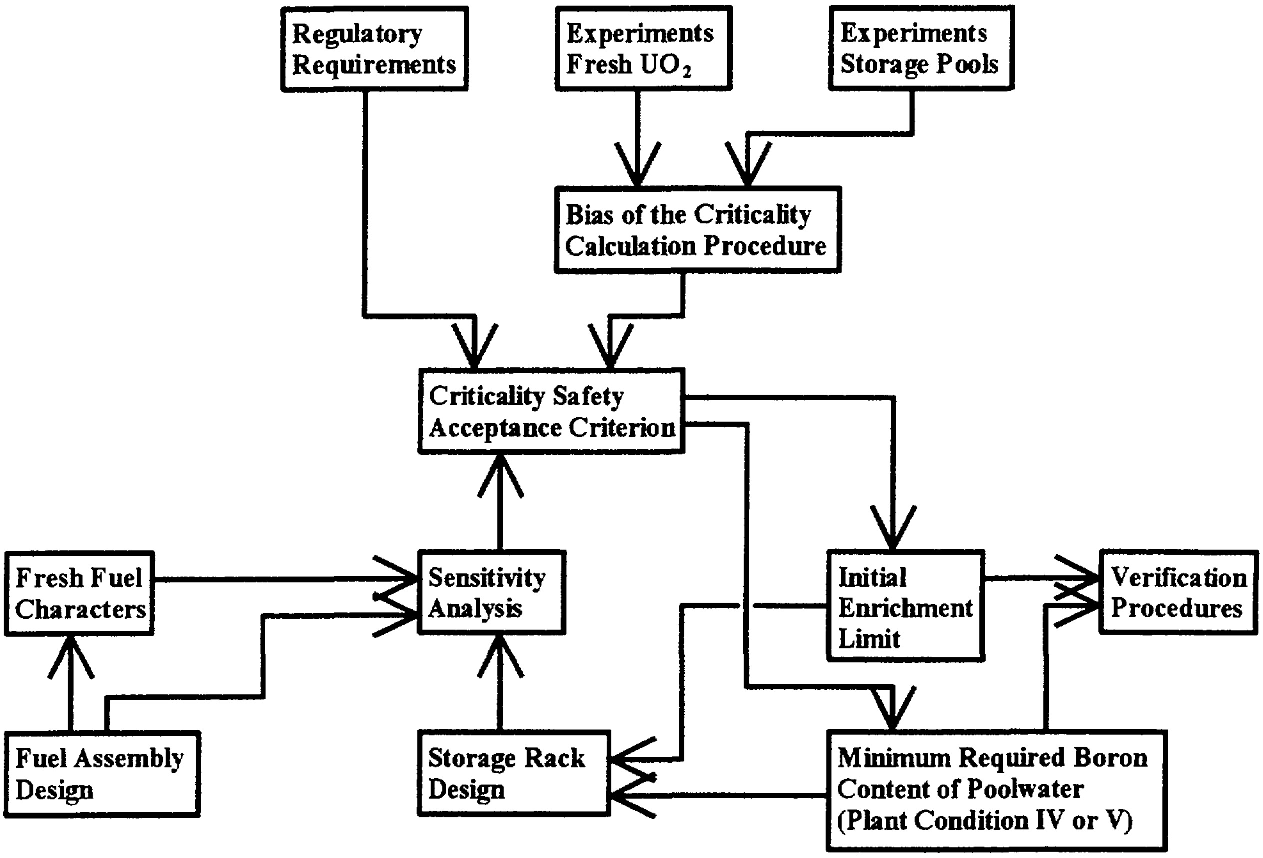


*FIG. 2. Criticality safety analysis of the region I racks (fresh fuel assumption procedure)*

(1) Description of the fresh fuel characteristics and the fuel assembly types to be stored;
(2) Depletion Analysis: Determination of the isotopic compositions of irradiated fuel for distinct initial enrichments and fuel burnup;
(3) Establishing of a conservative isotope set taking account of the uncertainty in the depletion analysis;
(4) Sensitivity analyses of the reactivity effects of the different fuel types, manufacturing tolerances and storage rack design features;
(5) Sensitivity analysis of the reactivity effect of axial burnup shapes;
(6) Determination of a criticality safety acceptance criterion establishing sufficient subcriticality taking account of:
   - the regulatory requirements;
   - the uncertainty arising from the depletion analysis;
   - the bias in the applied criticality calculation code as obtained from comparisons with appropriate benchmark experiments; and
   - the results of the sensitivity analyses.
(7) Determination of the loading curve;
(8) Determination of the minimum boron content of the pool water required for meeting the criticality safety acceptance criterion in case of plant condition IV or V events (as defined in [1]);
(9) Procedures necessary to verify that:
   - the right fuel assembly types are stored;
   - the loading curve is met;
   - individual fuel assemblies that do not comply with the loading criterion are segregated;
   - the minimum required boron content of the pool water is maintained.

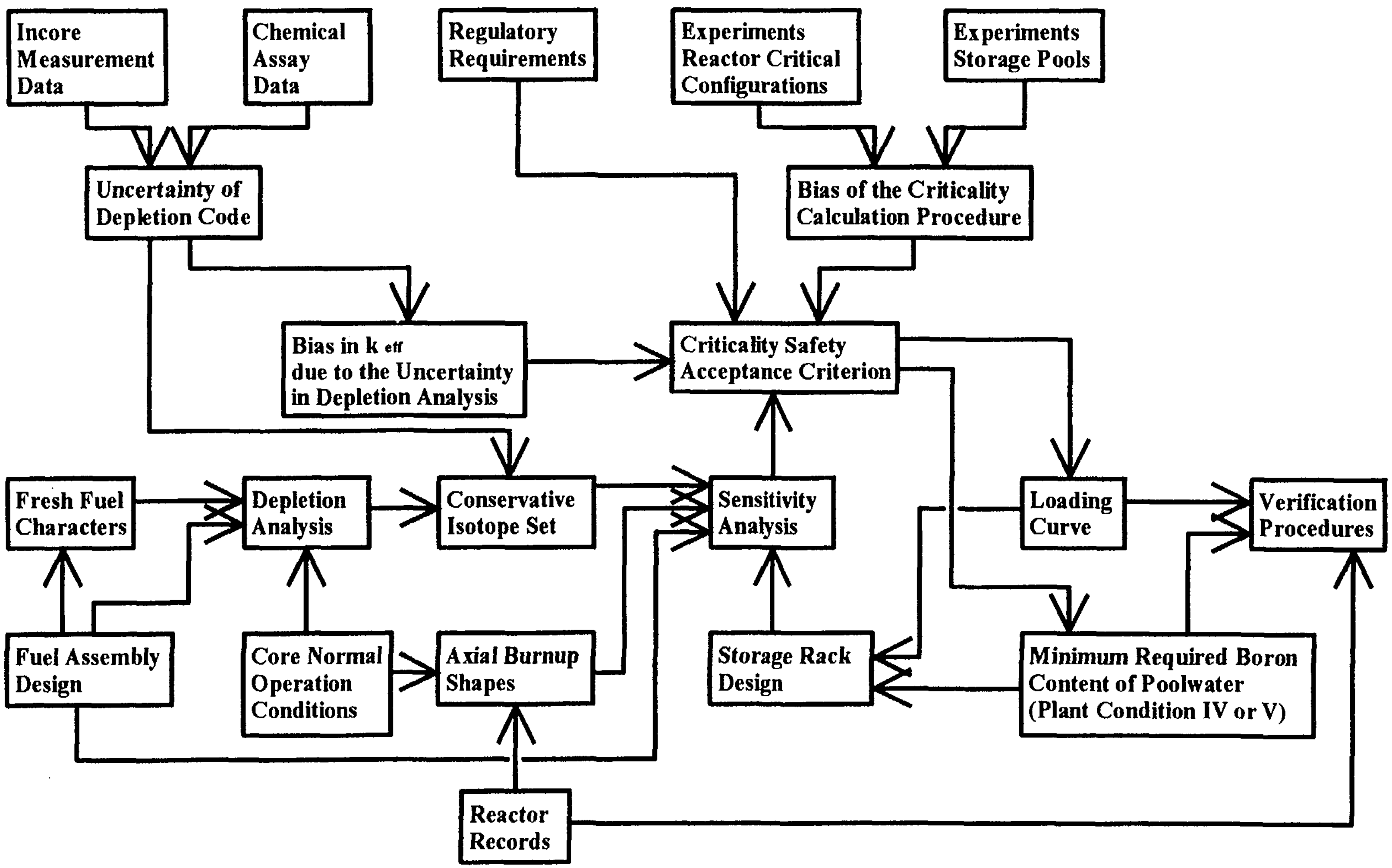


*FIG. 3. Criticality safety analysis of the region II racks (burnup credit procedure)*

Many of the above specified steps of this procedure are not different from steps that have traditionally to be taken in a fresh fuel assumption standard procedure (cp. Section 1.1). The attention is focused, therefore, on items entirely due to application of burnup credit. These items are (cp. Figure 3):

- Depletion analysis and depletion code validation;
- Establishing of a conservative isotope set;
- Criticality calculation code validation for burnup credit applications;
- Reactivity effect of axial burnup shapes;
- Estimation of the loading curve;
- Verification of the loading procedure used for the Region II storage racks.

## 2. DEPLETION ANALYSIS

Application of burnup credit requires calculation of the isotopic inventory of the irradiated fuel for which burnup credit is taken. The depletion analysis simulates the burnup of the fuel under reactor operating conditions. The result of the depletion analysis is the predicted isotopic composition of the discharged spent fuel. This composition is a necessary input to the criticality analysis.

### 2.1. Introduction

#### *2.1.1 Prediction of the isotopic composition*

The prediction of the isotopic composition is based on one's knowledge of the fuel's initial state, the irradiation history of the fuel, and the quality of the depletion code employed. The fuel's initial isotopic state is generally well defined.

Irradiation histories are obtained from reactor records. The wide variety of fuel irradiation histories makes it necessary to look for a bounding history given by the core operating conditions leading to the highest spent fuel reactivity. The determination of a Region II loading curve has to include, therefore, the evaluation of reactor operation effects which are due to partial control rod insertion during the operating cycle, presence of burnable poisons (absorbers), extended low-power operation (e.g. stretch out), and fuel assembly position inside core. These effects reveal themselves in variations of the axial power distribution in conjunction with variations of the axial moderator density and temperature distribution. These effects directly affect, therefore, the axial burnup profiles of the irradiated fuel assemblies. Selection of a "conservative profile" based on a study of a sufficient number of empirical profiles compensates for the range of irradiation histories and represents therefore, at a sufficient confidence level, a worst-case situation. The determination of the isotopic inventory of irradiated fuel assemblies can therefore be based on operating data typical of the plant of interest [3].

The quality of the depletion code employed is controlled and established through verification of that code, usually, by comparison with suitable and appropriate physics measurements and experiments. In-core reactor measurement data are very well suited to verifications of depletion codes [4]. In addition particular significance should be attached to direct comparisons of calculated to measured isotopic concentrations.

#### *2.1.2 Isotope set used for criticality analysis of spent fuel storage pools*

The isotopic composition of spent fuel consists of approximately 1300 different isotopes. Representing all these isotopes in an analytical model for critical analysis is neither practical nor essential. A limited set of nuclides has to be selected for the criticality safety analysis.

For spent fuel storage pools at nuclear power plants where soluble boron is present in the pool water application of the actinide plus fission product burnup credit level is accepted without prescribing the isotopes to be used [3-4]. However, the isotopes to be used should be selected on the basis of their reactivity worth and their nuclear and chemical stability. In order that the selected set of isotopes is enveloping with respect to the spent fuel reactivity the following essentials should be observed:

- Isotopes which have a significant positive reactivity worth (U-235, Pu-239, and Pu-241) must be included in the burnup credit safety analysis;
- Stable nuclides or radionuclides with half-lives much greater than the maximum possible cooling time of the spent-fuel may be included in the burnup credit safety analysis;
- Utilization of radionuclides with half-lives not much greater than the maximum possible cooling time has to be justified by examining the spent fuel reactivity as a function of cooling time. Such an examination is required in any case, if only because of the accumulation of Pu-239 due to the decay of Np-239.

The isotopes usually used therefore in spent fuel pool actinide plus fission product burnup credit criticality analyses are given in Table I [5-8].

The concentration and hence the contribution of the isotopes to neutron absorption, resulting in either fission or simple neutron capture reactions are dependent on cooling time. In the burnup credit methodology applied to spent PWR fuel storage pools the fission product isotopes are frozen at the concentrations existing at the time of discharge from core, except for I-135 and Xe-135 which are not considered because of their small half-lives. Consequently, as exemplified in Table II by the case of irradiated KONVOI $UO_2$ fuel assemblies stored in Region II type storage racks typical of KONVOI plants, among the fission products specified in Table II only the isotopes Rh-103, Xe-131, Nd-143, Sm-149, and Sm-151 show a considerable reactivity worth for all the burnups specified in Table II, whereas the isotopes Tc-99, Cs-133, Sm-150, and Sm-152 have a significant reactivity worth only in the range of the higher values of these burnups, cf. Figure 4. These burnups correspond to the loading curve of the Region II KONVOI fuel storage racks (cp. Figure 14). As can be seen from Figure 4, the reactivity worths of the isotopes Nd-145 and Pm-147 are of minor importance, and, as can be seen from Table II, the reactivity worths of all the remaining isotopes are more or less negligible. So therefore, as regards depletion code validation for burnup credit applications to spent fuel storage pools the attention can mainly be focused on the verification of the calculated inventory of actinides plus a few fission products.

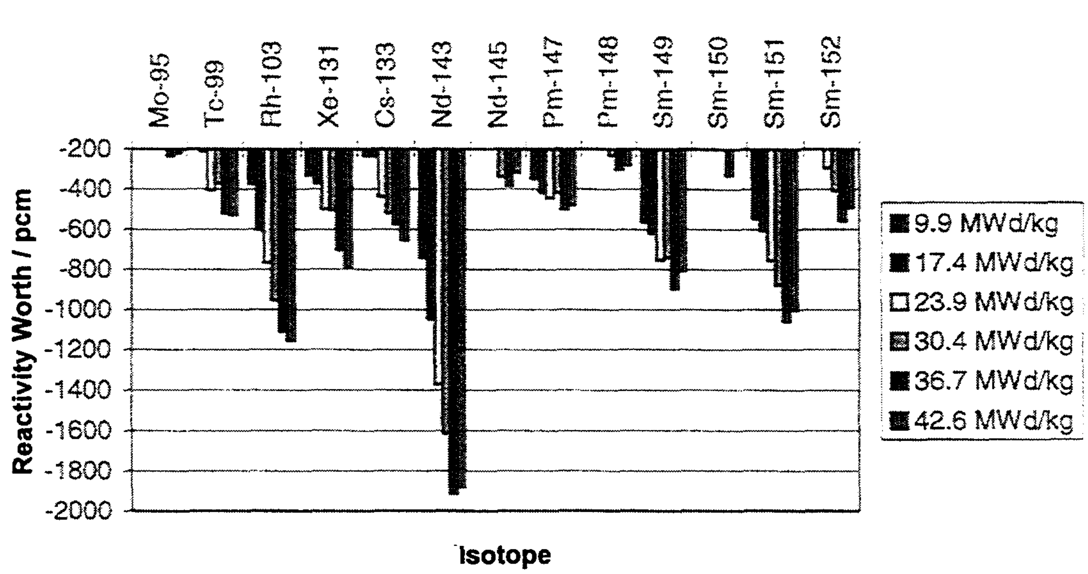


*FIG. 4. Storage of Spent KONVOI $UO_2$ Fuel in Storage Racks of the Region II Type: Reactivity Worths of Fission Products at the Time of Reactor Shut Down (cp. Table II)*

## 2.2. Depletion Code Validation

All the depletion codes usually employed for burnup credit applications [7] are extensively verified for burnups up to about 40 MW·d/kg U (cf. [9], Table I). In recent years, however, great efforts were made to verify depletion codes for high burnups as well as long cycle lengths.

TABLE I. ISOTOPES USED IN SPENT PWR FUEL POOL ACTINIDE PLUS FISSION PRODUCT BURNUP CREDIT CRITICALITY ANALYSIS

| | Isotope | $T_{1/2}$ 1) | $\sigma$ (n, $\gamma$) 2) / barn | Comment |
|---|---|---|---|---|
| **Major Actinides** | U-235 | $7.038 \cdot 10^8$a | | |
| | U-238 | $4.468 \cdot 10^9$a | | |
| | Pu-239 | $2.411 \cdot 10^4$a | | |
| | Pu-240 | 6550a | | |
| | Pu-241 | 14.4a | | |
| | Pu-242 | $3.763 \cdot 10^5$a | | |
| | Cm-243 | 28.5a | | for MOX fuel |
| | Cm-245 | 8532a | | for MOX fuel |
| **Minor Actinides** | U-234 | $2.446 \cdot 10^5$a | | |
| | U-236 | $2.342 \cdot 10^7$a | | |
| | Np-237 | $2.14 \cdot 10^6$a | | |
| | Pu-238 | 87.74a | | |
| | Am-241 | 432.6a | | |
| | Am-243 | 7370a | | |
| | Cm-244 | 18.11a | | for MOX fuel |
| **Fission Products** | Mo-95 | $\infty$ | 14.5 | |
| | Tc-99 | $2.1 \cdot 10^5$a | 19 | |
| | Ru-101 | $\infty$ | 3.1 | |
| | Rh-103 | $\infty$ | 11 and 135 | |
| | Ag-109 | $\infty$ | 4.5 and 89 | |
| | Cd-113 | $9 \cdot 10^{15}$a | 19910 | for MOX fuel |
| | Cs-133 | $\infty$ | 2.5 and 26.5 | |
| | Cs-135 | $2.10^6$a | 8.7 | |
| | Nd-143 | $\infty$ | 325 | |
| | Nd-144 | $2.1 \cdot 10^{15}$a | 3.6 | check of Nd-143(n,$\gamma$) |
| | Nd-145 | $\infty$ | 42 | |
| | Nd-146 | $\infty$ | 1.3 | check of Nd-145(n,$\gamma$) |
| | Nd-148 | $\infty$ | 2.48 | burnup indicator |
| | Nd-150 | $\infty$ | 1.2 | burnup indicator |
| | Pm-147 | 2.62a | 85 and 96 | decays to Sm-147 |
| | Sm-147 | $1.06 \cdot 10^{11}$a | 64 | |
| | Sm-149 | $\infty$ | 41000 | |
| | Sm-150 | $\infty$ | 102 | |
| | Sm-151 | 93a | 15000 | |
| | Sm-152 | $\infty$ | 206 | |
| | Eu-153 | $\infty$ | 390 | |
| | Eu-155 | 4.96a | 4040 | decays to Gd-155 |
| | Gd-155 | $\infty$ | 61000 | |

1) Half-life; $T_{1/2} = \infty$ denotes a stable nuclide.

2) Thermal cross section for neutron capture. If two values are given, the first refers to the formation of the product nucleus in the metastable, the second to the formation in the ground state.

TABLE II. STORAGE OF SPENT KONVOI $UO_2$ FUEL IN STORAGE RACKS OF THE REGION II TYPE: REACTIVITY WORTH $\Delta\rho = (1/k)\cdot(\Delta k/k)$ OF FISSION PRODUCTS AT THE TIME OF REACTOR SHUT DOWN

| **Isotope** | **Reactivity worth / pcm** | | | | | |
|---|---|---|---|---|---|---|
| *In. Enr.*[1] *burnup*[2] | 2.0 wt.-%, 9.9 MW·d/kg | 2.5 wt.-%, 17.4 MW·d/kg | 3.0 wt.-%, 23.9 MW·d/kg | 3.5 wt.-%, 30.4 MW·d/kg | 4.0 wt.-%, 36.7 MW·d/kg | 4.5 wt.-%, 42.6 MW·d/kg |
| Mo-95 | -76 ± 56 | -61 ± 55 | -156 ± 56 | -170 ± 50 | -235 ± 56 | -220 ± 55 |
| Tc-99 | -194 ± 54 | -213 ± 55 | -404 ± 56 | -370 ± 57 | -520 ± 56 | -529 ± 55 |
| Rh-103 | -375 ± 55 | -602 ± 55 | -766 ± 56 | -953 ± 56 | -1111 ± 56 | -1156 ± 54 |
| Xe-131 | -330 ± 54 | -372 ± 56 | -498 ± 56 | -503 ± 57 | -701 ± 56 | -788 ± 55 |
| Xe-133 | -30 ± 56 | 54 ± 56 | -6 ± 56 | -23 ± 57 | | 14 ± 55 |
| Cs-133 | -232 ± 55 | -236 ± 56 | -436 ± 56 | -518 ± 57 | -573 ± 56 | -654 ± 55 |
| Cs-134 | -7 ± 55 | -17 ± 55 | -48 ± 56 | 6 ± 56 | -170 ± 56 | -116 ± 56 |
| Cs-135 | -8 ± 55 | 42 ± 55 | -64 ± 56 | -6 ± 56 | -172 ± 56 | -100 ± 55 |
| Ce-144 | -23 ± 55 | | -22 ± 56 | 5 ± 56 | -54 ± 56 | -34 ± 56 |
| Nd-143 | -741 ± 56 | -1046 ± 55 | -1374 ± 56 | -1612 ± 56 | -1916 ± 56 | -1881 ± 55 |
| Nd-145 | -97 ± 56 | -58 ± 55 | -141 ± 56 | -337 ± 56 | -384 ± 56 | -316 ± 55 |
| Nd-146 | -35 ± 56 | -41 ± 55 | -3 ± 56 | -19 ± 56 | | 7 ± 55 |
| Nd-148 | -2 ± 55 | | | 35 ± 56 | -31 ± 56 | -19 ± 55 |
| Nd-150 | 13 ± 56 | | -18 ± 56 | 24 ± 57 | -19 ± 57 | |
| Pm-147 | -347 ± 55 | -416 ± 56 | -442 ± 56 | -419 ± 56 | -501 ± 55 | -481 ± 55 |
| Pm-148 | -174 ± 55 | -192 ± 55 | -189 ± 56 | -232 ± 57 | -304 ± 56 | -280 ± 55 |
| Pm-149 | 32 ± 56 | | -33 ± 56 | | -14 ± 56 | |
| Sm-149 | -564 ± 54 | -620 ± 55 | -753 ± 56 | -740 ± 57 | -897 ± 55 | -807 ± 55 |
| Sm-150 | -56 ± 55 | -114 ± 56 | -192 ± 56 | -181 ± 58 | -330 ± 56 | -113 ± 55 |
| Sm-151 | -546 ± 55 | -607 ± 56 | -755 ± 56 | -877 ± 57 | -1016 ± 56 | -1004 ± 54 |
| Sm-152 | -102 ± 56 | -200 ± 56 | -294 ± 55 | -407 ± 56 | -558 ± 56 | -494 ± 55 |
| Sm-154 | -30 ± 56 | 41 ± 55 | 40 ± 56 | -58 ± 57 | | 35 ± 55 |
| Eu-153 | 21 ± 55 | -3 ± 56 | -84 ± 57 | -72 ± 56 | -126 ± 56 | -98 ± 55 |
| Eu-154 | -8 ± 55 | -24 ± 56 | -82 ± 56 | -43 ± 58 | -197 ± 56 | -122 ± 56 |
| Eu-155 | -79 ± 56 | -48 ± 56 | -190 ± 56 | -54 ± 57 | -172 ± 56 | -174 ± 56 |
| Gd-155 | 32 ± 55 | | 42 ± 56 | 41 ± 57 | -21 ± 57 | -7 ± 55 |
| Gd-156 | -8 ± 56 | -34 ± 55 | -17 ± 56 | | | |
| Gd-157 | -15 ± 56 | | -13 ± 56 | | -56 ± 57 | -106 ± 55 |

1) Initial enrichment

2) Burnup assumed to be uniformly distributed within the fuel assemblies

### *2.2.1. Evaluation of Experimental Results from France*

A large experimental programme based on spent fuel chemical analysis has been performed in France since 1993 [10]. Results from this programme are shown in Figure 5, in the form of differences between measurement and calculation. The calculations were performed with the aid of the CESAR depletion code [11]. The cross-sections used were obtained from the APOLLO1 code and its CEA-86 multigroup library [12].

From the results shown in Figure 5 a bias in the neutron multiplication factor of a given Region II storage configuration can be derived as exemplified in Figure 6 by the case of the Region II KONVOI fuel storage racks already mentioned in section 2.1.2. The bias is given by:

$$\Delta k = k(E) - k(C) \tag{2.1}$$

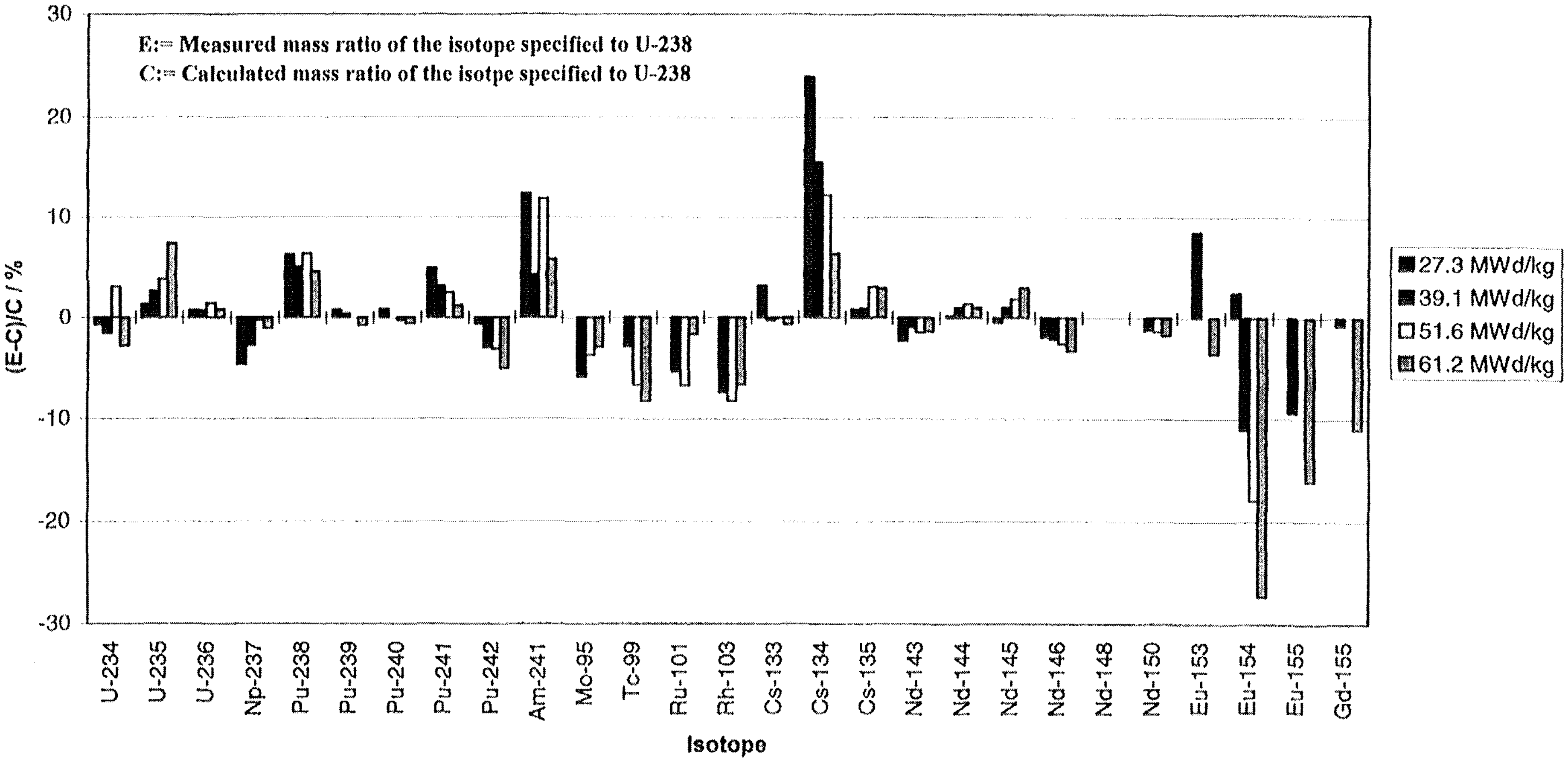


*FIG. 5. Results from the French experimental programme on high burnup [10]: Differences between measured (E) and calculated (C) isotopic mass ratios*

where *k(C)* refers to the calculated isotope number densities and *k(E)* refers to the corrected isotope number densities obtained with the aid of the differences shown in Figure 5. The neutron multiplication factors *k(E)* and *k(C)* were obtained by means of the SCALE-4.3 system using the 44-group cross section library 44GROUPNDF5 [13]. The actinide isotope Am-241 was not used in these SCALE calculations.

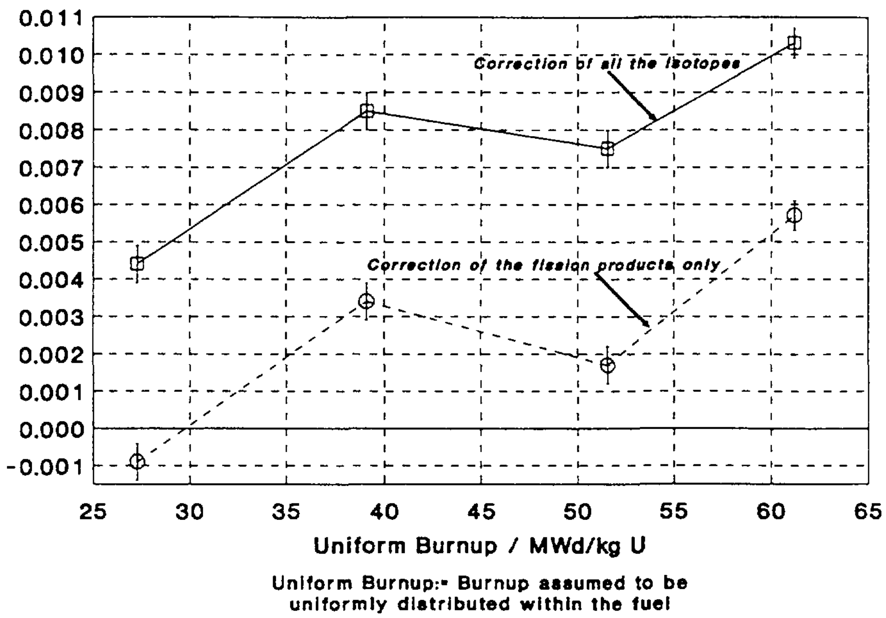


*FIG. 6. KONVOI fuel storage region II: Biases (2.1) in the neutron multiplication factor of the storage region due to the experimental outcomes shown in Figure 5*

As can be seen from Figure 6, the highest value obtained for the bias (2.1) is about $\Delta k = 0.01$ with a standard deviation of $\sigma(\Delta k) = 0.0004$. If only the differences between measurement and calculation which were obtained for the fission products are taken into account the highest bias being about $\Delta k = 0.006$ with a standard deviation of $\sigma(\Delta k) = 0.0004$ is obtained for the highest burnup investigated (61.2 MW·d/kg).

As can be seen from Figure 5, the APOLLO1(CEA86) - CESAR V.4 depletion calculations result in an underestimation of U-235 which increases with increasing burnup. However, the determination of the isotope number densities on which the estimation of the neutron multiplications *k(E)* and *k(C)* was based was not performed with the aid of the APOLLO1(CEA86) - CESAR V.4 code system but with the Siemens standard design procedure SAV90 [14-15]. This procedure is used by Siemens KWU for performing the depletion analysis for PWR fuel burnup credit applications [16]. As follows from comparisons of measured to calculated isotope concentrations in uranium rods as well as in reprocessed fuel assemblies the Siemens standard design procedure SAV does not underestimate the fissile actinides U-235, Pu-239, and Pu-241. As shown in Figures 7 through 9 the measured and calculated isotopic ratios are in agreement within the experimental error bounds. As can be seen from Figures 7 and 9 a slight underestimation of the non-fissile actinides U-236 and Pu-242 might be typical of the SAV procedure. (Figures 10 and 11 show the results obtained for the burnup indicator Nd-148. As can be seen, the measured and calculated ratios are in agreement within the experimental error bounds.)

Due to the fact that the SAV procedure does not underestimate the major fissile actinides and does not overestimate the major non-fissile actinides it is reasonable to take the bias

$$\Delta k \pm \sigma(\Delta k) = 0.0057 \pm 0.0004\,, \qquad (2.2)$$

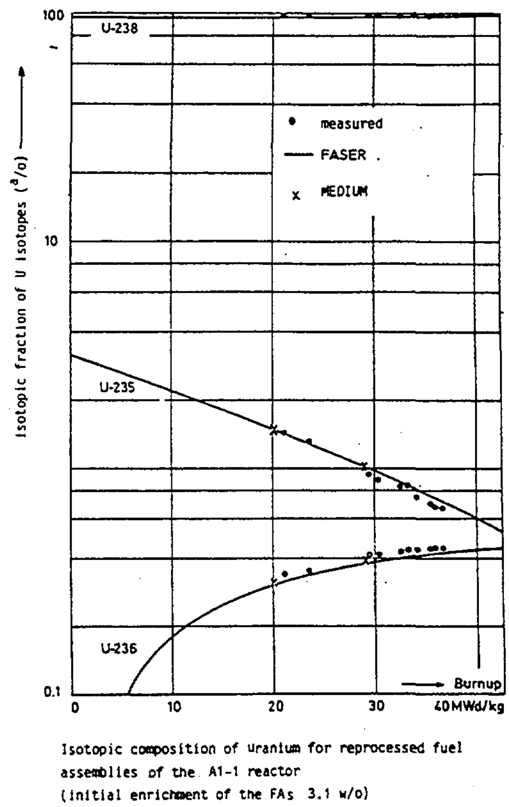


*FIG. 7. Verification of the Siemens standard design procedure SAV: Comparison of measured an calculated isotopic fractions*

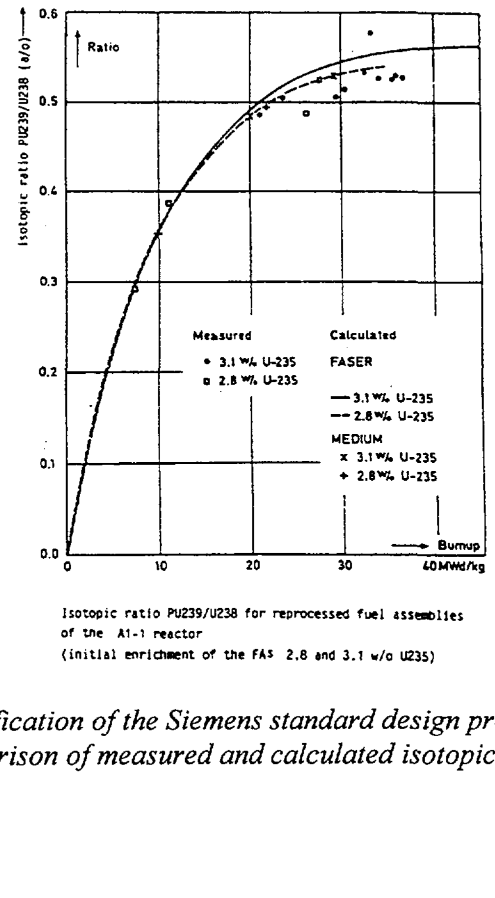


*FIG. 8. Verification of the Siemens standard design procedure SAV: Comparison of measured and calculated isotopic ratios*

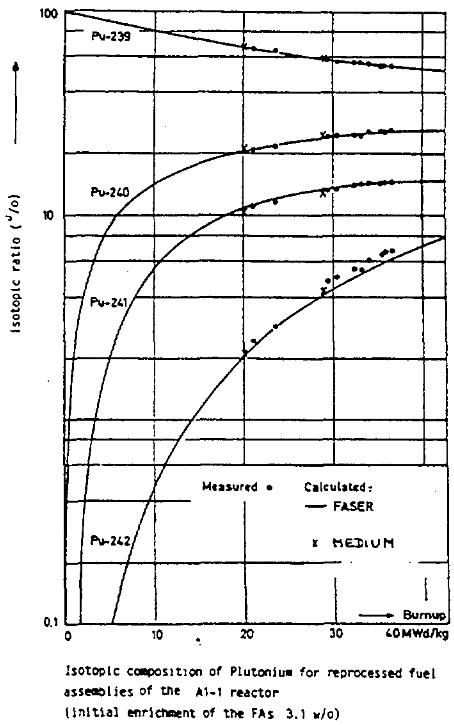


*FIG. 9. Verification of the Siemens standard design procedure SAV: Comparison of measured and calculated isotopic fractions*

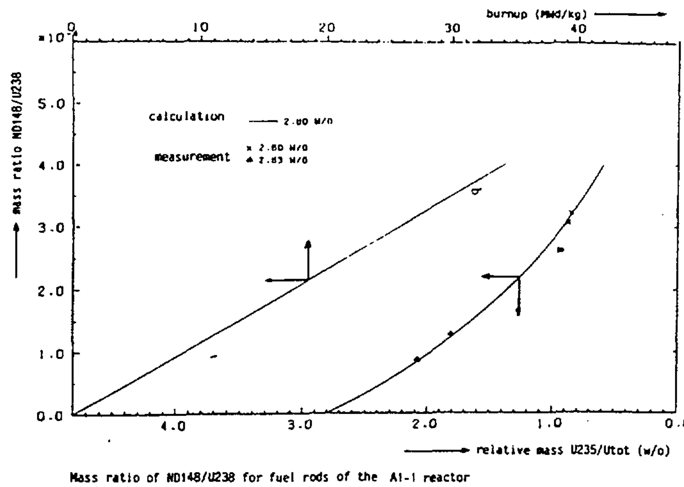


*FIG. 10. Verification of the Siemens standard design procedure SAV: Mass ratio Nd-148/U-238 as a function of the ratio U-235/U (curve a) (for the transformation of the Nd-148 Scale into the more convenient burnup scale the mass ratio Nd-148/U-238 is calculated as a function of burnup as shown by the curve b)*

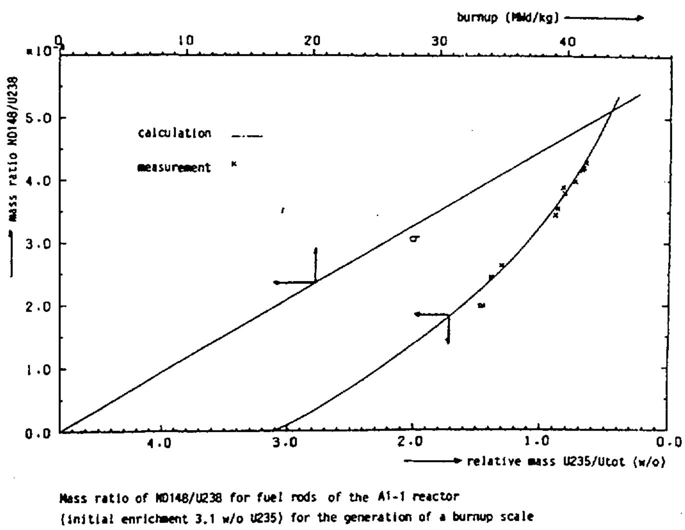


*FIG. 11. Verification of the Siemens standard design procedure SAV: mass ratio Nd-148/U-238 as a Function of the Ratio U-235/U (curve a) (for the transformation of the Nd-148 Scale into the more convenient burnup scale the mass ratio Nd-148/U-238 is calculated as a function of burnup as shown by the curve b)*

which was obtained by correcting only the fission product number densities with the aid of the French results shown in Figure 5, as the bias in the neutron multiplication factor of Region II storage configurations which arises from the uncertainties in the SAV depletion calculations.

This is confirmed by evaluations of chemical assay data recently obtained from the ARIANE programme [17] as well as by evaluations of in-core reactor measurement data.

*2.2.2. Evaluation of chemical assay data from the ARIANE international programme*

Results presently available from the ARIANE international programme are shown in Figure 12, in the form of differences between measurement and calculation. As can be seen from this figure, if these results were used as correction factors for calculated isotopic densities the results obtained for the actinides would always lead to a decrease of the neutron multiplication factor of a spent fuel storage pool. It is reasonable therefore to apply only the results obtained for the fission products to calculated isotopic densities.

As can be seen from a comparison of Figure 12 to Figure 4 and Table II, the smaller the reactivity worth of a fission product is the higher is the amount of the difference between measured and calculated concentration of the fission product, and vice versa. The impact of the big differences shown in Figure 12 on the neutron multiplication of a Region II storage configuration remains small, therefore. The bias $\Delta k = k(E) - k(C)$ (cp. eq.(2.1) in section 2.2.1) obtained with the aid of the Region II KONVOI fuel storage racks already used before amounts to:

$$\Delta k \pm \sigma(\Delta k) = 0.0052 \pm 0.0005. \quad (2.3)$$

Even if one should take note of the fact that the chemical assay data on which Fig. 12 is based are still under discussion, it is interesting to see that the result (2.3) is compatible with the result (2.2).

**E:= Measured mass ratio of the isotope specified to U-238,**
**C:= Calculated mass ratio of the isotope specified to U-238**

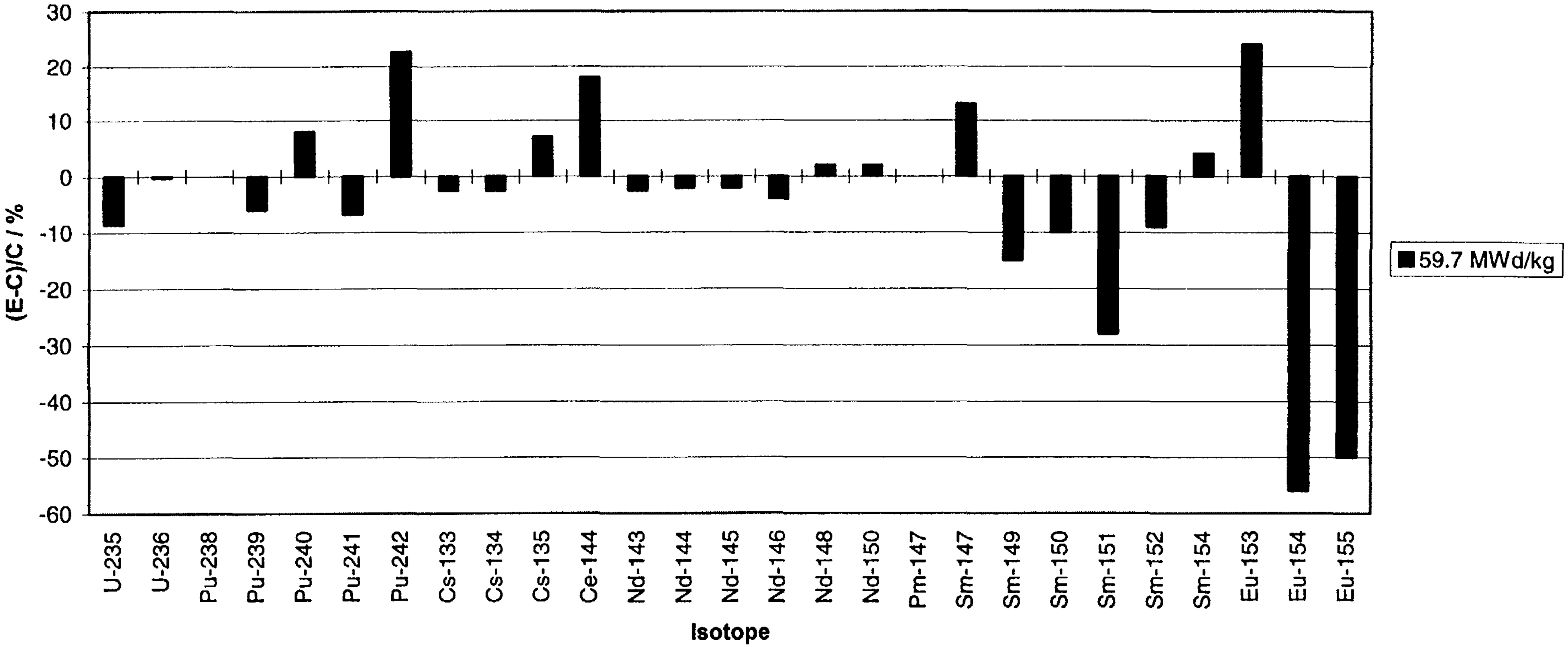


*FIG. 12. Results from the ARIANE programme [17]:*
*Differences between measured (E) and calculated (C) isotopic mass ratios*

*2.2.3. Evaluation of in-core measurement data: Comparison of calculated to measured critical boron concentrations*

It is obvious that significant errors in calculated isotopic number densities would lead to significant differences between calculated and measured critical boron concentrations. Figure 13 shows a statistics on the differences between calculated and measured critical boron concentrations. The one-sided lower 95%/95% tolerance limit:

$$L\ (95\%/95\%;\ \text{Fig. } 13) = -43\ \text{ppm} \tag{2.4}$$

referring to this statistics provides an appropriate estimate for the underestimation of the critical boron concentration. However the histogram Figure 13 has a significant skewness. This skewness results from the differences between boron concentrations obtained for non-Siemens reactors. Whilst the standard deviations of the histograms obtained for Siemens and non-Siemens reactors are in satisfactory agreement, the means of these histograms are different. The mean referring to Siemens reactors is compatible with zero [15], whereas the mean obtained for non-Siemens reactors is about 24 ppm. The reason for this bias in the means of the differences between calculated and measured critical boron concentrations is unknown. This bias might be put down to differences in the measurement devices, or procedures, or to different interpretation of the input data. With respect to neutron physics the characteristics of Siemens and non-Siemens reactors are not so different that a theoretical model improvement could reduce the bias for both reactor types simultaneously.

Nevertheless, if the histograms obtained for non-Siemens reactors are evaluated separately, one gets a lower 95%/95% tolerance limit of:

$$L\ (95\%/95\%;\ \text{non-Siemens}) = -70\ \text{ppm} \tag{2.5}$$

for the underestimation of the critical boron concentration.

1 ppm boron corresponds to $10^{-4}$ in the neutron multiplication. The result (2.5) corresponds therefore to a bias of:

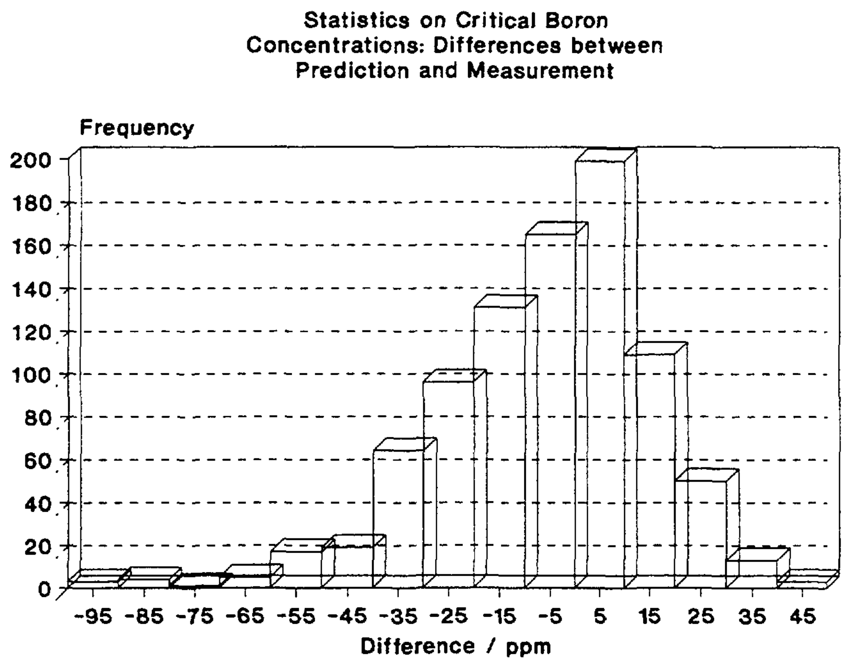


*FIG. 13. Verification of the Siemens standard design procedure SAV: Comparison of calculated to measured critical boron concentrations*

$$\Delta k\ (95\%/95\%) = 0.007 \tag{2.6}$$

in the neutron multiplication, expressed at the 95%/95% tolerance limit.

In Figure 14 the bias (2.6) is expressed as increment $\Delta B$ to the loading curve of the Region II KONVOI fuel storage racks already used before. The bias (2.3) obtained from the ARIANE data is also represented in this figure.

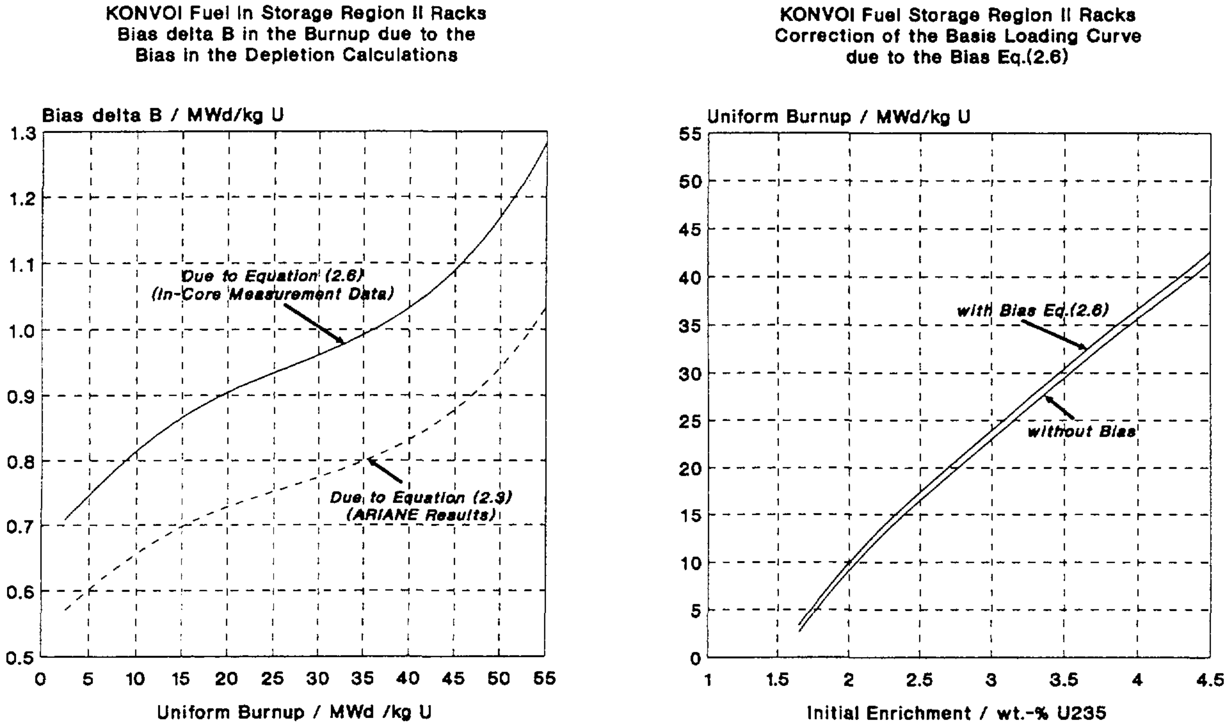


*FIG. 14. Bias in the loading curve due to the bias in the depletion calculations. (The loading curve shown still doesn't include the reactivity effect of axial burnup shapes, cp. Section 4.)*

## 3. CRITICALITY CALCULATION CODE VALIDATION

The biases $\Delta k$ shown in Figure 6 and given in equations (2.2) and (2.3) were determined with the aid of the criticality portion of the SCALE-4.3 system [13] due to the fact that Siemens KWU performs criticality analysis of fuel assembly storage facilities by using this system.

### 3.1. Validation of the SCALE-4 system with the aid of critical experiments and configurations

The validity of the criticality portion of the SCALE-4 system was verified many times with the aid of various kinds of critical experiments and configurations:

- Verifications of fresh fuel and net fissile content burnup credit calculations [7]: Evaluation of critical experiments covering a broad range of systems and fissile material types including homogeneous high- and low-enriched U-235 systems, heterogeneous low-enriched U-235 systems, U-233 systems, and Pu systems: See [18-20];
- Verifications of actinide only burnup credit applications [7]: Evaluation of critical experiments on mixed uranium-plutonium systems: See [18-22];
- Verifications of integral burnable absorber burnup credit applications [7]: Evaluation of critical experiments on integral burnable poisons: See [20, 22];

- Verifications of actinide plus fission product burnup credit applications: Evaluation of reactor critical configurations: See [22-23];
- Verifications of the temperature dependence of the neutron multiplication: See [24];
- Verifications of LWR spent fuel assembly storage pool analysis (with and without soluble boron in the pool water): See [20-21].

### 3.2. Comparison of SCALE-4.3 to MCNP4B

In order to evaluate the impact of different cross section processings on the calculated neutron multiplication factor the criticality portion of the SCALE-4.3 system was compared to the particle transport code MCNP [25] using the burnup credit benchmark problems specified in [26]. The SCALE-4.3 system was used with its ENDF/B-V derived 44-group library 44GROUPNDF5 [13], and the MCNP code was used with continuous-energy neutron cross-section data available from several libraries (cp. [25]). Results obtained are shown in Figures 15 through 17. As can be seen from these figures, the SCALE-4.3 and the MCNP results are in good agreement.

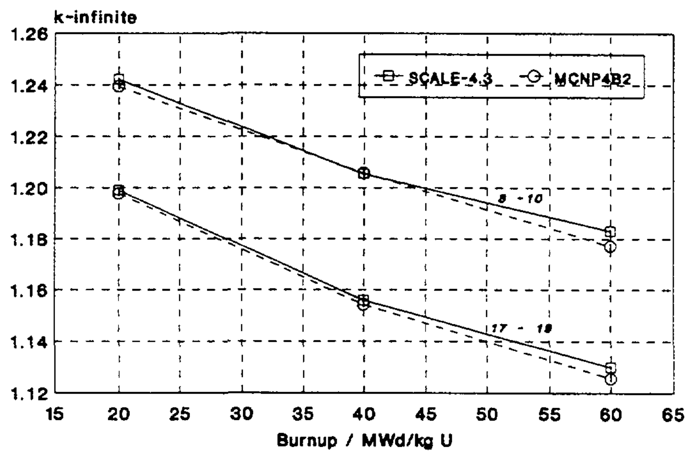


*FIG. 15. Comparison of SCALE-4.3 to MCNP4B2 for benchmark problems specified in [26]*

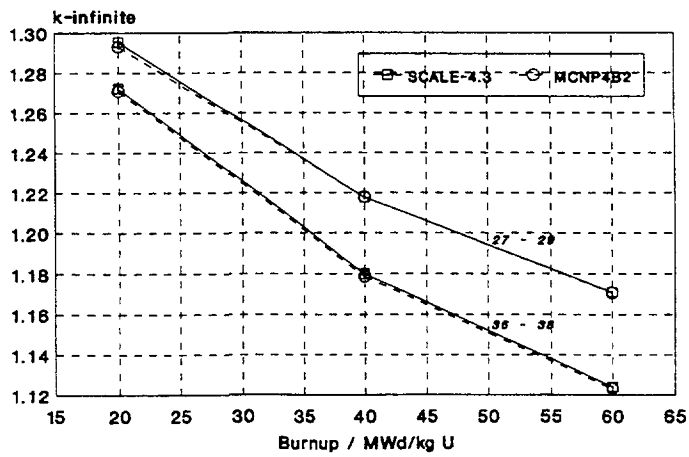


*FIG. 16. Comparison of SCALE-4.3 to MCNP4B2 for benchmark problems specified in [26]*

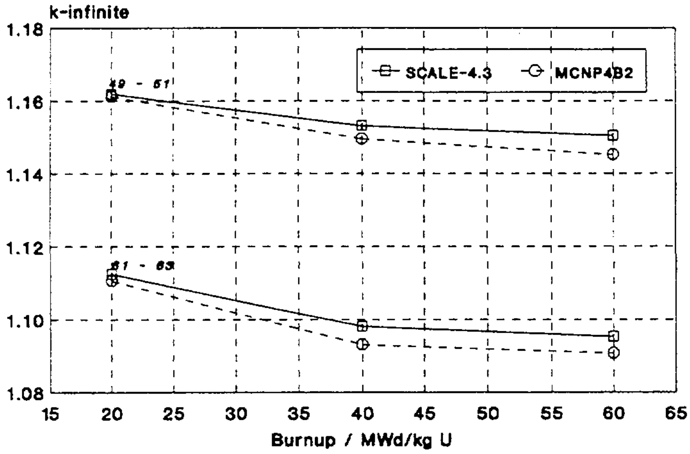


*FIG. 17. Comparison of SCALE-4.3 to MCNP4B2 for benchmark problems specified in [26]*

## 4. REACTIVITY EFFECT OF AXIAL BURNUP PROFILES

The reactivity effect of axial burnup profiles was studied in detail in [16] and [27-28].

The loading curve shown in Figure 14 still doesn't envelop the reactivity effect of axial burnup shapes. The Siemens KWU method used in the recent years for modeling PWR axial burnup shapes is illustrated in Figure 18. The real distribution is modeled by a step distribution. The number of steps is a free parameter because neighboring steps with differences smaller than a given threshold are combined to larger steps.

Results obtained with the aid of this step distribution model for the KONVOI wet storage case are shown in Figure 19. Each of the small bars shown in that figure represent an analyzed axial shape based on measured data delivered from the nuclear power plants under examination.

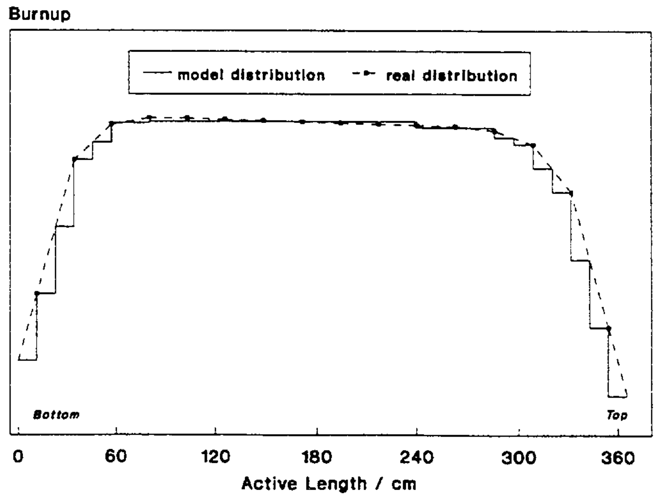


*FIG. 18. Siemens' method of modeling axial burnup shapes*

"Uniform burn-up" means constant burnup over the full active length of the fuel assemblies. The "equivalent uniform burnup" is the uniform burnup which has the same neutron multiplication as the analyzed axial shape characterized in Figure 19 by its average discharge burnup. If the equivalent uniform burnup obtained for an axial shape is less than the average discharge burnup referring to this shape, then this shape has a neutron multiplication higher than that one which would be obtained with a uniform distribution of the average discharge burnup. So, in the case shown in Figure 19 the effect of axial burnup shapes on the burnup credit is significant.

The difference $\Delta B$ between average discharge burnup and equivalent uniform burnup and the final loading curve obtained by adding $\Delta B$ to the basis curve Figure 14 are shown in Figure 20.

In this figure also a curve is given for $\Delta B$ that would be obtained if the axial burnup shapes were modeled in the common way (namely by putting the measured nodal burnups in the middle of the steps of the distribution model). As can be seen, the conservativeness maintained by using the method illustrated in Figure 18 is considerable.

## 5. ESTIMATION OF THE LOADING CURVE

The method of estimating a basis loading curve like that one shown in Figure 14 is described in detail in [28].

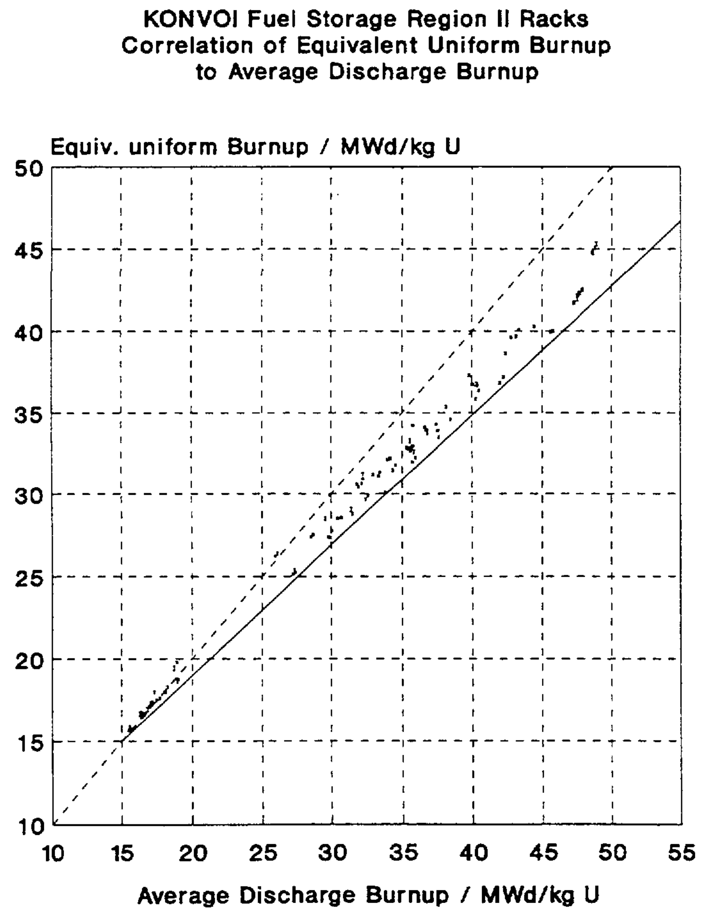


*FIG. 19. Effect of axial burnup shapes on burnup credit*

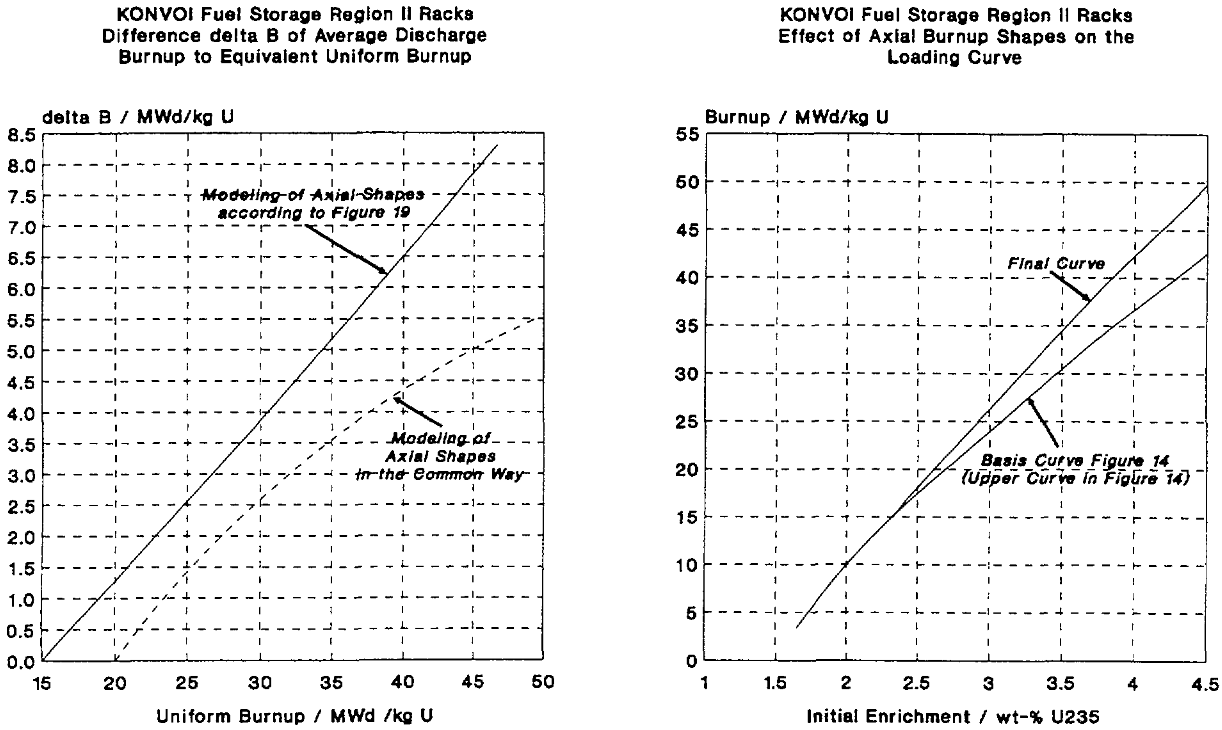


*FIG. 20. Effect of axial burnup profiles on the loading curve of KONVOI fuel storage racks of the Region II type*

## 6. VERIFICATION OF THE LOADING PROCEDURE USED FOR THE REGION II STORAGE RACKS

The result of applying a burnup credit criticality analysis methodology to the Region II storage racks is the determination of a loading curve like that one shown in Figure 1 which is the decision criterion on admitting the spent fuel assemblies to storage in the Region II storage racks. This decision criterion requires:

- the determination of the burnup of each fuel assembly intended for storage in a Region II storage rack; and
- the establishment of a control procedure that ensures:
  - that the loading curve is met; and
  - that individual fuel assemblies which do not comply with the loading criterion are segregated.

In compliance with the U.S. NRC draft Regulatory Guide 1.13 [29], for on-site storage fuel assembly burnup determination based on reactor records is accepted without any additional requirements, cp. [4, 7].

In accordance with draft Regulatory Guide 1.13 and the explanations given in [4], in order to minimize the probability that a loading error will be made written procedures are employed for the selection of fuel assemblies to be stored in the Region II racks including:

(1) initial placement of the discharged fuel assemblies in Region I storage racks;
(2) calculation by a utility staff member of the discharge burnup of each candidate for storage in a Region II storage rack;
(3) checking of the calculation by a second staff member who was not involved in the initial calculation;
(4) retention of the records for as long as the fuel assembly remains in the storage pool.

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IAEA-SM-352/35

# RADIOMETRIC CHARACTERISATION SUPPORTS, BURNUP CREDIT, SAFEGUARDS AND RADIONUCLIDE INVENTORY DETERMINATION FOR SPENT FUEL TRANSPORT, STORAGE AND DISPOSAL

XA9951803

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**Abstract**

Spent nuclear fuel characterisation measurements play an essential role in a range of fuel handling activities. In particular, they are necessary to support the application of burnup credit to the transport of spent fuel, to detect diversion of safeguarded nuclear material and to determine the radionuclide inventory of materials destined for final disposal. To apply measurements to these activities the measurement procedures need to be approved by the relevant regulatory bodies. Often key to the measurement procedures is the method of instrument system calibration and what a priori data is acceptable to aid the measurement process. Discussion of these, pertinent to the three areas of application mentioned above, is presented with suggestions of alternative approaches where considered appropriate.

## 1. INTRODUCTION

To encourage the renewed interest in nuclear power as an energy provider which produces very little greenhouse gas emissions, it is important to head off criticisms frequently aimed at the nuclear fuel cycle. These are often concerned with (i) the enhanced risk of nuclear weapons proliferation resulting from the increased amounts of plutonium generated from burning low enriched uranium (LEU) fuels and (ii) the lack of a closed cycle in terms of a satisfactory disposal route for the radioactive waste products. As in other areas of the nuclear fuel cycle, the use of better radiometric instrumentation may offer some help by improving the integrity of safeguards operations and reducing the costs of waste disposal whilst providing a traceable path for materials through the cycle.

Specifically, fuel characterisation measurements can provide a vital supporting role in a range of fuel handling activities including; (i) the use of burnup credit in storage, transport and disposal operations, (ii) the in situ verification of burnup and fissile content particularly for mixed oxide (MOX) fuels for safeguards and, (iii) the determination of radionuclide inventories for direct disposal of spent fuel or wastes resulting from reprocessing.

Better instrumentation may result from improvements in the way technology is applied to practical measurements, through to achieving greater measurement sensitivity and accuracy by using improved detectors and data processing technology. Presented in this paper are illustrations of the practical application of measurements associated with burnup credit for the transport of spent PWR commercial fuel in the U.S., along with a discussion of how far measurement sophistication should be taken to support safeguards.

## 2. FUEL CHARACTERIZATION MEASUREMENT APPLICATIONS

### 2.1. Burnup credit

Taking account of the reduction in the neutron reactivity (multiplication) of spent fuel, which results from irradiation, is known as burnup credit. The reduced reactivity is caused by the net loss of fissile and fissionable nuclides together with the generation of fission product poisons. The nuclides of major criticality importance were identified in an international study on burnup credit [1]. These are the fissile and fissionable nuclides: $^{235}U$, $^{236}U$, and $^{238}U$, and $^{239}Pu$, $^{240}Pu$ and $^{241}Pu$. The major fission products were also listed, these are: $^{95}Mo$, $^{99}Tc$, $^{101}Ru$, $^{103}Rh$, $^{109}Ag$, $^{133}Cs$, $^{147}Sm$, $^{149}Sm$, $^{150}Sm$, $^{151}Sm$ and $^{152}Sm$.

Burnup credit offers the nuclear industry a means of increasing the packing density of spent nuclear fuel in storage racks as well as in transport and disposal casks. Alternatively, it can allow a reduction in the amount of expensive neutron absorbers required in the containers. The present, very conservative, method of using the unirradiated or fresh fuel reactivity for spent fuel in criticality cask design calculations, known as the "fresh fuel assumption", leads to unnecessarily over-engineered and expensive cask designs of limited packing density. In anticipation of licensing approval of a burnup credit methodology, cask vendors are considering designs based on the reduced reactivity offered by burnup credit.

In the United States the Nuclear Regulatory Commission (NRC) controls the issue of licenses for spent fuel casks in accordance with the requirements of Title 10 to the Code of Federal Regulations (CFR), Part 72 (Storage), Part 71 (Transportation), and Part 60 (Disposal). A programme to change the licensing policy to one in which burnup credit can be used is being pursued by the United States Department of Energy (USDOE) through their series of topical reports on PWR actinide only[1] burnup credit [2,3]. The reports propose a methodology for the application of burnup credit. This is encompassed in five major steps:

1. Validate a computer code system to calculate isotopic concentrations in the spent nuclear fuel created during burnup in the reactor core and subsequent decay.
2. Validate a computer code system to predict the subcritical multiplication factor, $k_{eff}$, of a spent nuclear fuel package.
3. Establish bounding conditions for the isotopic concentration from criticality calculations.
4. Use the validated codes and bounding conditions to generate storage, transportation, and disposal package loading criteria (burnup credit loading curves).
5. Verify that spent nuclear fuel assemblies meet the package loading criteria and confirm proper fuel assembly selection prior to loading.

The last of the steps introduces the need to confirm the reactivity of spent fuel; this will, almost certainly, be achieved via the measurement of burnup. Such verification measurements are aimed at enhancing the administrative control to ensure beyond any doubt that fuel loaded into a cask is fully compliant with the prescribed burnup credit loading curves. In addition, the measurements will assist in the confirmation of the identity of each assembly by verifying other fuel history parameters.

The burnup credit loading curves, described in the topical reports, provide a means of segregating fuel assemblies into "specified" assemblies, that meet the acceptance criteria for loading into a particular fuel storage rack or transport cask designed to take account of burnup credit; and "non-specified" assemblies that do not meet the criteria. The criteria are based on a combination of fuel burnup and wt.% $^{235}U$ initial enrichment. Figure 1 shows an example of a loading curve.

Built into the curves are biases to account for any uncertainties in the data that relate burnup to the reactivity of the spent fuel. However, before using the cask loading curve to determine the loading status, it is necessary to determine the assembly's minimum assured burnup. The minimum assured burnup being the "actual" burnup minus the uncertainty on this value.

Based on the increases in cask capacities, significant commercial and operational advantages are anticipated giving in the region of 25% to 40% reduction in handling costs [4]. The USDOE estimate spent fuel transport cost savings of 35% using a 4 PWR assembly truck cask with actinide only burnup credit rising to 40% for full (principal isotope) burnup credit [5]. For rail transport, in which the anticipated unit costs are considered to be lower than those for truck due to the possibility of using larger casks of 21, 24 or 32 PWR assembly capacity, cost savings of up to 26% are anticipated. Depending on the mix of transport modes, the overall cost savings for transport of fuel from utility to repository is estimated to be between $200M to $1b if full burnup credit is used. These

[1] Consideration of fission products is not included . Only the following actinides, and their effect on neutron reactivity are considered: $^{234}U$, $^{235}U$, $^{236}U$, $^{238}Pu$, $^{239}Pu$, $^{240}Pu$, $^{241}Pu$, $^{242}Pu$ and $^{241}Am$.

figures are based on transporting 126,000 PWR assemblies in a mixture of General Atomics GA-4 truck casks and 24 to 32 assembly capacity rail casks.

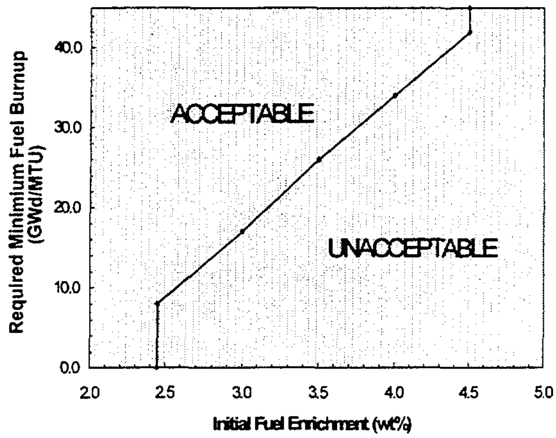


*FIG. 1 Typical fuel loading curve*

Although cost savings are expected by the use of burnup credit, various cost factors have to be considered to determine the total net savings. These factors include: (i) the reduced storage costs associated with the use of cheaper casks designs or the use of fewer casks as alluded to above, (ii) the potential added value of radiometric measurement data acquired prior to shipment. This could eliminate the need to re-open casks at the final repository for measurements that may be required to satisfy waste acceptance criteria, (iii) implementation costs of burnup credit in terms of license approval and administration, (iv) the cost of radiometric measurements and (v) the amount of burnup credit taken, i.e. actinide only or full burnup credit with fission product poisons included. The poisons reduce the multiplication, $k_{eff}$, by approximately a further 10%.

## 2.2. Application of burnup credit measurements

Verification of the "candidate" fuel assemblies, i.e. those expected to fall beneath the loading curves based on the reactor operator records, is anticipated to be made by physical measurement. The procedure detailed in the latest topical report [3] describes the use of a rejection criterion to judge whether the measured burnup of an individual fuel assembly is consistent with that declared in the reactor records. Rejection would result in the assembly being disqualified for loading into a burnup credit cask.

The specification of the rejection criterion can be used, however, as a good illustration of the difficulties involved with handling the uncertainties in two data sets when one is intended to verify the other. In this case, the two sets are the burnup values declared by the reactor operator and those derived from measurement.

In the topical, the proposed criterion is that "the measured burnup must be within 10% of the reactor record burnup". The measurement being intended to be used to confirm the reactor record value of burnup and the uncertainty in the reactor record to be accounted for by a related reduction in the burnup before comparison to the loading curve, i.e. to give the minimum assured burnup. Any disagreement between the measurement and the reactor record is not intended to be used to reduce the burnup credit but rather as an indication that something is wrong. The question then arises as to whether an unnoticed error of 10% would lead to an unsafe condition. The answer in the topical report is that approximately half of this difference is accounted for in the reduction of the assembly burnup due to uncertainty in the reactor records, i.e. 5%. However, if the assembly was at the low end of the reactor record uncertainty, the maximum error in burnup could be 10%, comprising 5% record and 5% measurement uncertainty. The DOE view this as acceptable (although not accepted by the NRC) because there is a significant change in the reactivity of the assemblies from fission products that are not accounted for in the derivation of the loading curves in the actinide only burnup credit proposal.

A calibration derived from the correlation between a measurable parameter, e.g. the activity of the fission product Cs-137 or the neutron emission rate principally from Cm-244, and the declared burnup for a representative set of assemblies is known as a "dependent" calibration. The use of such a calibration is viewed as appropriate for the application of burnup credit on commercial fuels because of the general acceptance that for a group of assemblies representing a reactor core there is very little, if any, systematic bias in the declared burnup. The proposed acceptance criterion to be used to qualify the dependent calibration and determine each assembly status is as follows:

(i) A calibration curve of the following form is to be derived and used to correlate the measured parameter (or count rate) to the reactor record burnup:

$$y_{counts} = a + bx_{reac}$$

where a and b are constants, $y_{counts}$ is the count rate of the measurable parameter, and $x_{reac}$ is the reactor record burnup value (when using neutron emission as a burnup indicator a non-linear expression may need to be substituted);

(ii) The validity of the calibration is then tested over the entire range of x by applying a 10% limit to the count rate Prediction Band Width [6], i.e.:

Prediction Band Width (converted to units of burnup) / Assembly Burnup < 0.1, where the

$$\text{Prediction Band Width (count rate)} = \left\{ \sqrt{(n+1)/n + \frac{(x-\bar{x})^2}{S_{xx}}} \cdot \sqrt{\frac{SS_R}{(n-2)}} \cdot t_{\frac{\alpha}{2},n-2} \right\}$$

$$S_{xx} = \sum_{i=1}^{n} (x_i - \bar{x})^2$$

$$SS_R = \sum_{i=1}^{n} (y_i - ax_i - b)^2$$

$n$ is the number of assemblies in the calibration set; and

$t_{\alpha/2,n-2}$ is the t-statistic at the $100(1-\alpha)\%$ confidence level for $n$-2 degrees of freedom ($\alpha = 0.05$ for 95% confidence);

The test therefore defines a range of x for which the inequality holds and the calibration is valid (procedures for dealing with the ranges of x that do not satisfy the inequality are suggested in the Topical; splitting the calibration range into smaller groups each with their own calibration is suggested);

(iii) For an assembly to be accepted for loading, the difference between the measured burnup derived via the measurement with the validated calibration and the declared burnup must be less than 10% of the declared burnup.

The procedure for taking a measurement and verifying the reactor records therefore appears to be well conceived and workable with the assumption that the reactor records data has for each assembly approximately a 5% uncertainty in burnup and that the measurement also contributes a 5% uncertainty. The difficulty, however, occurs when the uncertainty values, taken to be at a 95% confidence, do not meet the arbitrary value of 5%. In particular the reactor records, though accepted to be of a good accuracy with an average uncertainty at 1 sigma of 2% across a reactor core, for individual assemblies may be somewhat greater than 5%. The net result of this is that either the calibration may not meet the test criterion or some of the individual assemblies may fail the 10% test. If either of these occur, the identified assemblies may be disqualified from being loaded into a burnup credit cask, even though, as may be the case, the burnup is well below that prescribed by the applicable loading curve.

An alternative procedure, proposed by BNFL Instruments (BI) for discussion, recommends that measurements should play a greater role in the process of determining, with high confidence, the minimum assured burnup for each assembly. This procedure is as follows:

1. Calibrate the measured burnup indicator against the declared burnup records. In this, the linear expression is inverted to give;

$$y_{reac} = a + bx_{count}$$

in which $y_{reac}$ is the declared burnup and $x_{count}$ is the count rate of the measured burnup indicator;

2. The calibration set is recommended to be consistent, in number of assemblies, with a reactor core load of fuel comprising approximately 200 or more assemblies. This calibration should be carried out before commencing fuel loading;

3. Check the calibration data set for outlier assemblies. In this case an outlier assembly is defined as one for which the difference between the declared and the measured burnup is greater than a pre-defined percentage[2]. This is to eliminate assemblies that are clearly badly measured or incorrectly declared;

4. If any assemblies are identified as outliers these should be removed from the calibration data set. The assembly reference numbers of the rejected assemblies should be recorded pending an investigation that may include further measurement and other checking procedures. Failure to evaluate and rectify the cause of their outlier positions will make those assemblies ineligible for burnup credit;

5. If any assemblies are rejected during step 3 then a new reduced calibration data set will be used to recalibrate the burnup indicator;

6. Steps 2, 3 and 4 are repeated until there are no rejections identified at step 2;

7. Determine the assembly burnup, $y$ using the measured burnup indicator in conjunction with the established calibration curve for each of the assemblies that remain in the calibration data set and where appropriate other assemblies in the larger measurement campaign;

8. Determine the uncertainty on each of the measured burnup values by propagating the uncertainty in the calibration and the uncertainty in the individual measurement of the burnup indicator;

The uncertainty in $y$ based on the scatter in the calibration data set, is:

$$y = a.x + b \pm \left\{ \sqrt{\frac{1}{n} + \frac{(x-\bar{x})^2}{S_{xx}}} \cdot \sqrt{\frac{SS_R}{(n-2)}} \cdot t_{\alpha,n-2} \right\} \qquad (1)$$

where

$$S_{xx} = \sum_{i=1}^{n} (x_i - \bar{x})^2$$

$$SS_R = \sum_{i=1}^{n} (y_i - ax_i - b)^2$$

and $t_{\alpha,n-2}$ is the t-statistic at the 100(1-á)% confidence level for n-2 degrees of freedom.

[2] Identification of outliers can be based either on data points that fall outside a specified confidence interval, or, as in this case outside a specified percentage range. The choice of a fixed percentage is suggested to ensure that the probability of assembly rejection is lower for calibration data sets in which the amount of scatter is small. In this case it is possible that there are no rejected assemblies. If on the other hand a confidence limit, derived from the scatter in the calibration set, is chosen, there will always be a fixed proportion of the set rejected regardless of the quality of the data.

The overall uncertainty $\sigma_y$ (shown at 95% confidence level) is calculated at a stated confidence level based on the uncertainty in the measured count rate, $\sigma_x$ and the scatter in the calibration data:

$$\sigma_y = \sqrt{(a.1.65.\sigma_x)^2 + \left\{\sqrt{\frac{1}{n} + \frac{(x-\bar{x})^2}{S_{xx}}}.\sqrt{\frac{SS_R}{(n-2)}}.t_{\alpha,n-2}\right\}^2} \quad \text{.................................(2)}$$

9. Calculate the minimum assured burnup for each fuel assembly by decreasing the measured burnup by its total uncertainty to a specified confidence level. The confidence level, to be defined by the regulators, will ensure that the reduced burnup value gives a minimum assured burnup at the required level of confidence. (For example, to be 95% confident that the true value of burnup is greater than or equal to the minimum assured value one would have to use a t statistic with $\alpha = 0.05$ and a multiple of 1.65 for the uncertainty on the individual measurement. This uncertainty would be determined from a critical review of the measurement procedure with appropriate error propagation. The value of 1.65 assumes, however, that the uncertainties are well described by a Gaussian distribution). From equations (1) and (2) the minimum assured burnup, $M_{BU}$ at the specified confidence level is:

$$M_{BU} = y - \sigma_y$$

10. Compare the minimum assured burnup, as defined by the measured burnup and its associated uncertainty, with the cask loading curve for each assembly to establish its loading qualification.

BI considers that this methodology has several beneficial features, compared to methods that use the measurement purely as a verification of the declared burnup. Firstly, it is a very simple method that does not rely on any arbitrary assumptions about the scatter of the declared data set used during the production of the calibration. Secondly, it is capable of providing a determination of the uncertainty in burnup for each individual fuel assembly.

As with any dependent calibration this approach relies on the accepted position that the operator declared values for burnup have, when taken en-masse, negligible systematic error. This is commonly viewed as a key strength of the declared data, which enables an unbiased dependent calibration to be defined. The weakness in the reactor records is that the uncertainty in the burnup associated with individual assemblies is often undetermined. This weakness is overcome by the use of the declared data with the measured data as outlined by this alternative proposed methodology. The improvement stems from the use of a verifiable measure of the burnup and its associated uncertainty for each individual fuel assembly.

It should also be noted that this approach takes credit for both the quality of the declared burnup records and for the precision of the measurements. The better these are the greater will be the minimum assured burnup for each assembly. This, in turn, means that the number of assemblies that qualify for burnup credit loading may be increased.

In summary, it is suggested that the alternative measurement based approach offers a more realistic determination of minimum assured burnup for each assembly. Their values are likely to be higher, and hence of greater economic value, than those derived from a method that utilises an assumed operator declared uncertainty for each assembly. BI anticipate that this latter value would have to be fairly pessimistic to ensure that the worst uncertainties in the records are assumed for each assembly. This would result in lower minimum assured burnup values at the required level of confidence.

The BI methodology has been tested on a campaign of commercial PWR spent fuel. The campaign comprised 203 assemblies measured in the U.S. in 1997. In these the burnup was measured using the burnup indicator Cs-137 corrected for cooling time and axial burnup profile to give the assembly average burnup. Figure 2 shows the measurement data used to determine a calibration for the burnup indicator. Table I shows the derived minimum assured burnup along with the operator declared burnup for a set of 31 assemblies chosen randomly from the 203 assemblies in the campaign. The effectiveness of the approach is demonstrated by the relatively small amount that the minimum assured burnup is below the declared burnup. For the 31 assemblies in the table this is 4% ± 3%.

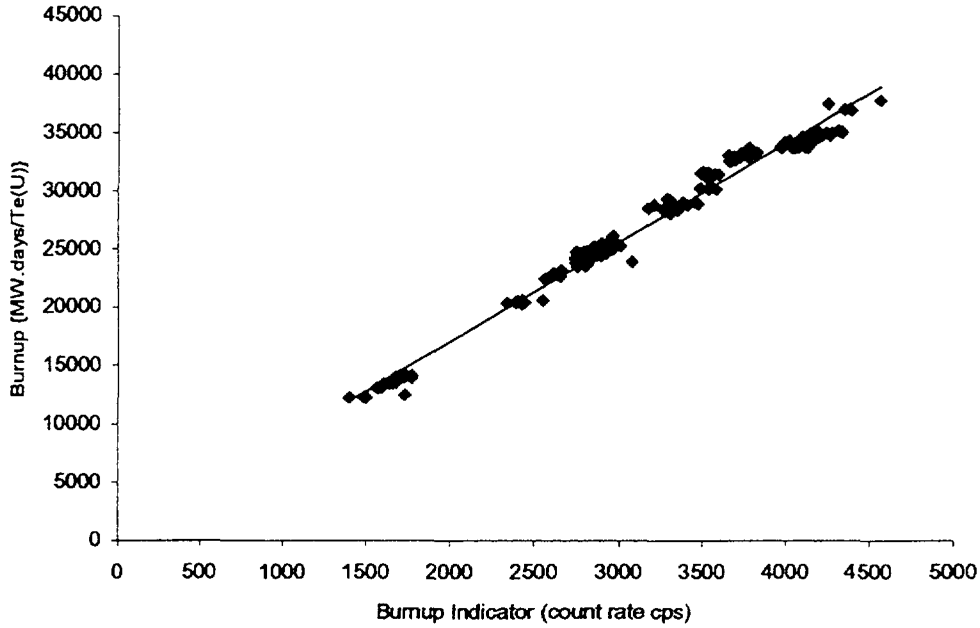


*FIG. 2. Calibration of Cs-137 662 keV gamma ray emission burnup indicator for a campaign of 203 assemblies*

## 2.3. Safeguarding of spent fuel

As the global quantity of spent nuclear fuel steadily grows the need for rigorous control of the large quantities of fissile nuclides, predominantly $^{235}U$, $^{239}Pu$, and $^{241}Pu$, within the fuel is becoming increasingly important from a safeguards standpoint. Plutonium content represents about 1% of spent fuel assembly mass. Globally the current stocks of more than 150,000 t HM in spent fuel assemblies contain more than 1,000 tonnes of plutonium. The amount accumulated through the lifetime of the currently operating reactors may rise by a factor of 2 or 3 depending on the quantities reprocessed and recycled in the form of mixed oxide (MOX) fuels.

Measurement and verification of such large quantities of plutonium and fissile uranium within spent fuel assemblies, beyond the level of simply item counting, are potential requirements. If so, rigorous measurement methodologies, similar to those proposed for burnup credit, will be equally relevant to safeguards for fissile material quantification or verification. Improved safeguards measurements may not only be of benefit to aid non-proliferation but could enhance the public acceptability of handling and transportation of fissile materials. For example, in the U.S. this may improve the prospects for conversion of DOE owned fissile material into commercial fuel for burning as MOX. Alternatively, improved faith in safeguards could encourage the earlier transfer of material from DOE sites and power utilities to a long term repository.

## 2.4. Measurements in support of safeguards

The application of measurements to safeguards, in contrast to burnup credit, is likely to require direct measurement of fissile content. For burnup credit, though fissile content is the real issue for the purpose of criticality calculations, the measurement of burnup in combination with a given initial U-235 enrichment is an approach that is generally accepted (though not yet approved in the U.S.) for commercial power generation spent fuel. Such an arrangement for safeguards would not guarantee diversion detection.

TABLE I. A RANDOM SELECTION OF 31 ASSEMBLIES SHOWING THE DERIVED MINIMUM ASSURED BURNUP COMPARED TO THE OPERATOR DECLARED BURNUP

| Assembly | Burnup Indicator | | Burnup {MW·d/t(U)} | | |
|---|---|---|---|---|---|
| Reference | Value | Uncertainty | Declared | Measured | Minimum Assured |
| 2 | 1,504 | 41 | 12,310 | 12,827 | 12,221 |
| 5 | 1,773 | 46 | 13,946 | 15,124 | 14,455 |
| 11 | 2,753 | 69 | 23,509 | 23,487 | 22,505 |
| 19 | 2,648 | 67 | 23,066 | 22,590 | 21,645 |
| 27 | 2,744 | 69 | 24,221 | 23,405 | 22,421 |
| 33 | 1,773 | 46 | 14,173 | 15,127 | 14,458 |
| 44 | 1,666 | 44 | 13,650 | 14,209 | 13,567 |
| 50 | 2,342 | 60 | 20,444 | 19,973 | 19,123 |
| 51 | 2,896 | 73 | 25,001 | 24,700 | 23,663 |
| 58 | 1,640 | 43 | 13,493 | 13,988 | 13,347 |
| 61 | 2,640 | 67 | 22,815 | 22,520 | 21,572 |
| 67 | 2,888 | 73 | 25,250 | 24,635 | 23,595 |
| 74 | 2,823 | 72 | 24,792 | 24,076 | 23,059 |
| 79 | 2,850 | 73 | 25,212 | 24,313 | 23,285 |
| 83 | 2,947 | 74 | 25,350 | 25,136 | 24,087 |
| 90 | 2,935 | 74 | 25,236 | 25,032 | 23,984 |
| 96 | 2,828 | 72 | 24,304 | 24,126 | 23,113 |
| 99 | 2,592 | 66 | 22,572 | 22,111 | 21,177 |
| 113 | 3,655 | 90 | 33,075 | 31,174 | 29,898 |
| 119 | 3,778 | 93 | 33,733 | 32,228 | 30,909 |
| 123 | 3,693 | 91 | 32,736 | 31,498 | 30,214 |
| 132 | 3,663 | 90 | 32,647 | 31,248 | 29,981 |
| 142 | 3,412 | 84 | 28,801 | 29,104 | 27,914 |
| 153 | 3,277 | 81 | 28,434 | 27,954 | 26,802 |
| 157 | 3,279 | 83 | 28,657 | 27,969 | 26,801 |
| 170 | 4,104 | 101 | 34,043 | 35,010 | 33,581 |
| 176 | 4,141 | 102 | 34,247 | 35,322 | 33,879 |
| 184 | 4,241 | 104 | 35,001 | 36,177 | 34,703 |
| 192 | 4,216 | 104 | 34,766 | 35,966 | 34,498 |
| 198 | 4,568 | 113 | 37,797 | 38,963 | 37,362 |
| 202 | 2,895 | 73 | 24,891 | 24,695 | 23,657 |

The above table contains a subset of data from a campaign of 203 assembly measurements. The minimum assured burnup is calculated in each case from the measured burnup indicator and its associated uncertainty and the uncertainty derived from the dependent calibration. The following are the calculated terms used in the uncertainty analysis:

The number on assemblies measured during this campaign, $n=203$
The average burnup indicator, $\iota=3073$
The calibration parameters, $a=8.53$, $b=0$
Summations f the residuals, $SS_R=134026293$, $S_{xx}=138643969$
The one tailed t-statistic for n-2 degrees of freedom at the 95% confidence interval, $t_{0.05,\,n-2}=1.65$

Safeguards monitoring therefore requires direct measurement of fissile content with an ability to discriminate between fissile uranium and plutonium. The candidate measurement approaches are therefore likely to comprise a combination of active and passive neutron techniques with gamma spectrometry. Elements of such systems are currently being developed in the nuclear countries.

As for burnup credit, the question arises, as to the most appropriate method of calibration of these instruments. Clearly a calibration independent of any operator declared information would be preferred. Though at first this appears quite straightforward, it is difficult to fully achieve because of the strong dependence of a fissile measurement on the fuel geometry as well as the fissile content. The standard approach would be to calibrate the measurements systems by modelling or simulating the given fuel and measurement geometry. During safeguards measurement it is therefore important that the geometry is as expected, otherwise the measurement could be invalidated or at least inaccurate. This is an important aspect of safeguard monitoring, as good knowledge of measurement uncertainty is key to gaining high confidence that a fraction of the declared amount of fissile material has not been diverted (partial defect).

The solution may be, therefore, to combine the radiometric techniques (active and passive neutron counting and gamma spectrometry) with a means of confirming the geometrical arrangement, using for example real time radiography (RTR). This approach could also offer the ability to correct a measurement for the effects of damaged fuel or research fuels for which detailed geometrical information may be lacking. Currently an instrument offering this combination of techniques is not yet available but could be developed in response to the demands of the safeguards regulators.

### 2.5. Spent fuel waste disposal

Under present policies a significant proportion of the world's commercial spent fuel is viewed as waste. Although the waste in the UK is not in general spent fuel, but industrial radionuclides and residues from reprocessing, there is a requirement to measure (or infer from measurement) some 78 radionuclides. Similar requirements for radionuclide content assessment therefore seem likely for spent fuel disposal. The measurement of burnup and associated irradiation history parameters such as cooling time could be used, as it is for waste under the U.K definition, to provide the required radionuclide inventory data for spent fuel.

### 2.6. Measurement techniques and methodologies

Available techniques include high resolution gamma spectrometry, passive neutron counting and active neutron counting. When used in conjunction with nuclide inventory computer codes, such as ORIGEN or FISPIN, the radiation measurements allow burnup, cooling time, initial wt.% $^{235}U$ enrichment, residual wt.% $^{235}U$ equivalent enrichment and radionuclide inventories to be determined for intact fuel assemblies and for dismantled assemblies or fuel debris.

The successful use of the characterization measurements depend, in a similar way to the other applications outlined above, on development of appropriate techniques together with the availability and acceptance of methodologies that cover the measurement process and the related calibration procedures. These are necessary to correlate the measurable radiation emissions with the required spent fuel parameters, such as burnup, and will be essential to the regulatory control of spent fuel contaminated waste destined for interim or final disposal.

A range of modular spent fuel monitoring systems for fuel characterization has been developed by BNFL Instruments. Historically, these were based on instrument systems produced for reprocessing facilities at Sellafield; their primarily role being related to process control, radionuclide inventory assay and safeguards applications. The systems use a variety of radiometric techniques along with different approaches to calibration and validation procedures necessary to ensure reliable and accurate operation that is appropriate to the customer requirements. In the case of the direct disposal of spent fuel, as is currently favored in the U.S., the calibration of measurement systems is likely to be more akin to the application of burnup credit where it may be appropriate to rely on some operator declared parameters. This is in contrast to safeguards measurement in which, little if any, operator declared data should be relied upon.

## 3. CONCLUSIONS

It is clear that radiometric measurements play a key role in a range of fuel handling activities and can provide benefits that may be financial or safety related. In relation to this there are essentially two questions posed in this paper. The first is, what is the best approach to confirm reactor operator declared burnup for the application of burnup credit, and the second is how far should safeguards measurements go in terms of the degrees of blindness under which they are made.

In answer to the first; a burnup credit measurement methodology that is an alternative to the USDOE approach is suggested. This alternative approach puts a greater emphasis on the measurement. It is the measured term minus its uncertainty that is used to derive the "minimum assured burnup". The minimum assured burnup for each spent fuel assembly is then compared to the appropriate loading curve for the selected transport cask. This approach differs from that proposed by the USDOE in which the measurement is intended only to route out badly declared assemblies. The minimum assured burnup being determined by deducting an assumed uncertainty from the declared burnup before application to the loading curves. This is criticised on the grounds that because of the variability of the reactor records burnup uncertainty for individual assemblies, it will be necessary to assume a pessimistic uncertainty value and thereby produce a smaller and less valuable minimum assured burnup.

With regard to the second question, should safeguards measurements for the purpose of identifying diversion operate totally blindly from operator (or state) input. In particular, for fuel assemblies should this blindness include the geometrical structure of the fuel? If this is the correct way forward, the measurement challenge will have to be met with more sophisticated measurement techniques and systems that will be able to measure the fissile content of fuel assemblies in absence of geometrical and other supporting information.

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# EXPERIENCES FROM THE OPERATION OF THE SWEDISH CENTRAL INTERIM STORAGE FACILITY FOR SPENT FUEL, CLAB

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**Abstract**

Today about 50% of the electric power in Sweden is generated by means of nuclear power. The Swedish nuclear programme comprises 12 plants. According to political decisions no more nuclear power plants will be built and the existing plants shall not be operated beyond the year 2010. The programme will give rise to not more than 7,800 metric tonnes (U) of spent fuel, which will be directly disposed of in the crystalline bedrock without reprocessing. A keystone in the spent fuel management strategy is the central interim storage facility, CLAB. After an intensive preproject work the licensing of CLAB according to the Building Act, Environment Protection Act and Atomic Energy Act took place in 1978-1979. After a total licensing time of about 20 months the last permit was obtained in August 1979. By June 1998, CLAB had received and unloaded some 1,000 fuel transport casks corresponding to about 2,752 tonnes U and 81 casks containing highly active core components. The performance of the plant has been very satisfactory and with increasing experiences it has been possible to reduce the operating and maintenance costs. The extensive efforts during the design phase have resulted in a collective dose of 25-30% of the dose calculated in the final safety report. Due to a low activity release from the fuel and an optimised management of the used water filtering agents the number of waste packages emanating from CLAB has been less than 10% of what was originally expected. The activity release to air and water from the facility during the five first years of operation has been around 0,01% of the permissible release. In order to postpone the building of additional storage pools, new storage canisters has been developed which has increased the storage capacity from 3,000 to 5,000 tonnes U.

## 1. BACKGROUND

The Swedish nuclear programme comprises 9 BWRs and 3 PWRs with a total electrical output of 10,300 MWe and an annual production of around 70 TW·h. This means that about 50% of the electricity produced are generated in nuclear plants, the remainder being mainly hydropower. According to a decision of Parliament, no more reactors are to be built in Sweden and the existing ones shall not be operated beyond the year 2010. This decision was taken after an intense debate on the nuclear issue during the 70s and an advisory referendum in 1980. In all, the twelve reactors are expected to have produced approximately 2,000 TW·h of electric power and 7,800 metric tonnes (U) of spent fuel by the year 2010. These facts form the current basis of planning for the Swedish radioactive waste management programme.

Sweden has chosen the direct disposal option for the management of spent fuel. This choice has been judged to be the most rational and cost-effective solution under the prevailing conditions in Sweden. According to this, the spent fuel from the power plants will be stored in an AFR (Away From Reactor) interim storage facility (CLAB) for some 40 years before final disposal. This will allow the decay heat to decrease considerably and make final disposal easier. It will also allow great flexibility to adjust to future developments in the area of spent fuel management. Further, it will allow ample time for the research and development necessary for the selection of the site for the final repository and for the design and optimisation of the back-end of the fuel cycle.

## 2. DESCRIPTION OF THE FACILITY

### 2.1. Site

CLAB is located on the Simpevarp peninsula near the Oskarshamn nuclear power plant with its three reactors O1, O2 and O3, owned by OKG Aktiebolag. The choice of site provided a number of co-ordination advantages, for example access to a common harbour, the interim storage facility for low- and intermediate-level waste, central workshop etc. OKG has been contracted for the operation and maintenance of the CLAB facility.

## 2.2. Main features

CLAB comprises one above ground and one underground section. The above ground complex consists of a number of interconnected buildings. Preparation and handling of incoming transport casks and spent fuel and core components take place in the receiving building. Directly connected with the receiving building are buildings for auxiliary systems, e g for water cooling and purification, waste handling, ventilation, compressed air etc. and for the electric power and control system.

The receiving building has three receiving pool lines, two of which are specially equipped for receiving the TN17 Mk2 cask. The third receiving pool can be used to receive casks other than TN17 Mk2, although additional equipment specific to the actual cask must be provided. This pool also accommodates fuel leakage detection equipment. Filled canisters can be stored temporarily in other pools in the receiving building before being transferred to storage.

The storage section is located underground in a rock cavern located 25-30 metres below the surface. The cavern is 120 metres long, 21 metres wide and 27 metres high. It contains four storage pools and one smaller central pool connected to a transport channel. Each storage pool contains about 3,000 $m^3$ of water and can hold 1,250 metric tonne (U) of spent nuclear fuel, giving a total storage capacity of 5,000 metric tonnes (U).

The spent fuel is transported from the receiving building to the storage section in a fuel elevator. The elevator shaft is connected with the pools via a channel. The storage section is also connected with the surface building through a shaft containing a passenger elevator, ventilation ducts, electricity and water supply.

The Swedish reactors generate about 220 metric tonnes (U) of spent fuel a year. With a fuel transport rate of 210-250 metric tonnes (U) a year, which is well below the designed capacity of the Swedish transport system and CLAB, the storage pools in CLAB will cover Swedish needs through 2004. An expansion will be required to accommodate all the spent fuel and core components emanating from the 12 Swedish reactors. Excavating one additional rock cavern parallel to the first one will perform this expansion.

## 2.3. Handling of transport casks and fuel

In 1980, the TN17 Mk2 cask holding 17 BWR or 7 PWR fuel assemblies was selected as the standard fuel transport cask in Sweden. One of the reasons for this was the fact that the cask is dry, which was considered safer than wet casks. The core components are transported in a version of the cask without cooling fins and neutron shielding.

Today, depending on the initial enrichment of the fuel, sometimes one position in the cask must be blocked because of criticality reasons. The transport vehicle with the transport cask is driven into the facility through a combined passageway and airlock beneath the floor of the receiving building. From there, the cask is lifted into the receiving building of the facility, where it is placed in one of the three preparation cells. There, the cask is provided with a metal skirt that protects the outside of the cask against contamination and damage. Hoses are connected to the skirt and the cask is cooled externally via its fins by circulating water through the annulus between the skirt and the cask. The water and the skirt itself reinforce the radiation shielding of the cask. Preparation of the cask involves, among others, connection of the cask's internal cavity to a cooling system by means of hoses.

Prior to internal cooling, the cask is vented to detect any radioactive gases inside. If such gases are detected this indicates that fuel damage has occurred during transport and a special treatment and handling procedure is applied. Normally, circulating water, which passes a filter where particles are collected, cools the fuel assemblies. When the temperature and activity of the water have reached acceptable values, cooling is terminated and the cask is prepared for lifting out of the cell and transfer to one of the receiving pool lines. There, the lid is removed so that the fuel assemblies can be lifted out one by one. Hydraulic handling machines perform unloading -an all subsequent handling of the fuel assemblies - underwater. The pools are arranged so that the cask is immersed in non-contaminated water during the unloading operations. The fuel assemblies are transferred to storage canisters.

The original storage canisters can hold 16 BWR or 5 PWR fuel assemblies and have an internal structure made of normal stainless steel. Recently the canister capacity has been increased to 25 BWR or 9 PWR fuel assemblies, thereby enlarging the total storage capacity of CLAB from 3,000 tonnes of uranium to 5,000 tonnes. The storage canisters are used for all internal transports between the different pools in the facility.

From the receiving building, the storage canisters are transported one at a time to the rock cavern by a fuel elevator with a water-filled cage. This handling is remote-controlled. On passing from the pools in the receiving section to the elevator shaft, the elevator cage goes through a water trap. The elevator shaft itself is not water-filled. The storage canister is thus transferred down to the storage section's transport channel, which is also water-filled. The canister is lifted out of the cage by a handling machine and taken to a predetermined position in one of the storage pools. The canister now also serves as a storage rack.

Prior to the return of the empty cask to the preparation cell and subsequent dispatch from the facility, the cask can be cleaned internally by means of underwater desludging equipment that works according to the "vacuum cleaner" principle.

## 3. SAFETY AND RADIOLOGICAL WORKING ENVIRONMENT

### 3.1. Criticality

The criticality safety is based on USNRC Reg. Guide 1.13 "Spent Storage Facilities". According to this the plant is designed in such a way that a $k_{eff}$-value of $\leq$ 0,95 is always maintained for fresh, unbent fuel. The guide allows a $k_{eff}$-value of $\leq$ 0,98 for very unlikely events. Due to the very great population of fuel assemblies in CLAB $k_{eff} \leq 0{,}95$ has been a target value even for accidental conditions.

### 3.2. Major mechanical impact

The rock cover of the storage section provides good protection of the stored fuel against any impact from the outside, such as acts of war, sabotage and extreme natural conditions. The site offers solid and seismically stable bedrock. Rock protection is not needed for the receiving section, as the fuel is stored there only for a short period of time and the amount is relatively small.

### 3.3. Loss of water in the storage pools

The storage pools have very thick concrete walls with extremely strong reinforcement. Loss of water from the pools can only occur by evaporation in case of total loss of electricity supply and cooling. The pool water will heat up to close to 100°C. in about one week. If no feed water is supplied, the water level will reach the top at the fuel after about one month. In order to make up for the loss due to evaporation, water can be supplied by gravity by means of a pipe installed from above ground level to the pools. There is ample time to arrange for external water supply e g by tank trucks if necessary. The steam generated is vented to the atmosphere by natural draught through the existing ventilation ducts. The emergency cooling is thus safeguarded by a passive system depending on gravity only.

### 3.4. Radiological working environment

In view of the fact that the CLAB facility will be in operation for around 60 years, great importance has been attached to the radiological working environment. The design philosophy is based on a maximum permitted average dose commitment equal to 1/10 of the ICRP limits. In order to achieve this, adequate biological shielding has been installed around different process systems.

The fuel transport cask cooling system, where the greatest accumulation of radioactivity could be expected, has been equipped with a comprehensive system permitting remote removal of components by means of shielded casks. The active components can then be handled in "hot cell" using master-slave manipulators. There the components are repaired, maintained or, if appropriate, conditioned for disposal as waste. Most of the process is remote-controlled with supervision and control system based on computers and graphic terminals.

In order to reduce the impact of possible airborne contamination in the receiving hall, the normal air change rate is as high as five times per hour in the floor zone where the operators work. If necessary, this air change rate can be extended to the total volume of the receiving hall by use of an extra ventilation system.

## 4. INVESTMENT, FINANCING AND TIME SCHEDULE

### 4.1. Investment and financing

In 1980, the total investment in current money was estimated to be 1,270 MSEK (including assumed index-tied adjustments, excluding interest) roughly corresponding to 170 MUS$. The true investment at the end of construction in 1985 was 1,720 MSEK (225 MUS$) signifying an increase of 36% in comparison with the original calculations. The main contribution to the increase emanated from the engineering cost, which became more than 90% higher than expected. The civil engineering and building costs were 54% higher than expected to some extent reflecting the rather extensive changes in process and handling systems rather late in the project. These increases can in many cases be attributed to the uncertainties and lack of experiences mentioned earlier.

According to Swedish law all costs for the management of the radioactive waste, including the decommissioning of the nuclear power plants, have to be borne by the owners of these plants. A fee determined annually by the government covers the costs. The money collected is funded in accounts in the National Bank. The funds can be used by SKB after approval of a governmental authority. This financing system is the reason why the interest was not included in the investment calculations above. The basis for the fee is a cost calculation of all the activities for the back-end of the nuclear fuel cycle, which is carried out by SKB each year. The cost calculation is based on a scenario for the back-end including R&D, construction, operation, replacement and dismantling of all necessary facilities and equipment.

### 4.2. Time schedule

The first blasting work for CLAB took place in May 1980. At that time, the storage section was scheduled to be completed in two stages and the first fuel to be received in July 1985. It was later decided that the two stages should be carried out simultaneously and that the first fuel should be received in January 1985. In actual fact the first fuel was received in July 1985.

## 5. NUCLEAR OPERATION

In June 1998 CLAB has been in operation for 13 years. Approximately 1050 transport casks have been received, 970 containing fuel and the remainder highly active core components (control rods etc), see Fig. 1. The fuel amount stored corresponds to 2,752 metric tonnes (U) of uranium. At the end of June 1998, the inventory was:

| | |
|---|---|
| 12,219 | BWR fuel assemblies |
| 1,384 | PWR fuel assemblies |
| 217 | MOX fuel assemblies |
| 222 | Ågesta fuel assemblies (PHWR) |
| 83 | Canisters containing spent core components |
| 16 | Canisters with fuel debris. |

888 Fuel assemblies have been reloaded from the original storage canisters to the new high-density storage canister. The performance of the plant has been excellent. Improvements are gradually introduced along with the experiences gained.

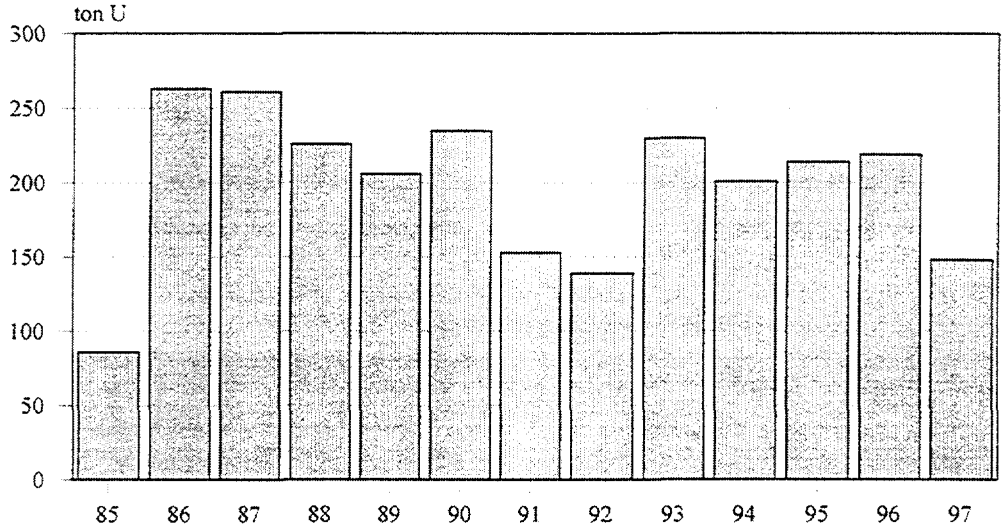


*FIG. 1. Tonnes of U received*

**5.1. Activity release to process systems**

One of the greatest uncertainties during the design period was as mentioned before the amount of crud activity that would be released from the fuel. The fuel is exposed to a certain thermal shock when the dry cask is filled with water in CLAB. Very high crud release figures were reported from abroad. The experience is that the amount released to the cask cooling down system is 50-100 times less than was assumed in the Final Safety Analysis Report (FSAR). This fact may to a great extent be attributed to the good water chemistry in the Swedish reactors and the material in the turbine and feed water systems resulting in relatively small quantities of crud.

During the first month of operation, a problem with the slot filter and backwash filter arose. By developing a new backwash filter and changing the slot filter to a sintered metal filter the problem was resolved. A surprising fact is that the activity release to the storage pool water is for more than 95% ionic. In the FSAR, the opposite was assumed predicting 90% to be in particulate form. The activity released is to 90-95% $^{60}Co$, see Fig. 2, the remainder being mainly $^{54}Mn$, less than 1% is $^{137}Cs$.

The activity concentration with 2,752 metric tonne (U) stored is low, 4 $MBq/m^3$, and has been decreasing the last years. The concentration is, however, higher than expected. The reason for this is the above-mentioned proportion of ionic release. Particles would have settled down on the pool bottom and would not have been observed in the water samples taken from the pools. Instead the activity now remains in the water.

The influence of water temperature on the activity release was measured in 1988, when the cooling was reduced so that the temperature rose from 28°C to 36°C. The activity concentration reached a new equilibrium level 2.1 times higher. The lesson to be learnt from this is that it is an advantage to keep pool water temperature as low as reasonably possible, thereby delaying the release and taking advantage of the decay of the activity while it still is fixed to the fuel surface. The load on the waste treatment systems becomes correspondingly lower.

During 1997, micro-biological fouling was found on the surface of the lining in the storage pools. The germs have been identified as bacterium.

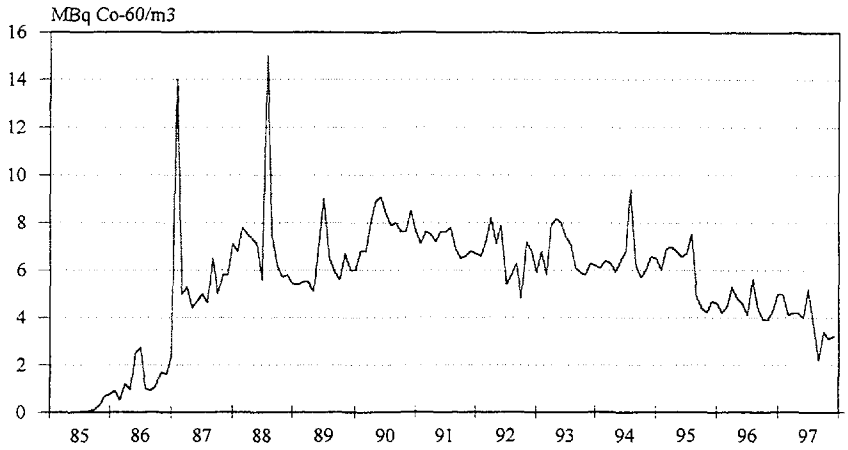


*FIG. 2. Co-60 concentration in storage pools*

## 5.2. Waste treatment

The demands made on the waste treatment systems have so far been much lower than expected. One of the obvious reasons for this is the low activity release from the fuel at arrival and cooling down in the transport cask. The backflush filters in the cooling circuit, where the activity from several casks is collected, were designed for an activity inventory of 300 Ci (11 TBq) $^{60}$Co, which required heavy shielding and handling with remote maintenance casks. After removal, the filters were to be put in heavy cast iron containers constituting the package for final disposal. 2-5 Such containers were assumed to be required annually. Up to now, not one single container of this kind has been used. Instead, the filters, now being exchanged because of high-pressure drop and not high activity, can be put in the concrete moulds normally used for solidifying spent ion exchanger resins, sludge and filter aids. Each mould has an outer volume of 1.7 m$^3$. The number of concrete moulds generated is by a factor of at least 5 lower than originally expected, one of the reasons being the relatively low activity release from the fuel in the storage pools.

After having passed a particle filter, the cooling water from the storage pools is polished in big mixed bed ion exchangers where the ionic impurities are removed. CLAB is equipped with tanks for storage of spent bead resins large enough to permit the resins to remain for decades if necessary before being solidified in concrete. The decay of the $^{60}$Co, which constitutes the major part of the activity makes it possible to put a lot more waste in each concrete mould, the limiting factor being the surface dose rate. 15 years of storage means a factor 10 less activity and a corresponding lower number of waste packages. No concrete mould containing mixed bed ion exchanger resins has yet been cast.

## 5.3. Releases to the environment

The activity release to air and water during the five first years of operation has been negligible, amounting to around 0.01% of the permissible release from CLAB and the three collocated reactors together, which amounts to 1.0 NU/year (0.1 mSv/year)[1]. The activity release to the air has been

[1] NU means "Norm Unit" or "Norm Release"; 1 NU is defined as the dose corresponding to 0.1 mSv given to the critical group near a NPP.

below the detection limit. The release to the water even if extremely low showed an increasing tendency in the first 2.5 years of operation probably due to the increasing fuel inventory. In 1988, a change in wastewater management broke the tendency and, in 1989, the release was one third of that in 1987. The measures taken involved improved filtering and better reuses in the facility of water that earlier were discharged.

### 5.4. Radiation doses

The collective dose to CLAB staff and contractors was for the years 1986-1997 between 65 and 50 mmanSv, which were about 25% of expected values in the FSAR. In 1997, the dose was 53 mmanSv (Fig. 3).

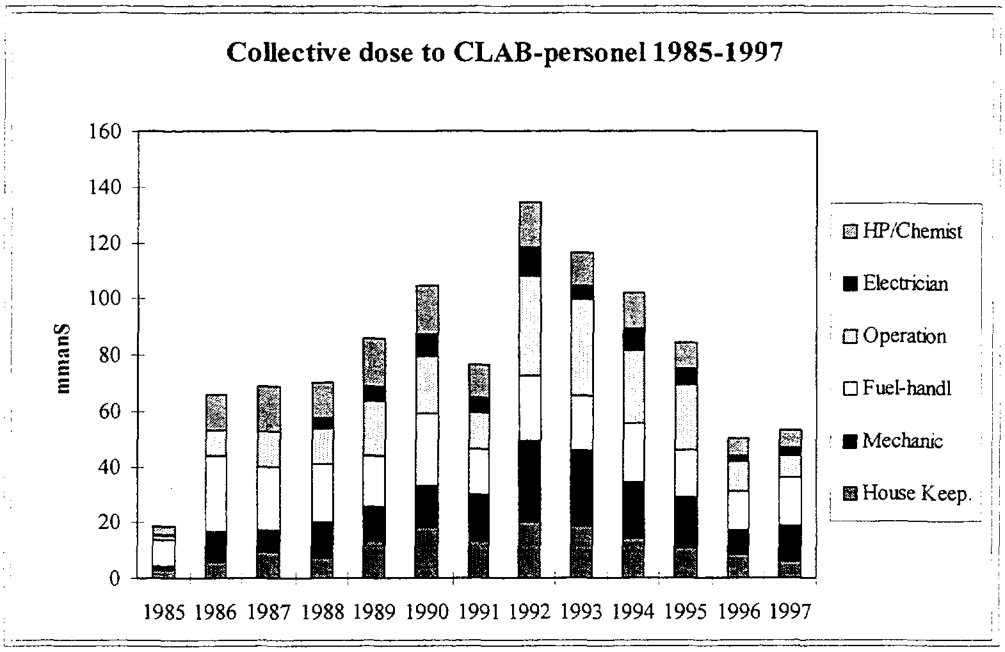


*FIG. 3. Occupational exposure*

## 6. INTERNATIONAL SAFEGUARDS

The experience of the safeguards and the implications for the operation related to the inspections carried out by Euratom and IAEA and is good. The inspections are performed four times a year and take about 1-2 days depending on the type of inspection. Once a year a complete physical inventory verification (PIV) is done. The authorities also perform gamma-scanning of a few fuel assemblies every year.

## 7. FUTURE

SKB has plans for an encapsulation plant to be connected to CLAB. This plant will be in operation in 2008.

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IAEA-SM-352/37

# SPENT NUCLEAR FUEL INTEGRITY DURING DRY STORAGE



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**Abstract**

Information on spent fuel integrity is of interest in evaluating the impact of long-term dry storage on the behavior of spent fuel rods. Spent fuel used during cask performance tests at the Idaho National Engineering and Environmental Laboratory (INEEL) offers significant opportunities for confirmation of the benign nature of long-term dry storage. The cask performance tests conducted at INEEL between 1984 and 1991 included visual observation and ultrasonic examination of the condition of the cladding, fuel rods, and fuel assembly hardware before dry storage and consolidation of the fuel; and a qualitative determination of the effect of dry storage and fuel consolidation on fission gas release from the spent fuel rods. A variety of cover gases and cask orientations were used during the cask performance tests. Cover gases included vacuum, nitrogen, and helium. The nitrogen and helium backfills were sampled and analyzed to detect leaking spent fuel rods. At the conclusion of each performance test, periodic gas sampling was conducted on each cask as part of a cask surveillance and monitoring activity. A spent fuel behavior project (i.e., enhanced surveillance, monitoring, and gas-sampling activities) was initiated by the U.S. Department of Energy (DOE) in 1994 for intact fuel in a CASTOR V/21 cask and for consolidated fuel in a VSC-17 cask. The results of the on going gas sampling activities are reported in this paper.

## 1. INTRODUCTION

In the United States of America, the Nuclear Waste Policy Act of 1982 assigned the U.S. Department of Energy (DOE) the responsibility of assisting utilities with their spent fuel storage problems. In response to the Nuclear Waste Policy Act of 1982, DOE issued a Solicitation for Cooperative Agreement Proposal to help the private sector with their spent fuel storage problems in May 1983, and proposals were received in August 1983. Virginia Power (VP) proposed that pressurized water reactor (PWR) spent fuel storage cask performance testing be conducted at a federal site in support of its at-reactor license demonstration. VP and DOE signed a Cooperative Agreement in March 1984, and VP signed a separate agreement with the Electric Power Research Institute (EPRI), essentially establishing a three-party cooperative agreement. Prior to the solicitation for cooperative agreements, DOE initiated performance testing of boiling water reactor (BWR) spent-fuel assemblies at the General Electric (GE) Morris facility in Illinois.

The scope of the Cooperative Agreement included performance testing three different metal storage casks loaded with unconsolidated spent nuclear fuel. The tests were conducted at INEEL with the GNS CASTOR V/21, Transnuclear TN-24P, and Westinghouse MC-10 casks. After cask performance testing with unconsolidated fuel was completed in the VP/DOE cooperative programme, a decision was made by DOE and EPRI to extend the performance testing to include consolidated fuel in the Transnuclear Inc. TN-24P cask. Dry rod consolidation was conducted at INEEL as a separate DOE-only funded programme.

Prior to testing, the Surry PWR spent fuel assemblies used in the cask performance tests were characterized using in-basin ultrasonic examinations and video scans. Cask internal cover gas samples were taken during testing. After testing, selected fuel assemblies were videotaped and photographed. Then fuel assemblies used in the TN-24P and MC-10 cask performance tests, along with a few Turkey Point reactor spent fuel assemblies, were consolidated and loaded into the TN-24P cask for a DOE-funded performance test. Later, a cooperative agreement was established with Sierra Nuclear Corporation (SNC), and 17 of the consolidated fuel canisters from the TN-24P cask were used in a performance test of SNC's ventilated concrete cask.

Performance test runs involved a combination of cover gases and cask orientations. The backfill environments used were vacuum, nitrogen, and helium; nitrogen and helium were sampled and analyzed to detect leaking spent fuel rods. The integrity of the fuel assemblies was determined from cover gas sampling [1–8]. At the conclusion of each performance test, periodic gas sampling was conducted on each cask as part of a cask surveillance and monitoring activity.

This report combines gas sampling information from cask performance tests and monitoring activities. It documents the condition of the fuel from the Surry reactor before testing and the effect of testing on fuel integrity as ascertained through gas sampling during cask performance tests at INEEL using both intact and consolidated PWR spent fuel. It also includes results of a prior test using BWR fuel and the REA-2023 cask at Morris, Illinois. Recent gas sampling data related to cask surveillance and monitoring at INEEL is also included. The pretest condition of the fuel and the significant results obtained from gas sampling during and after performance testing are described.

## 2. SPENT FUEL INTEGRITY

Three sources of spent fuel and five casks have been used in the cask performance testing and demonstration programme. BWR 7x7 spent fuel assemblies were used for the performance test of the REA-2023 cask. These assemblies were designed by General Electric and taken from the Nebraska Power Cooper reactor. Intact Westinghouse 15x15 PWR spent fuel from VP's Surry reactors were used in the CASTOR V/21, TN-24P, and MC-10 cask performance tests. A portion of this fuel and similar fuel from the Turkey Point Reactor were consolidated at INEEL and used in performance tests of the TN-24P and VSC-17 casks, also at INEEL. Table 1 summarizes the fuel used in each performance test. A more complete description of the fuel is contained within the performance reports.

### 2.1. Pretest fuel inspections

Four examination methods were used to assess the integrity of the spent fuel used in the cask performance tests. Methods common to the PWR and BWR fuel included visual observations (both full-length black and white videos and color photographs) and analyses of the cover gas in the cask. In addition to these methods, the BWR spent fuel was examined by in-basin sipping, and the PWR spent fuel from the VP's Surry Reactor was examined using an in-pool ultrasonic examination.

#### *2.1.1. In-basin sipping*

In-basin sipping consisted of placing a hood over the selected BWR assembly and analyzing the water that was drawn off the top of the assembly. All sipping data were compared with background readings to assess fuel integrity. Although the pretest and post-test radionuclide concentration differences vary, the values were lower than would exist if the assembly contained leaking fuel rods. The sipping results did not indicate any leaking fuel rods in any of the fuel assemblies used in the cask either before or after cask testing.

#### *2.1.2. In-basin ultrasonic inspections*

In-basin ultrasonic inspections were performed on the PWR fuel at VP's Surry Reactor using the Babcock & Wilcox Failed Fuel Rod Detection System (FRDS). The FRDS system uses ultrasonic techniques to differentiate between non-leaking and leaking rods by detecting the presence of moisture in the latter. Only Surry Reactor fuel assemblies with non-leaking fuel rods were used in the performance tests.

#### *2.1.3. Visual, video, and photographic examinations*

The PWR fuel assemblies were examined visually to establish their general condition after shipment from VP, after handling at the INEEL hot shop, after cask performance testing, and during consolidation. Similar exams were made of the Cooper BWR fuel during the REA 2023 performance tests at GE-Morris. Two kinds of visual examinations were used: black-and-white videos and color photography of selected fuel assemblies.

TABLE I. CHARACTERISTICS OF SPENT FUEL USED IN CASK PERFORMANCE TESTS

| | REA-2023 | CASTOR V/21 | TN-24P | TN-24P[a] | MC-10 | VSC-17[a] |
|---|---|---|---|---|---|---|
| Fuel Type | BWR | PWR | PWR | PWR | PWR | PWR |
| Assembly Type | 7 x 7 | 15 x 15 | 15 x 15 | Consolidated 15 x 15 | 15 x 15 | Consolidated 15 x 15 |
| Burnup, GWd/MTU | 24-28 | 24-35 | 29-32 | 24-35 | 24-35 | 26-35 |
| Cooling Time, years[b] | 2.3-3.4 | 2.2-3.8 | 4.2 | 6.2-12.2 | 4.6-10.1 | 8.8-14.3 |
| Discharge Date(s) | 1981-82 | 1981-83 | 1981 | 1975-81 | 1975-81 | 1976-81 |
| Enrichment, wt% | 2.5 | 2.9-3.1 | 2.9-3.2 | 1.9-3.2 | 1.9-3.2 | 2.56-3.2 |
| Test Temperatures, Peak Clad, C | 110-230 | 350-405 | 215-290 | 205-295 | 140-220 | 320-400 |
| Assembly Decay Heat, W | 235-370 | 1,000-1,800 | 832-919 | 701-1,185 | 400-700 | 700-1,050 |
| Average, W | 290 | 1,350 | 860 | 970 | 530 | 877 |
| Cask, kW | 15.2 | 28.4 | 20.6 | 23.3 | 12.6 | 14.9 |

[a] Performance test using consolidated fuel in the cask.
[b] Cooling time at the beginning of the cask performance test.

The black-and-white videos taken at GE-Morris, VP, and INEEL did not provide sufficient detail to characterize the crud or very small features on the fuel rods. They did not reveal any indication of significant variations in the fuel rods after shipment, handling, and performance testing. The resolution of the videotapes did not provide enough information to adequately determine the integrity and condition of the fuel and fuel cladding. Examination of the video scans showed that all the fuel assemblies and fuel rods look basically the same when viewed from outside the assemblies. There was some discoloration of the fuel rod cladding in the area of the grid spacers, which was expected.

Color photographs showed that a typical orange/reddish crud (probably $Fe_2O_3$) was evenly deposited on all of the Zircaloy 2 cladding and fuel assembly hardware. There were no noticeable changes in the characteristics or adherence of the crud during handling operations involving the spent fuel assemblies at GE-Morris or INEEL. Some scratches and worn spots were apparent on the spacer grids and some fuel rods, but these features did not change as a result of examination or handling operations. In general, the fuel rods were in excellent condition with a very adherent crud layer.

Additional visual examinations of the fuel were conducted during the dry rod consolidation programme. According to Vinjamuri (see [9 and 10]):

> No noticeable cladding defects in the rod surfaces were observed for any of the fuel processed. The oxide layer on the surface of the fuel rods appears to be intact and firmly attached to the cladding. The oxide layer does not appear to be loose, thick, soft, or powdery. However, the oxide layer and some of the zirconium cladding was scraped from the rod surface by the spacer grids as the rod was pulled during fuel consolidation. Very little crud buildup on the surfaces of the rods was observed. The surfaces of the rods displayed only a thin oxide layer, which had the appearance of surface discoloration rather than any rough or loose material. The rod surfaces are discolored near the spacer grids. The discoloration has an appearance of a dark mottling of the surface and is progressively

more predominant from the middle of the rod length toward the rod bottom. The rods are generally clean, with limited amounts of clad discoloration and oxidation . . . The evidence of fuel rod growth since fabrication was visually obvious during the consolidation process . . . Length variation between rods appears to be as much as 2 cm (0.8 inch). The rods that grew longer than others appeared to be randomly located within the fuel assembly. (Ellipses byMcKinnon.)

#### *2.1.4. Cask Cover Gas Sampling*

The cask cover gas was sampled several times during each cask performance test to evaluate the integrity of the spent fuel rods. Each sample was collected in a separate 500-cc stainless steel cylinder. The cylinders were checked for leaks before sampling. Initially, during the CASTOR-V/21 cask performance test, the cylinders were equipped only with quick disconnect-fittings and no bellows-sealed valves as part of the closure. During the early sampling efforts with the CASTOR-V/21 cask, the cover gas samples in the cylinders were diluted with ambient air from the vicinity of the sampling apparatus, air that leaked into the cylinder during shipment, and argon introduced at Lawrence Livermore National Laboratory (where some of the samples were analyzed). In many cases, this dilution was made more severe by the collection of small amounts of cask cover gas, presumably due to short equilibration times between the cask and the sample bottle during the actual cask cover gas collection procedure. The end effect of small, diluted samples on the cask cover gas analyses was to increase detection limits and measurement uncertainties and introduce questions of sample validity. Once bellows-sealed valves were added to the sampling cylinders, the problem of air leakage into the sampling cylinders was eliminated.

Gas sample analysis included mass spectroscopy and radiochemical gamma analysis. Mass spectra were analyzed for all common fixed gases with masses less than 100 to verify the purity of backfill gas composition. Only $N_2$, $0_2$, He, Ar, and $CO_2$ concentrations above 0.01 % are detected in any of the samples. The integrity of the fuel rods was assessed from the radionuclide concentration based on gamma spectroscopy.

### 2.2. Fuel integrity based on cover gas sampling results

Radiochemical gamma analysis was used to detect krypton-85. The relatively low amounts of krypton-85 detected indicate that no leaking fuel rods were present in the GNS CASTOR-V/21 and MC-10 casks during performance testing with unconsolidated fuel and up to about a year after testing. At this time gas sampling in these casks was discontinued. The final gas sample taken from the CASTOR-V/21 cask during this period of time was taken in December 1986. In September 1994, the CASTOR cask was opened and backfilled with a fresh charge of helium gas. The pre- and post- test backfills were checked for purity. The krypton-85 concentration in gas samples taken from the cask in March and July 1995 (after six and nine months of residence in the cask, respectively) were less than would result from a single leaking fuel rod, indicating that there were no leaking fuel rods during this storage period. Yearly gas samples have been taken since 1995 with the most recent gas sample being taken in August 1998. Analysis of these gas samples continue to show that the fuel assemblies are intact and are not leaking fission gases. This is particularly significant because the first few assemblies loaded in the CASTOR-V/21 cask were exposed to air for approximately 200 hours during incremental loading of the cask and fuel assembly/basket inspections at a reduced temperature. In addition, after testing was completed and long-term surveillance started, all the fuel assemblies were in a 70% helium and 30% air environment for approximately four months, because a quick disconnect fitting on the CASTOR-V/21 cask lid had not sealed properly.

Two casks loaded with intact fuel have shown krypton gas concentrations indicative of a leaking fuel rod. During the performance test of the REA-2023 cask loaded with BWR fuel, krypton gas was detected in the cask after being rotated from a vertical to horizontal orientation. The accumulated amount of krypton gas released to the cask was consistent with the release from a single BWR fuel rod (see [11, 12]), likely a corner rod that experienced higher power during reactor operation. The total krypton released represented about nine percent of the krypton gas generated in a single rod. The cladding defect was assumed to be very small since the release rate was essentially linear during 2.5 months of testing. There was no confirmation of a leaking fuel rod either by visual inspection or sipping of the fuel assemblies after the cask test. The gas analyses provided the only indication of a leaking fuel rod. The leaking fuel rod had no impact on the basin operation or handling of the fuel assemblies subsequent to the cask test.

The other cask loaded with intact fuel that showed krypton levels indicative of a leaking fuel rod was the TN-24P cask loaded with PWR fuel. In this performance test, the cumulative amount of krypton-85 detected just after the cask was rotated from a vertical to a horizontal orientation indicated a fuel rod leaked during this portion of the test. A value of 0.5% release of the total krypton-85 gas in a single fuel rod was used as the basis for assuming one leaking fuel rod. The rest of the gas was assumed to be captured in the fuel. The decay in the leak rate, as indicated by subsequent gas samples, indicates that the leak was small. It took several days to vent the gas from the fuel rod.

In May 1987, 36 of the 48 intact fuel assemblies in the TN-24P and MC-10 casks and 12 intact assemblies that had been in the Turkey Point Reactor were consolidated into 24 fuel canisters as part of INEEL's Dry Rod Consolidation Technology Project. The consolidated fuel canisters were then used in performance tests of the TN-24P and VSC-17 casks. During the fuel rod consolidation process, the exhaust gases from the consolidation area were monitored to detect the release of radioactive gases from the fuel that would indicate a cladding failure. In the consolidation reports (see [9, 10]), one of the conclusions reached was that all fuel rods from the 48 assemblies were pulled and canisterized without rod failures or leakage.

Later, during the performance test of the TN-24P cask using consolidated fuel, krypton-85 was released to the cask. Based on a combination of ORIGEN2 predictions and experimental measurements (see [1, 12]), the expected krypton-85 gas release for a single rod was selected to be 0.5% of the total rod inventory of krypton-85. Using this value for krypton-85 release, it was estimated that four or more fuel rods may have developed leaks between the end of cask loading and the beginning of cask performance testing, three or more fuel rods during cask performance testing, and another five fuel rods in the six-month period following testing. The rate of krypton-85 release was observed to decrease with time from cask loading. Shortly after the last gas sample was taken from the fully loaded TN-24P cask, 17 canisters of consolidated fuel were removed from the TN-24P cask and loaded into the VSC-17 cask. The performance tests for the VSC-17 cask showed a nominal amount of krypton-85 but not enough to indicate a new leaking fuel rod. From the end of the VSC-17 performance testing in early 1991 until September 1994, the VSC-17 was undisturbed. Gas samples, taken from September 1994 to August 1998, indicate that the atmosphere in the VSC-17 has not changed significantly. There has been a small amount of krypton-85 release, below the quantity expected for a single rod release, and there has been a small buildup of hydrogen in the cask. The amount of hydrogen (~2%) is consistent with off-gassing of the RX277 neutron shield material in the lid and appears to have reached an equilibrium level. The hydrogen in the cask is an artifact of penetrations made through the lid to accommodate temperature measurement instrumentation. These penetration are peculiar to the test lid and would not exist in the standard lid design. Similar amounts of hydrogen were observed during cask performance testing.

The amount of krypton-85 released during and after the TN-24P cask performance test with consolidated fuel is significantly higher than that released in previous cask testing with unconsolidated fuel. Before this test, four cask performance tests of similar duration and scope had been performed; only two indications of krypton-85 release were observed. The magnitude of the releases in the previous tests and surveillance periods indicated that each was limited to a single rod cladding breach. The previous tests involved about 16,700 spent fuel rods, whereas this test involved about 9800 rods. It is hypothesized that the greater magnitude of krypton-85 released in this test and post-test surveillance is because of additional cladding leaks caused by enlargement of incipient cladding flaws during pulling and flexing of the fuel rods during the consolidation process. The enlarged cladding flaws combined with cladding creep during cask testing and surveillance periods allowed leak paths to develop. The leakage has not affected operations.

## 3. SUMMARY

Radiochemical gamma analysis of gas samples from cask performance tests and subsequent cask surveillance and monitoring activities provide an indication for spent fuel integrity during dry storage. The gas sampling analysis indicates that dry storage of spent fuel in an inert atmosphere is benign. In general, fuel handling activities have a more significant impact on fuel rods than does extended dry storage in an inert atmosphere.

## ACKNOWLEDGEMENT

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# SESSION 4

# R&D AND SPECIAL ASPECTS

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# CREEP PROPERTIES OF NON-IRRADIATED Zr1Nb CLADDING TUBES UNDER NORMAL AND ABNORMAL STORAGE CONDITIONS

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**Abstract**

The Czech Republic has, for its long-term storage of WWER-440 spent fuel, opted for the dry storage alternative and uses the CASTOR 440/84 casks for this purpose. The maximum burnup of the spent fuel stored under a protective helium atmosphere in these casks is limited to 40 MW·d/kg U and the cladding temperature should not exceed 350°C. This temperature limit was established as a result of corrosion cracking analyses. However, with regard to the low pressure of the fission gases it can be anticipated that corrosion cracking will not be a critical mechanism of the cladding degradation. Thus, the damage is creep - controlled, as it was also indicated in the Russian paper [1] and the temperature of 350°C is unnecessarily conservative. This was a reason, why ČEZ, a.s. (the Czech Power Utility), supported by the Ministry of Industry and Trade and the State Office for Nuclear Safety (SONS), initiated an experimental study of Zr1Nb creep behaviour. This paper summarises experimental results of thermal creep of non-irradiated Zr1Nb cladding tubes after recrystallization annealing, which modelled the creep behaviour of dry stored WWER spent fuel under normal and accident conditions. Determination of the storage limiting temperatures, at which there will be no creep caused damage over 40 and 70 year of storage period, taking into account the irradiation and „cooling" history of fuel, is also discussed. Zircaloy-4 specimens, exposed to creep similarly as Zr1Nb ones, served to verify the tests methodology and the evaluation applied.

## 1. EXPERIMENTS

### 1.1. Tested materials and manufacturing of creep specimens

Creep test specimens were made of standard Zr1Nb and zircaloy-4 (Zry-4) cladding tubes supplied respectively by Russia and Sweden (SANDVIK). Their nominal external diameter ($D_0$) and wall thickness (t) were:

| | | |
|---|---|---|
| Zr1Nb: | $D_0 = 9.178 \pm 0.010$ mm; | $t = 0.709 \pm 0.018$ mm |
| Zircaloy -4: | $D_0 = 10.178 \pm 0.009$ mm; | $t = 0.712 \pm 0.013$ mm. |

Zircaloy-4 tubes annealed to remove the cold forming stress have significantly higher strength properties. On the other hand, recrystallized annealed Zr1Nb tubes have higher values of uniform elongation, both axial and circumferential. Mechanical properties of the tubes depend also on their oxygen content, see Table I.

The full-length tubes of both alloys were cut into 120 and 100 mm sections (semi-products) which were then used to manufacture 200 Zr1Nb creep test specimens and 140 Zry-4 specimens of the same kind. To determine the average ovality the wall thickness of each specimen was measured at roughly 25 perimeteral points at their half-length, and the external diameter - with an accuracy of 0.001 mm. The minimum wall thickness was marked on the specimens external surface - to establish a possible anomalous creep behaviour.

TABLE I. MECHANICAL PROPERTIES OF Zr1Nb AND Zry-4 TUBES (AS SUPPLIED STATE)

| Alloy | Property [Unit] | Zr1Nb XR | Zry-4 SR |
|---|---|---|---|
| Tensile tests at 20°C : | $R_{p0.2}$ [MPa] | 223 | 550 |
| | $R_m$ [MPa] | 407 | 720 |
| | $A_{50}$ [%] | 39 | 21 |
| Tensile tests at 350°C : | $R_{p0.2}$ [MPa] | 105 | 300 |
| | $R_m$ [MPa] | 198 | 412 |
| | $A_{50}$ [%] | 50 | 25 |
| Burst tests at 20°C : | $R_{p0.2}$ [MPa] | 461 | 745 |
| | $R_{mo}$ [MPa] | 568 | 896 |
| | $A_{50}$ [%] | 45 | 12 |
| Burst tests at 350°C : | $R_{mo}$ [MPa] | 280 | 480 |
| | $A_o$ [%] | 36 | 12 |
| Approximate Oxygen content | [weight ppm] | 500 | 1,000 |

In the case of Zr1Nb alloy, the initial dimension and their range are somewhat different from the nominal ones for the cladding tubes, mentioned above, this is immaterial. More important, however, is the fact that these tube specimens, as a result of the tube manufacturing technology applied, have a significantly larger ovality compared with that for Zircaloy-4 alloy. Differences between the maximum and minimum wall thickness are in the range of 15 to 80 mm. Such significant ovality indicates that a larger scatter of the results should be expected, as well as tendency to non-uniform deformation and its early localisation.

Creep tests were carried out on the specimens with welded closing caps pressurised by pure argon: as a first step - both closing caps were electron-beam welded to the tube semi-products, one of these caps had a 0.7 mm filling-in hole. Actual pressurisation was carried out in a special evacuated and pressurized box equipped with a welding electrode. After a specimen was put into correct position, the box was several times evacuated and rinsed with argon and, finally, pressurized to the required pressure and then the hole was welded by using the TIG (tungsten inert gas) technique.

### 1.2. Testing equipment and its temperature characteristics

#### *1.2.1. Testing equipment for the temperature range 325 - 400°C*

Creep tests were carried out in a laboratory muffle furnace provided with a proportional temperature controller and a maximum temperature cut-out. Two furnaces were operated at the same time, each for a different test temperature. Tests were performed in an air environment. The test specimens were placed horizontally into the heat-resistant tubes (block of tubes, each 150 mm long and 35 mm diameter) installed inside the furnace. 35 Creep specimens were tested for each selected temperature.

The furnace temperature is measured with four jacketed thermocouples, fixed in the block of heat-resistant tubes. The measured temperature is periodically registered in the measuring computer unit. The maximum difference between data provided by the individual thermocouples was about 5°C, and the average furnace temperature was maintained with an accuracy of ± 4°C. The procedure of experimental results evaluation took into account the registered temperature differences.

#### *1.2.2. Testing equipment for the temperature range 420 - 530°C*

To limit specimen oxidation, these tests were performed in an argon environment.

The equipment consisted of a resistance furnace with programmable temperature controller and instrumented tube retorts, made of sintered corundum. The retorts were equipped with valves, which enables to evacuate and fill them with argon, and with a cap, housing the draw rod of the specimens holder as well as four measuring thermocouples.

The furnace volume was evacuated with a rotary vacuum pump, the residual pressure of < 10 Pa is sufficient. Argon was fed (up to atmospheric pressure] through the rubber bellows which during heating-up and cooling-down periods also served as a pressuriser. The whole capacity was 30 specimens, which were positioned horizontally and supported at both ends. The measuring ends of thermocouples were fixed within the specimen holder, in its upper and bottom parts, which enables to register the vertical differences for each experimental temperature.

The pre-experimental temperature distribution and continuous temperature measurements, registered each 5 minutes for all thermocouples in the course of the actual creep tests, have provided the following characteristics of the test facility under stabilised conditions:

- temperature differences in the constant temperature zone (250 mm) : ≤ 4°C;
- vertical temperature differences in both test sections : ≤ 0.5°C;
- the average temperature differences in both test sections : ≤ 2°C.

### 1.3. Test conditions and experimental procedure

Creep specimens pressurized up to basic pressure to achieve the hoop stresses in the range from 50 to 110 MPa for the temperatures of 325 - 400°C and from 40 to 130 MPa for the temperatures of 420 - 530°C, were weighted at analytical balance. The accurate hoop stress values were determined from the mass of argon and internal dimension of each specimen. Five stress levels were used for each testing temperature, always for three Zr1Nb specimens or two Zry-4 specimens. The initial minimum and maximum diameter of the specimens has been measured along the whole length, in prior established intervals, by a digital micrometer. The length of the specimens was measured by a digital slide gauge.

The actual exposures of the specimens at the selected temperatures were achieved by putting these specimens into the pre-heated furnace under atmospheric air conditions. Test in argon were started by moving the specimen holder into the pre-heated furnace without losing the argon protecting atmosphere. The actual beginning of the exposition was established on the basis of the specimens heating rate tests. After the required time had elapsed, the specimens were either removed from the furnace or moved into its cold part.

A control weighting of the specimens was performed after each exposure, as well as dimensional measurements, and for significantly deformed specimens - also the volume determination. Usually, the creep tests were finished at each temperature after the demonstrable steady-state creep had been achieved for all specimens with the lowest hoop stress. In all cases, the end of the tests was decided in accordance with the Zr1Nb alloy creep behaviour.

The total time of the tests at the individual temperatures decreased from 9,600 hours at the temperature of 325°C to 23 hours at 530°C.

## 2. DETERMINATION OF THE CREEP EQUATION CONSTANTS FOR NORMAL AND ABNORMAL STORAGE CONDITIONS

The experimental results were expressed in the form of creep curves - time dependence of the hoop strain. Stationary creep rate, saturated transient strain and the transient strain course were evaluated for each individual creep curve. The obtained results confirmed certain differences in the Zr1Nb and Zry-4 creep behaviour. The primary (transient) stage for Zr1Nb alloy is practically always shorter and its saturated transient strain is low. At higher temperatures tertiary creep sets in Zr1Nb earlier. The general tendency, to a higher creep resistance of recrystallized material compared with the annealed one, has also been confirmed at low stresses. The creep equation derived in [2] was used to model the time dependence of the deformation.

The overall hoop creep strain of a tube specimen, creep stressed up to the steady creep region, is composed of the transient and stationary deformation parts and is described by the following set of equations:

$$\varepsilon = \varepsilon_t^s\left[1 - e^{\{-D.(\dot{\varepsilon}_s t)^n\}}\right] + \dot{\varepsilon}_s t \tag{1}$$

where:

$\varepsilon$ - hoop strain

$\varepsilon_t^s$ - saturated transient strain

$\dot{\varepsilon}_s$ - steady state creep rate

$$\dot{\varepsilon}_s = B \times (E/T)\, e^{(C.\sigma/E)} \times e^{(-Q/RT)} \tag{2}$$

$$\varepsilon_t^s = e^{(FT+G)} \times \dot{\varepsilon}_s^{(HT+I)} \tag{3}$$

$$D = K \times e^{(L.T)} \tag{4}$$

where:

$\sigma$ - applied hoop stress [MPa],
E - Young modulus of Zr1Nb alloy ($E = 1.121\ 10^5 - 64.4\ T$), [MPa] [3]
T - temperature [°K]
Q - apparent activation energy [J/mol]
R - gas constant (8.314 J/mol)
B, C, F, G, H, I, K, L and n - experimentally obtained constants.

To obtain the constants in the equations, the following procedure has been applied:

1. In evaluating the stress dependence of the stationary creep rate for individual temperatures within the anticipated range of equations validity, the C values for the individual temperatures are derived;
2. Using the average C value, the temperature dependence of the stationary creep rate is evaluated and the Q and B values are obtained;
3. In evaluating the dependence of the saturated transient strain on the stationary creep rate and temperature, the F, G, H and I constants are obtained;
4. In evaluating the time dependence of the transient strain, the n and D constants are obtained; for the temperature range between 325 and 400°C the D constant is temperature dependent, which may be expressed by using the L and K constants.

The C and Q constants for Zry-4 were determined to verify the test methodology and the evaluation procedure. The obtained values C = 2,300 and Q = 207,000 J/mol were in good compliance with the results included into [2].

Two groups of constants for both the temperature and stress ranges included in Table II, were determined for Zr1Nb alloy. For the interval of temperatures between 400 - 420°C the calculation is assumed to be in accordance with both models; a higher value of deformation for the given stress and time will be used in further applications.

TABLE II. CREEP EQUATIONS CONSTANTS

| Temperature | 325 - 400°C | 420 - 530°C |
|---|---|---|
| Hoop stress | 50 - 110 MPa | 40 - 130 MPa |
| C | 2,400 | 3,900 |
| B | $1.3\ 10^2$ | $8.718\ 10^6$ |
| Q | 185,000 J/mol | 256,000 J/mol |
| F | -0.0782 | - 0.0700 |
| G | 59.8120 | 53.2300 |
| H | -0.002410 | - 0.002767 |
| I | 2.2720 | 2.4650 |
| D | K = 56140<br>L= -0.009547 | 45 |
| n | 0.5 | 0.6 |

More accurate values of the creep equations constants will result from the experiments in progress, with gradually increasing and decreasing the temperature within the given temperature range, taking into account the model of the deformation hardening, taken from [4 and 5]. Fig. 1 and 2 show experimental values of the hoop strain in comparison with the values computed using the equations above, for the temperatures of 350°C and 450°C, that is for the normal and abnormal storage conditions.

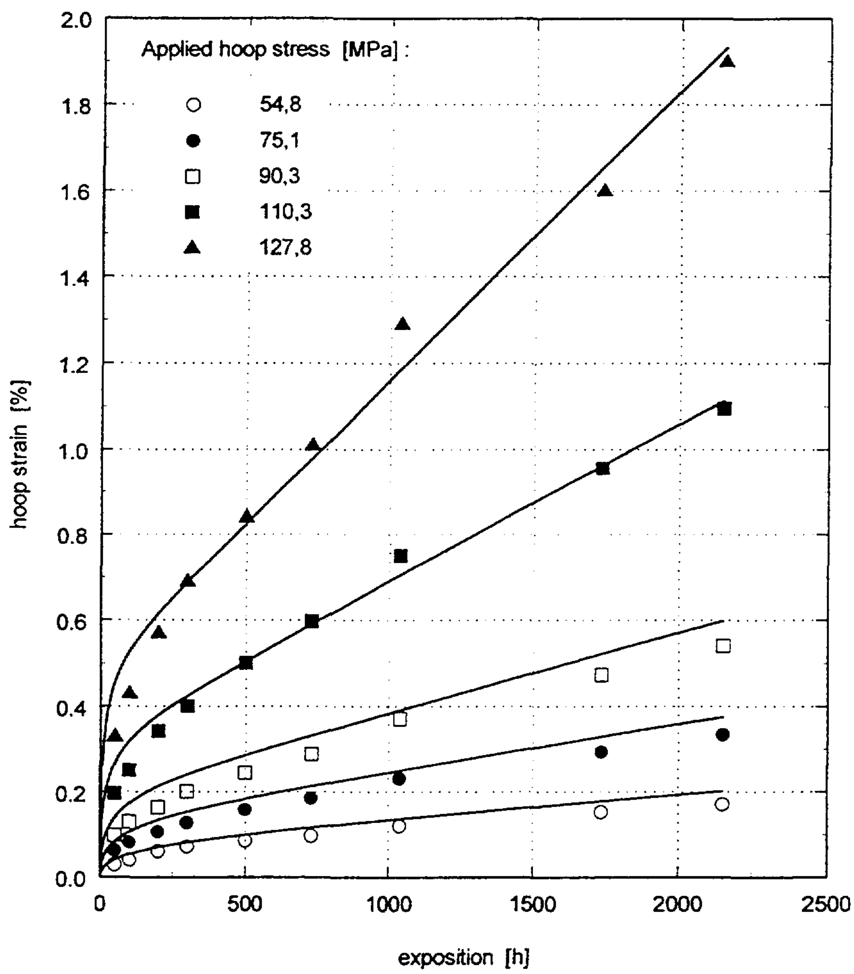


*FIG. 1. Comparison between experimental measured creep strain values and predicted creep curves at temperature 381 °C*

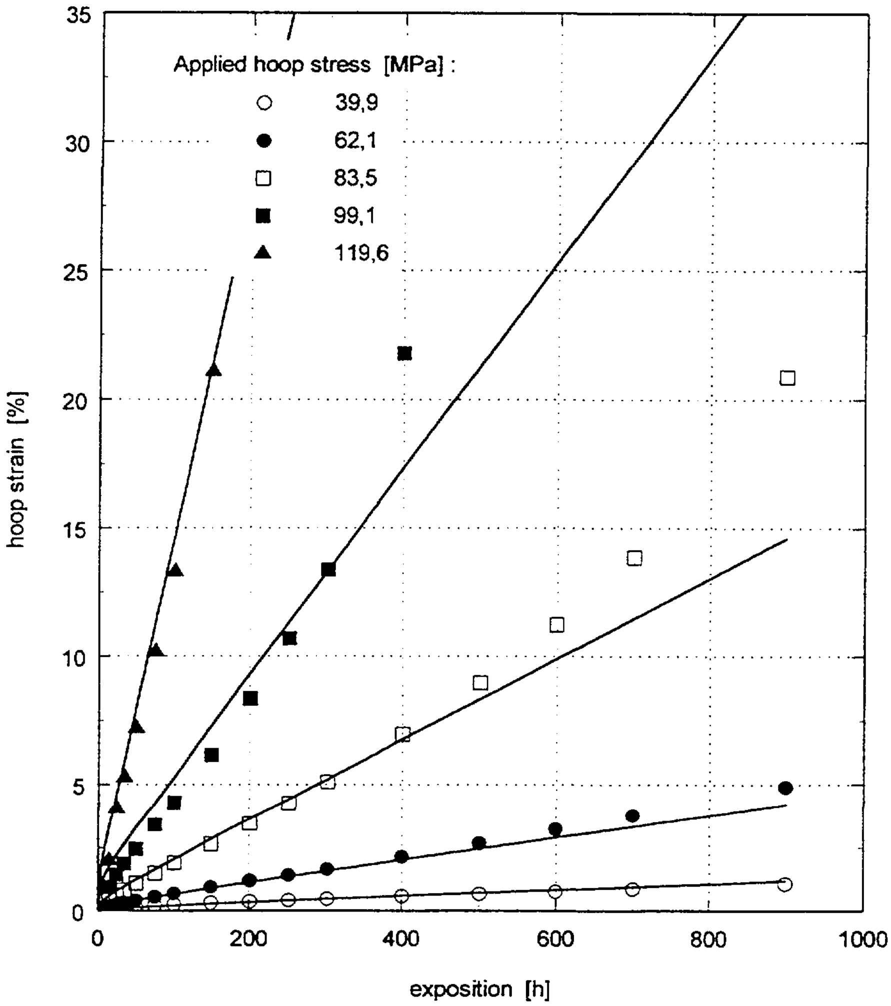


*FIG.2. Comparison between experimental measured creep strain values and predicted creep curves at temperature 453 °C*

## 3. METHOD OF EVALUATION OF MAXIMUM ALLOWABLE TEMPERATURES OF THE SPENT FUEL CLADDING

In accordance with the proposed criterion of 1% total hoop strain for long-term storage, which should eliminate that the stored fuel rods may lose their initial tightness [6], a computer code was developed which is capable to calculate the maximum allowable WWER fuel cladding temperatures during cooling within CASTOR 440/84 casks. The code enables to calculate the limiting cladding temperatures in dependence of the internal gas pressure in the spent fuel rods with a burnup up to 40 MW·d/kg U [7]. Fig. 3 presents the code block diagram. The computer code was tuned by using the creep equation for Zry-4 alloy taken from [2]. Implementation of the Zr1Nb creep equations is foreseen after the evaluation of the experiments with increasing and decreasing temperature is finalised.

The computer code used to calculate the increase of the cladding creep deformation may also serve to assess the progress of vacuum drying after a cask is filled, using the criterion of maximum 0.1 % of hoop strain [8]. The same code is applicable for the assessment of casks emergency states (insufficient cooling) by using the criterion of maximum 10 % hoop strain [9].

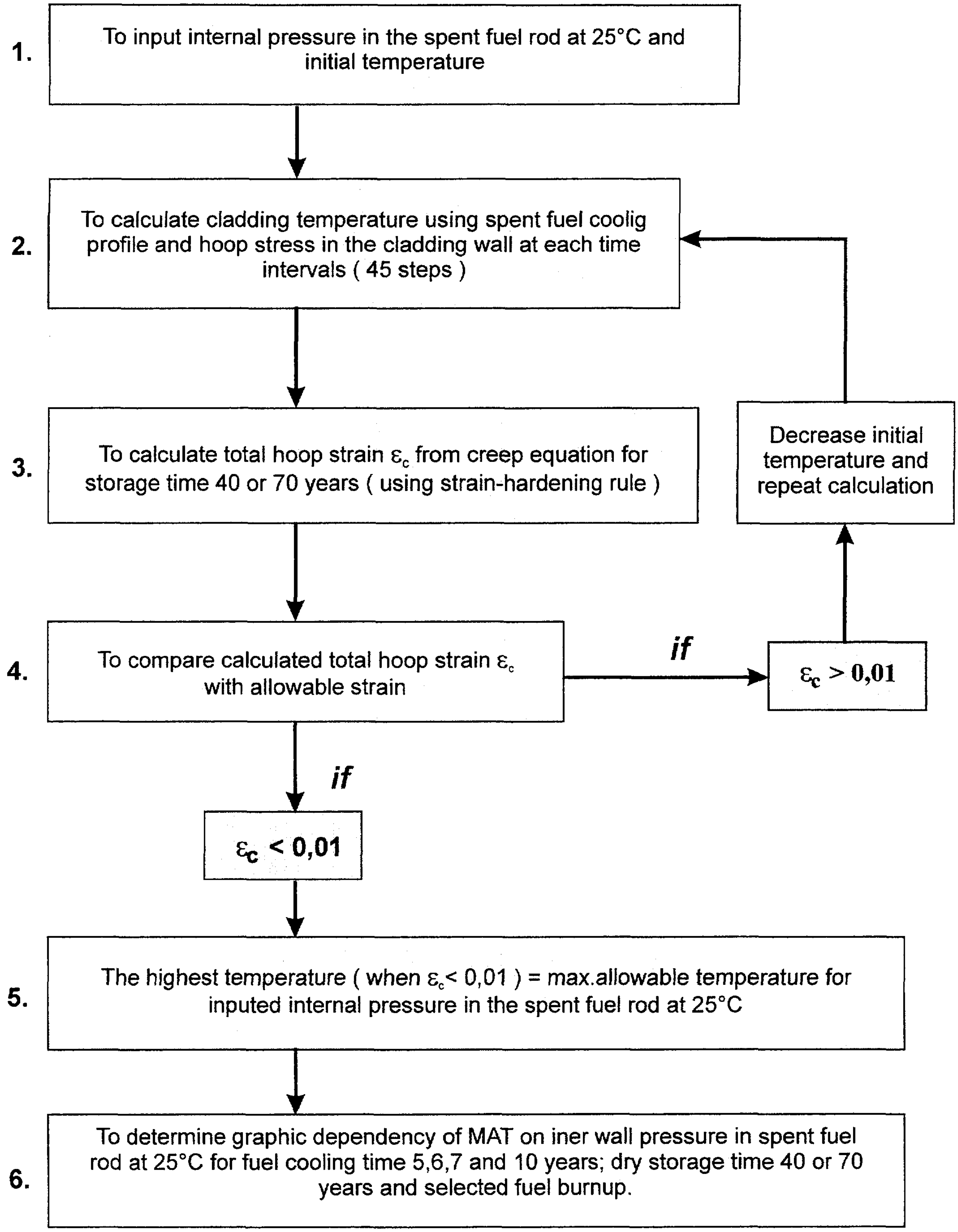


*FIG. 3. Block diagram of the computer code to determine the maximum allowable temperature of fuel cladding*

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# ASSESSMENT OF DRY STORAGE PERFORMANCE OF SPENT LWR FUEL ASSEMBLIES WITH INCREASING BURNUP



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**Abstract**

Although the safety of a dry long-term spent fuel store is scarcely influenced if a few fuel rods start to leak during extended storage - since all confinement systems are designed to retain gaseous activity safely - it is a very conservative safety goal to avoid the occurrence of systematic rod defects. To assess the extended storage performance of a spent fuel assembly (FA), the experience can be collated into 3 storage modes: I - fast rate of temperature decrease $\delta_{max} \geq \delta \geq 300°C$, II - medium rate of decrease for the fuel rod dry storage temperature $300°C > \delta \geq 200°C$, III - slow to negligible rate of temperature decrease for $\delta < 200°C$. Mode I is typical for early interim storage, mode III covers extremely long-term storage. Mode II dry storage is characterised by the fact that all creep deformation of the spent fuel cladding can already be regarded as terminated as well as the corrosive attack of the cladding. Reviewing the fission product behaviour results shows that the fission products in the $UO_2$-fuel are practically immobile during storage. Consequently all fission-product-driven defect mechanisms will not take place. The leading defect mechanism - also for fuel rods with increased burnup - remains creep due to the hoop strain resulting from the fuel rod internal fission gas pressure. Limiting the creep to its primary and secondary stages prevents fuel rod degradation. The allowable uniform strain of the cladding is 1 - 2%. Calculations were performed to predict the dry storage performance of fuel assemblies with a burnup $\leq 55$ GW·d/tHM based on the fuel assemblies end of life (EOL)-data and on a representative curve T = f(t). The maximum allowable hot spot temperature of a fuel rod in the CASTOR V cask was between 348°C (U FA) and 358°C (MOX FA). The highest hoop strain predicted after 40 years of storage is 0.77% proving that spent LWR fuel dry storage is safe.

## 1. INTRODUCTION

Nuclear fuel supply and disposal in Germany is characterised by a situation whereby key cost components are assessed in specific terms, i.e. per kg of processed fuel [1]. It thus becomes immediately clear that, apart from the respective specific costs, it is above all the mass of fuel required to generate a given quantity of electric power that has the greatest influence on the total cost. A reduction in the mass of fuel that is in circulation means, in particular, significant savings in storage and disposal costs. The key to any reduction in the mass of fuel in circulation within the nuclear fuel cycle is the fuel assemblies themselves and the in-core fuel performance of which determines how much fuel (enriched uranium or MOX) is needed to generate a given quantity of electric power [2]. Figure 1 uses Siemens fuel assemblies for PWRs as an example to illustrate developments over the last three decades.

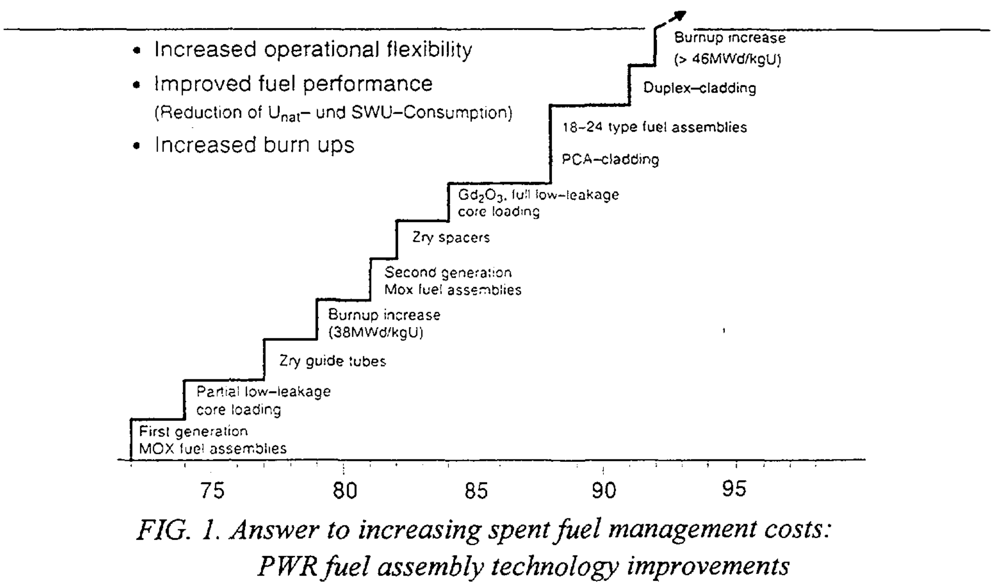



*FIG. 1. Answer to increasing spent fuel management costs: PWR fuel assembly technology improvements*

Fuel utilisation and operating flexibility optimise uranium and separative work requirements and in-core fuel management. However, the most important part of optimising fuel performance lies in the possibility of achieving higher burnup, which have an inversely proportional effect on the mass of spent fuel to be disposed of per generated $kW{\cdot}h_e$. Advances made in fuel assembly design and manufacture have now made batch average discharge burnup of over 55 MW·d/kgU achievable.

Figure 2 shows the corresponding development in the average discharge burnup of the Siemens reload fuel assemblies most frequently supplied for BWRs and PWRs. It can be concluded from this figure that, from the point in time at which a sharp rise was experienced in specific disposal costs, higher burnup enabled the mass of fuel requiring disposal to be reduced by:

- approximately 28% in the case of PWRs and even by around;
- 42% in the case of BWRs.

This also explains how it has been possible to achieve continual reductions in fuel cycle costs per $kW{\cdot}h_e$ since the mid-1980s, in spite of disposal costs that continue to rise. For direct disposal of spent nuclear fuel there is, however, the need to store the spent fuel for an extended period of time. The necessary interim storage might last between 40 and 100 years. In Germany, the dry storage technology had been chosen to store the spent fuel after its removal from wet storage in the pool of the reactor [3]. This paper reviews the dry storage performance of spent LWR fuel with a burnup reaching values up to 55 GW·d/tHM.

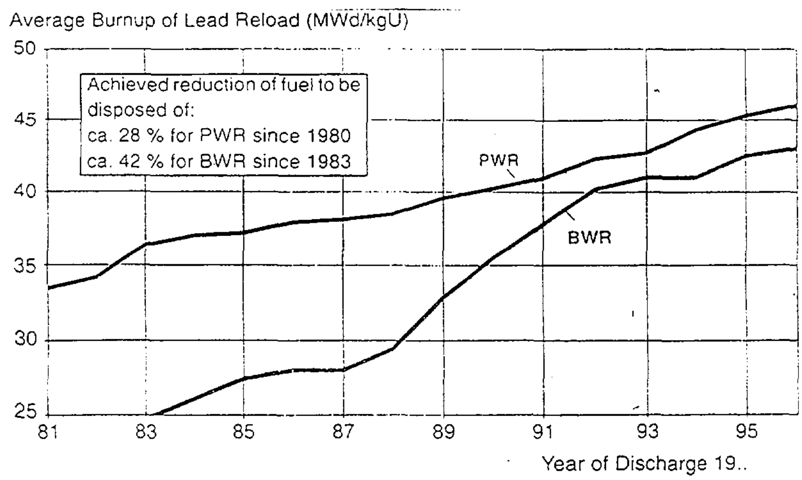



*FIG. 2. Burnup increase reduces the circulating fuel quantity and thus reduces the primarily the spent fuel management costs*

## 2. DRY INTERIM STORAGE OF FUEL ASSEMBLIES WITH INCREASED BURNUP

### 2.1. Definition of storage modes with regard to spent fuel dry storage performance

To assess the performance of spent LWR-fuel in extended storage, the available experience can be collated into 3 storage modes:

- **mode I:** fast rate of decrease in temperature between maximum of licensed dry storage temperature and 300°C;
- **mode II:** medium rate of decrease for in the fuel rod dry storage temperature between 300°C and 200°C;
- **mode III:** slow to negligible rate of decrease in the fuel rod dry storage temperature for temperatures less than 200°C.

**Mode I** is typical for early interim storage, **mode III** covers extremely long-term storage which is encountered presumably for nearly all dry storage extensions to be considered. **Mode II** dry storage is characterised by the fact that all creep deformations of the spent fuel cladding can already be regarded as terminated as well as the corrosive attack of the cladding. Under the assumption of air ingress to the inert system there - if the oxidative condition holds for a longer period of time - $UO_2$-fuel-oxidation needs to be considered in more detail. If the $UO_2$ will be converted in such a case to $U_3O_8$ the fuel will swell and may lead to fuel rod splitting by the mechanical stress applied to the cladding by the oxidised fuel.

### 2.2. Source term considerations

Higher fuel assembly burnup leads to higher decay heat power levels and higher n- and γ-source terms [4] (Fig. 3). This correlation is not generally linear in nature, but is frequently characterised by a disproportionate increase with rising burnup levels. These two aspects are in fact in conflict with each other here:

- on the one hand, the cask walls must be as thick as possible and contain sufficient moderator material in order to provide shielding against n- and γ-radiation;
- on the other hand, however, it is precisely this kind of configuration which hampers heat removal, which would be better with thin cask walls.

Cask design thus incorporates a compromise between these two requirements, which ultimately results in restrictions on how spent fuel assemblies can be loaded in the cask.

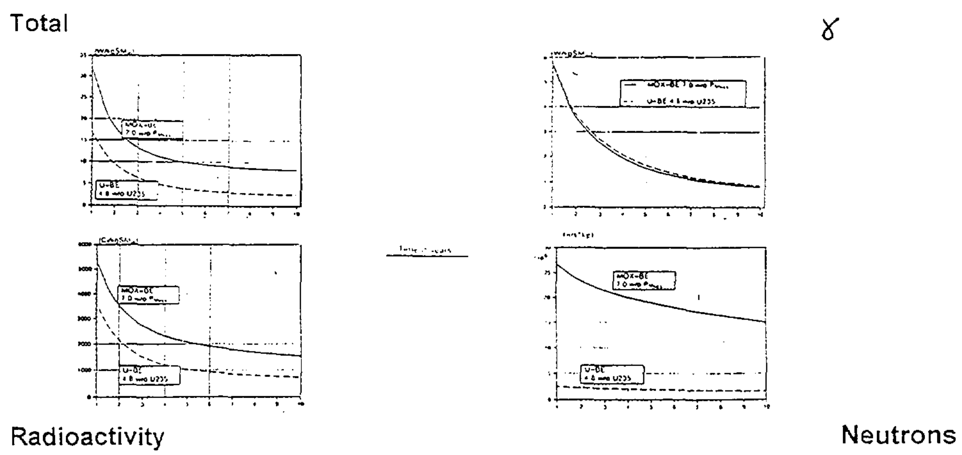



*FIG. 3. Thermal and nuclear source terms of spent LWR FA with 65 GW·d/tHM burnup*

### 2.3. Spent fuel rod integrity criteria in dry storage

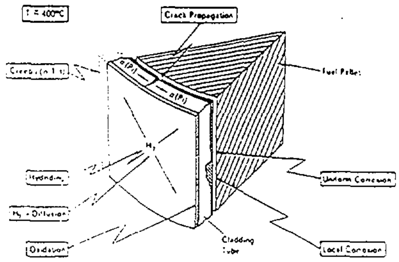
The integrity of the fuel rod cladding tubes is a crucial factor in interim dry spent fuel storage, and particularly their function as a barrier over the duration of interim storage. Figure 4 compiles all noteworthy fuel rod degradation mechanisms under dry storage conditions:

#### *2.3.1. Oxidative corrosion*

Oxidation of the Zircaloy (Zry) is a thermally-induced process. Dry storage under inert gas conditions leads to no further increase in the oxide layer over and above the condition upon final discharge from the reactor, since the storage conditions rule out the presence of oxidising substances. Limited access of air can be assumed if the sealing system of the dry storage system had failed as an off-normal event. If the spent nuclear fuel stored is free of defects no alteration of the $UO_2$-fuel will occur since the Zry-cladding can easily withstand the oxidative corrosion at the temperature level to be expected in the storage mode II and mode III period. There is also experimental evidence that oxidation will not defect the cladding for temperatures as high as the primary spent nuclear fuel insertion temperatures licensed when first loading the system after the wet storage period (storage mode I) . In the temperature range of 400°C to 200°C the rate of cladding and fuel oxidation decreases by an order of magnitude if the temperature drops about 10% [5]. If there is a defective fuel rod in the storage system and the temperature is less than 200°C (mode III) still no fission product release is expected, since the $UO_2$-fuel will practically not oxidise. If the fuel does not oxidise no restructuring takes place and hence no fission product release need be expected. If the temperature exceeds 200°C (mode I and mode II), $UO_2$ will be converted to $U_3O_8$ predominantly and Kr will be released [6].

1. Source terms
- n- and $\gamma$- radiation are important for shielding considerations
- decay heat defines the temperature for a given confinement, especially the hot spot cladding temperature

2. Fuel rod cladding integrity
- provides a reliable first barrier for the fission product retention




Mechanisms affecting spent fuel cladding performance during dry storage

*FIG. 4. Source terms of spent FA and spent FA integrity are essential criteria for dry storage*

*2.3.2. Fission-product-induced cladding corrosion*

Fission-product-induced stress corrosion cracking (crack propagation) occurs only within a particular temperature range in the presence of chemically-active iodine and adequate stresses. In dry spent fuel storage, fission products are not present in a chemical form which could trigger any kind of corrosion, and also the stress conditions required for the occurrence of stress corrosion cracking are absent.

Why fission products are stabilised in the $UO_2$-lattice? $UO_2$-fuel crystallises in a lattice from $CaF_2$-type and contains a plurality of vacancies which aid in the retention of fission products generated during the burnup. Additionally it must be considered that each fissioned U-atom leaves a further U-vacancy. Thus even in a fuel with a higher burnup not all vacancies will be occupied by fission products indicating that the $UO_2$ crystal provides excellent capabilities to retain the fission products.

Why the fission product chemistry stabilises the fission products in the $UO_2$-lattice? The thermodynamic assessment in the system „U-O-fission products" provides a further argument for the stability of fission products in the $UO_2$ fuel (Fig. 5). The fission product isotopes from Ce and Zr occur in the tetravalent state and can therefore easily replace a fissioned tetravalent U. Y and the lanthanides may occur in a trivalent state and need - if positioned on a U side - another neighbouring atom in the monovalent state. The Pt metals are stable in the non-ionised metallic state. Since the O potential in a fuel rod is high enough to oxidise Zr and to keep the $UO_2$-fuel in an almost stoichiometric, the Cs released from the fuel may occur in the form of a Zirconate. Mo as a fission product has a nearly identical free energy function as $UO_2$-fuel. Establishing a valancy balance for the fissioned $UO_2$-atoms and their related fission products without Mo results in a slight surplus of negative valences. The Mo will therefore balance the system by occurring partly in the metallic state and partly in oxidised form.

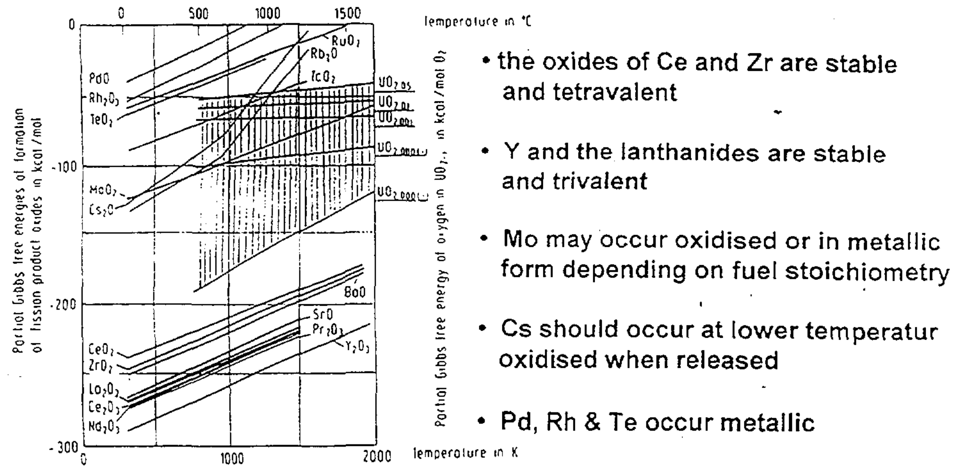



*FIG. 5. Gibbs free energies of formation of FP oxides and partial Gibbs free energy of the O in $UO_2$ of various stoichiometries*

Why the fission product atomic radii stabilise the fission products in the $UO_2$-lattice? Another interesting aspect for the stability of the fission products in the $UO_2$-crystal results from the comparison of the atomic and ionic radii of the tetravalent U with those of the fission products in their characteristic chemical state. The ionic radii of La, the lanthanides and of the tetravalent Ce are practically identical to those of the tetravalent U. Most of the other fission products including the noble metals lie a +/- 30% scatter around U (Fig. 6). The incorporation of such atoms into vacant sites of the $UO_2$-crystal is energetically favourable. The atomic radii of the noble gases as well as those of the Rb-, Cs and I-atoms however are exceeding largely those of the tetravalent U ions. The situation is even worse for the Cs and Rb in ionic form. Therefore it must be concluded that the noble gases, the earth-alkali-metals and iodine will tend to be released from the $UO_2$-crystal if the relevant kinetic processes will allow.

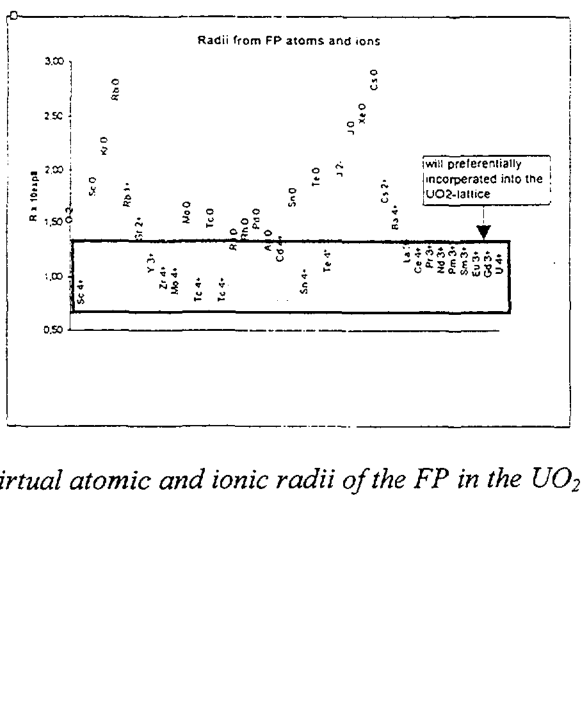



*FIG. 6. Virtual atomic and ionic radii of the FP in the $UO_2$ lattice*

Why the are the fission products in the $UO_2$-lattice so stable? The release of meta-stable implanted fission products occurs by more or less complicated diffusion processes. Since the diffusion under spent nuclear fuel dry storage conditions is a thermally activated process the temperature distribution in the fuel will be of great importance. A large data base and extensive knowledge is available to describe the release of the fission gases from the fuel under in-service conditions during reactor operation. Typically, the fuel temperature during reactor operation on its surface is around 400°C and in its centreline between 1,300°C and 1,800°C depending on the heat rating and burnup. For temperatures above 1,000°C, the diffusion coefficients are only dependent on the temperature. Those coefficients decrease between 1,800°C and 1,000°C from $10^{-12}$ $cm^2/s$ to $10^{-16}$ $cm^2/s$ by 4 orders of magnitude. Between 1,000°C and 400°C the pure thermal diffusion is combined in-pile with a radiation supported term only decreasing by 2 orders of magnitude. Since in storage the radiation level is negligible in comparison to in-pile conditions, only the pure thermal diffusion will contribute to fission gas release. Therefore, the diffusion coefficient at storage temperature is at least 8 orders of magnitude less than compared with those at fuel centreline temperature, or 5 orders of magnitude less than that of the fuel average temperature. Therefore, it can be concluded that under dry storage conditions for fuel temperatures less than 400°C, no fission gas will be released at all even under extended storage periods. Investigation of the Cs and I release from irradiated fuel has shown [7], that the release is diffusion controlled and the diffusion coefficients of those fission product isotopes are very similar to those of the fission gases. It was found that below a specific temperature there was no release at all. This temperature decreases with increasing burnup and will be above 800°C and 900°C (Fig. 7). This corresponds to the findings for the fission gas release as discussed above. Therefore, it can be concluded that neither Cs nor I will be released under dry storage conditions from the fuel to the gap of a spent LWR fuel rod.

In summary it can be stated that the fission products generated in the $UO_2$ fuel under in-service conditions are practically immobile in the $UO_2$ fuel lattice during storage. Consequently all fission-product-driven defect mechanisms such as:

- stress corrosion cracking (SCC);
- uniform fuel rod internal fission product corrosion of the cladding;
- localised fuel rod internal fission product corrosion of the cladding will not take place.

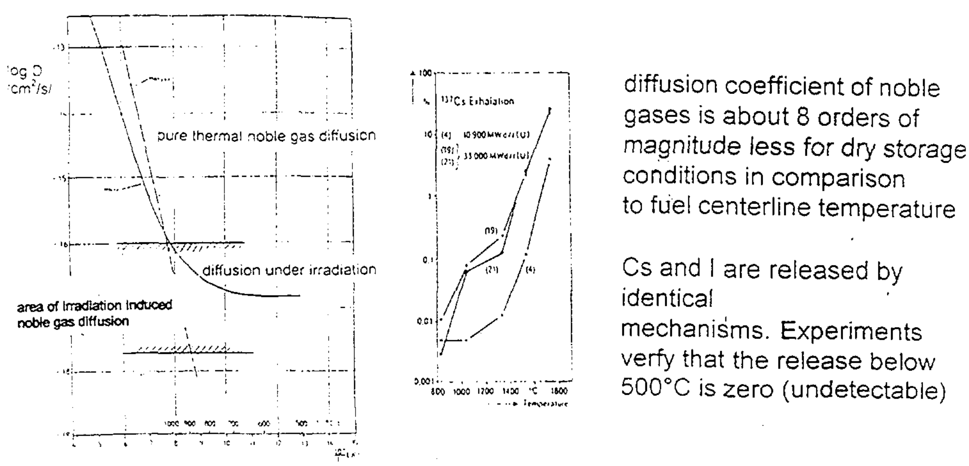



*FIG. 7. Thermal release of noble gases, Cs and I*

## *2.3.3. Delayed hydrogen cracking*

If the hydrogen content of a Zry-cladding exceeds the solubility - which is 160 ppm at 400°C - the hydrogen is precipitated in form of Zr-hydride platelets. Fuel rods with a burnup exceeding 50 GW·d/tHM may have a greater hydrogen content. The hydrogen precipitates reduce the ductility of the cladding if orientated orthogonal to the applied stress.

**Crack initiation**: Incipient cracks at the inner cladding surface may initiate delayed hydrogen cracking if hydrogen hydrogen precipitates at the crack tip decreasing the critical stress intensity factor. Radially-orientated hydrogen precipitates may serve as a incipient crack only if the temperature of the cladding is less than 160°C which represents the ductile-brittle transition for the precipitated Zr-hydride platelets.

**Hydride re-orientation**: The cladding tube manufacturing process generates a texture in the cladding causing the hydride-platelets to precipitate mostly tangentially. Re-orientation of hydride platelets may occur if the texture of the cladding changes under specific stress conditions and hydrogen is precipitated afterwards. At the grain sizes typical for LWR cladding those stress conditions are [8]: $\delta$ = 400°C ; $\sigma$ = 120 -180 N/mm$^2$ and $\delta$ = 250°C ; $\sigma$ = 250 - 350 N/mm$^2$ . Those stresses in the cladding are precluded under dry storage conditions in Germany by limitations in the storage license.

**Critical crack size**: Based on Canadian results [9,10], the critical crack size can be calculated by the following equation:

$A = C \times ( K_{IH} / \sigma)^2$ ............/ m/

with C=0,4

resulting in the data in Table I.

TABLE I. CRITICAL CRACK DEPTH FOR DELAYED HYDROGEN CRACKING

| T /°C/ | $K_{IH}$/ MNm$^{-3/2}$/ | $\sigma$/ MN/m$^2$/ | Critical crack depth |
|---|---|---|---|
| 150 | 10 (unirradiated) | 120 | 695 μm |
| 150 | 5 ( irradiated) | 120 | 348 μm |

The stress intensity factor for the unirradiated material results from experimental investigations. The value for the irradiated material status is assumed to be only half of that from unirradiated material. As a result the critical crack depth is 348 μm (Fig. 8). Neither incipient cracks from such depth nor re-oriented hydride-platelets had been observed in spent fuel in the EOL condition or after dry storage. In Germany the maximum hoop stress in the cladding during storage is limited by the licenses to 120 MN/m$^2$, the likelihood that hydride re-orientation might occur during storage can therefore be excluded. Altogether it can be concluded, that delayed hydrogen cracking will not occur in dry storage, even if the hydrogen contents rises in the cladding with the burnup. This conclusion is in agreement with the experience that this defect mechanism has never been observed world-wide.

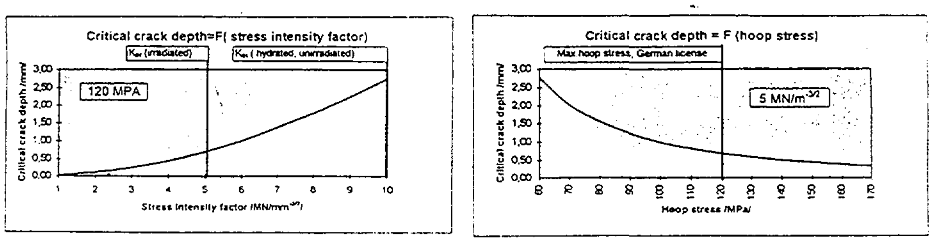



*FIG. 8. Critical crack depth in relation to stress intensity factor and hoop stress*

*2.3.4. Creep*

Creep is the enveloping criterion for consideration of cladding integrity during storage. At the temperatures of between 300 and 400°C (storage mode I), which prevail at the start of dry storage, the cladding undergoes strain. Its numerical value is largely determined by the fuel rod internal pressure and the temperature-time history in the course of dry storage. At maximum temperatures of less than 400°C, a total cladding strain of approximately 2 to 3% has no negative effect on cladding integrity, and is therefore used as a basis for current licenses [11,12].With higher burnup the dry storage temperature increases due to the higher decay heat from fuel assemblies. The also increased fission gas release results in a higher internal gas pressure within a spent fuel rod. This will generate higher stresses and strain in the fuel rod cladding. Since the longer residence time of a fuel assembly in the core to achieve the higher burnup tends to reduce the residual wall thickness through in-reactor corrosion, stress and strain is furthermore increased. MOX fuel exhibits, already at the same burnup as U fuel, increased decay heat and fuel rod internal EOL-pressure. MOX fuel with increased burnup will therefore have the highest stress and strain while in dry storage.

## 3. PREDICTION OF SPENT LWR FUEL ASSEMBLY DRY STORAGE PERFORMANCE

### 3.1. Method and procedure

The results from Section 2.3.4 show that cladding creep from inner gas over pressure is the rate determining degradation and the failure mechanism for setting maximum allowable storage temperature limits in dry inert storage. If creep were allowed to proceed to rupture, the fracture mode would most likely be of a pinhole type. Nevertheless, it is desirable to avoid any degradation. This can be accomplished by confining the creep degradation mechanism to its primary and early secondary stages. If it can be shown that the creep strain never exceeds that critical strain domain during inert dry storage, the tertiary creep with its subsequent fuel rod defection can be excluded. This approach is in compliance with standard creep engineering practices. This methodology relies on the availability of:

- a date base of cladding creep test results under internal pressurisation conditions [13]. The data base was published for the traditional type of Zry cladding at the spent fuel storage symposium in Seattle in the year 1985 [14] and on the occasion of the SMIRT-conference in Brussels in the same year [15]. The data on cladding especially developed for higher burnup fuel was published recently[19] (Fig. 9);
- a correlation which allows the prediction of post-pile creep from creep of unirradiated material. As could be shown by comparison of the creep of unirradiated and irradiated Zry the creep for unirradiated material always describes the post-pile creep conservatively (Fig. 10);
- a date base of the potential total strain of fuel rod cladding under dry storage conditions. For a burnup less than 40 GW·d/tHM the data from the post-pile burst test indicate that all cladding will reach, at minimum, 1% uniform strain before tertiary creep starts (Fig. 11). Since the straining capability decreases with burnup the very conservative measurement of the allowable strain by the burst test with its very high strain rates was replaced by a creep burst test with much lower strain rates to verify that Zry will have more than 1% strain under dry storage conditions. Siemens performed such a test programme under contract to GNB providing the spent fuel from a Siemens high performance programme having a burnup up to 64 GW·d/tHM. W. Sowa et al. reported from that programme that its result provided the basis for the recent German licence to store spent LWR fuel with increased burnup allowing 1% strain for U FA and MOX FA with an average batch burnup as high as 55 GW·d/tHM [16];
- a numerical model to predict creep strain during dry storage. This model was first published in Seattle [17,18] and was revised for cladding developed specially for increasing burnup in 1997 [19].

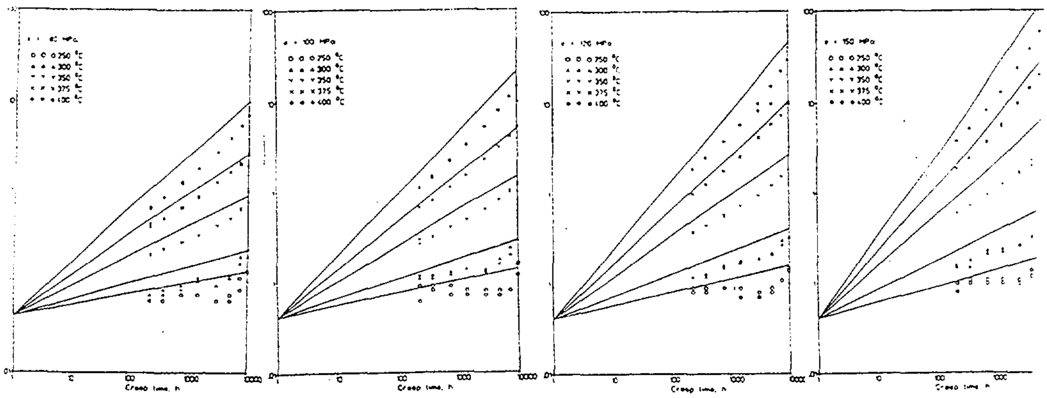


*FIG. 9. Experimental creep data and calculated values using the creep equation derived from the experimental data base*

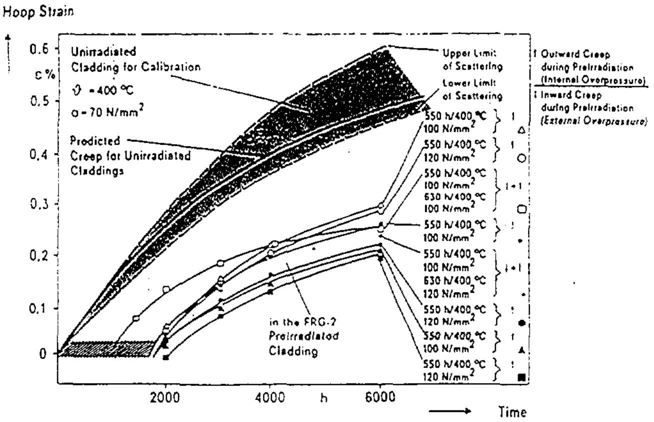



*FIG. 10. Comparison of creep strain of unirradiated and irradiated Zry-4 cladding*

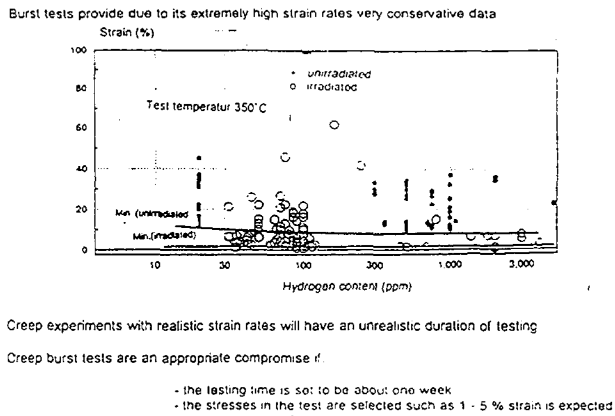



*FIG. 11. Burst strain data of Zry cladding in relation to its H content*

### 3.2. Dry storage predictions for spent fuel with increased burnup

As an example, a CASTOR V/19 is considered loaded with:

- 19 $UO_2$ LWR FAs having a maximum rod burnup of 55 GW·d/tHM; or
- 15 $UO_2$ LWR FAs **and** 4 MOX LWR FAs both having a maximum rod burnup of 55 GW·d/tHM.

Additionally, 2 types of fuel rod cladding is looked at to assess the influence of different cladding material. Table II shows the scope of the assessment.

Table II. SCOPE OF STRAIN CALCULATION FOR CASKS LOADED WITH SPENT LWR FA

| Case | 1 | 2 | 3 | 4 | units |
|---|---|---|---|---|---|
| Cladding | fast creeping clad | | less fast creeping clad | | |
| Burnup (rod) | 55 | 55 | 55 | 55 | GW·d/tHM |
| Fuel | $UO_2$ | $UO_2/PuO_2$ | $UO_2$ | $UO_2/PuO_2$ | |
| Pool storage | 5.6 | 6.5 | 5.6 | 6.5 | years |
| Decay heat/cask | 32.3 | 37.5 | 32.3 | 37.5 | kW at start of storage |
| FR hot spot temp. | 348 | 357 | 348 | 357 | °C |
| Storage period | 40 | 40 | 40 | 40 | years |

Siemens-PCA-2 is an example of a relatively fast creeping cladding, whereas the Siemens-DUPLEX cladding represents a relatively strong cladding. In calculating the end-of-life conditions for reactor operation, the different in-service behaviour of both cladding types has been considered. Fast creeping cladding contacts, due to the creep down under the external coolant, pressure the oxide fuel earlier than strong cladding. As a result of the earlier closing fuel cladding gap, the fuel temperature and consequently the fission gas release is less than for the stronger cladding. Therefore, at the start of dry storage the internal fuel rod pressure in the cases 1 and 2 is less than for cases 3 and 4. The starting conditions and the hoop strain after 40 years for dry storage of the 4 considered cases are compiled in the Tables III and IV respectively. The results of the calculation for case 1 through 4 are given in Figs. 12 through 13.

TABLE III. BEGIN OF DRY STORAGE CONDITIONS FOR CASE 1 THROUGH 4

| type of cladding | lower temperature | higher temperature |
|---|---|---|
| fast creeping cladding | 348°C<br>$\sigma_{hoop} = 60$ N/mm$^2$<br>case 1 | 357°C<br>$\sigma_{hoop} = 60$ N/mm$^2$<br>case 2 |
| less fast (stronger) creeping cladding | 348°C<br>$\sigma_{hoop} = 80$ N/mm$^2$<br>case 3 | 357°C<br>$\sigma_{hoop} = 80$ N/mm$^2$<br>case 4 |

TABLE IV. HOOP STRAIN AFTER 40 YEARS DRY STORAGE FOR CASE 1 THROUGH 4

| Type of cladding | lower temperature | higher temperature |
|---|---|---|
| fast creeping cladding | 0.37% | 0.51% |
| stronger cladding | 0.52% | 0.77% |

Since it is generally accepted that spent Zry-type cladding can withstand, under dry storage conditions, at least 1% hoop strain, the results presented indicate that under the considered conditions spent LWR fuel up to an average fuel rod burnup of 55 GW·d/tHM can be stored free of systematic defects for periods up to 40 years. Since follow-on storage up to 100 years will be mostly within the mode 3 an extension of the storage period will not result in defects in such spent LWR fuel.

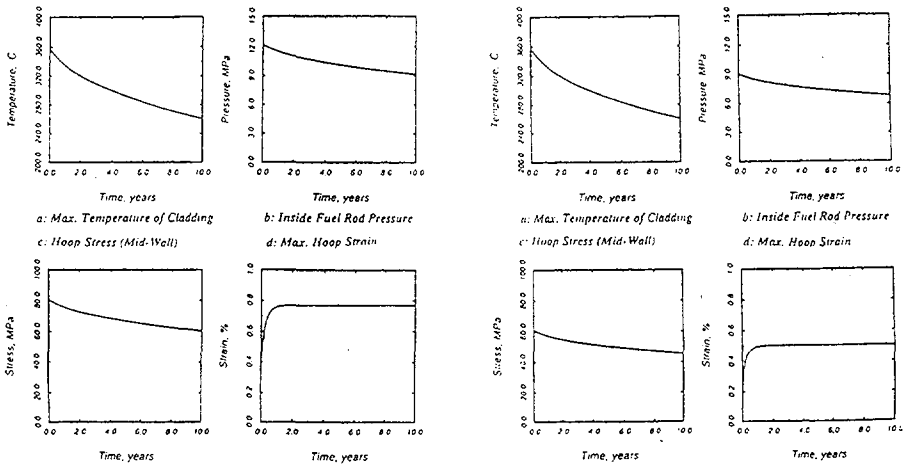



*FIG. 12. Spent PWR MOX FA, predicted strain in dry storage, 55 GW·d/tHM (left strong cladding, right fast creeping cladding)*

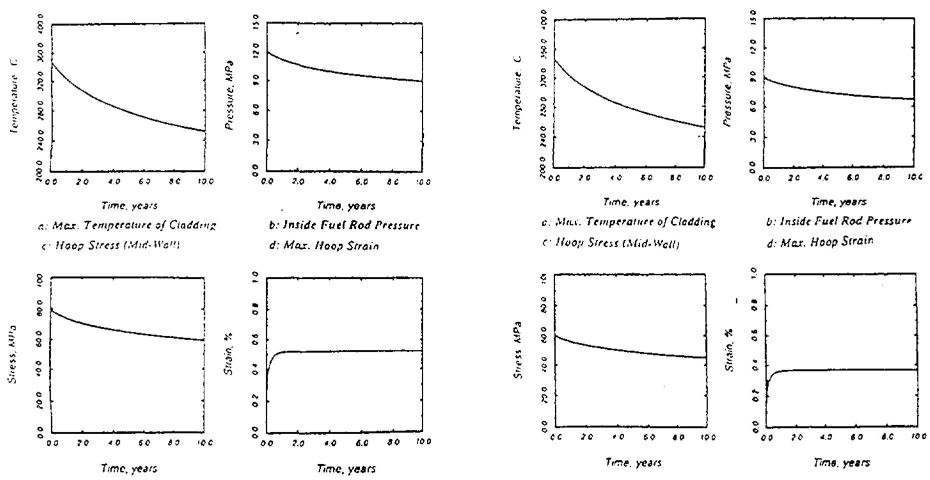



*FIG. 13. Spent PWR $UO_2$ FA, predicted strain in dry storage, 55 GW·d/tHM (left strong cladding, right fast creeping cladding)*

## 4. CONCLUSIONS

The leading defect mechanism for spent fuel rods in dry storage - also for fuel rods with increased burnup - remains creep due to the hoop strain resulting from the fuel rod internal fission gas pressure. Limiting the creep to its primary and secondary stages prevents fuel rod degradation. Post-pile creep of fuel rod cladding can be described conservatively by the creep of unirradiated cladding. The allowable uniform strain of the cladding in its typical post-pile condition preventing tertiary creep under dry spent fuel storage conditions is 1 -2%. Calculation of the predicted dry storage performance of fuel assemblies with a burnup $\leq$55 GW·d/tHM was based on the fuel assemblies' end-of-life data and on a representative curve T = f(t). The maximum hot spot temperature assumed for a fuel rod in the CASTOR V cask was between 348°C (U FA) and 358°C (MOX FA) at the beginning of storage. The highest hoop strain predicted after 40 years of storage is 0.77% and 0.52% respectively. This result proves that dry storage is safe for LWR fuel of such burnup.

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# MEASUREMENTS OF DECAY HEAT AND GAMMA-RAY INTENSITY OF SPENT LWR FUEL ASSEMBLIES

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**Abstract**

Calorimetric measurements of the decay heat of a number of BWR and PWR fuel assemblies have been performed in the pools at the Swedish Central Interim Storage Facility for Spent Nuclear Fuel, CLAB. Gamma-ray measurements, using high-resolution gamma-ray spectroscopy (HRGS), have been carried out on the same fuel assemblies in order to test if it is possible to find a simple and accurate correlation between the $^{137}$Cs -intensity and the decay heat for fuel with a cooling time longer than 10-12 years. The results up to now are very promising and may ultimately lead to a qualified method for quick and accurate determination of the decay heat of old fuel by gamma-ray measurements. By means of the gamma spectrum the operator declared data on burnup, cooling time and initial enrichment can be verified as well. CLAB provides a unique opportunity in the world to follow up the decay heat of individual fuel assemblies during several decades to come. The results will be applicable for design and operation of facilities for wet and dry interim storage and subsequent encapsulation for final disposal of the fuel.

## 1. INTRODUCTION

The Swedish Nuclear Fuel and Waste Management Co, SKB, is conducting test measurements of decay heat and gamma radiation on spent fuel assemblies in the pools at the Swedish Central Interim Storage Facility for Spent Nuclear Fuel (CLAB). The decay heat is measured in a calorimeter and for the gamma-ray measurements high-resolution spectroscopy (HRGS) technique is used. The objectives of these measurements are to see if it is possible to:

PRIMARILY:
- Achieve accurate, quick and simple determination of decay heat by gamma-ray measurements on old fuel assemblies prior to encapsulation for final disposal;
- Achieve verification of burnup and cooling time of fuel prior to encapsulation for final disposal;
- Provide a basis for BU-credit (if needed) in final disposal canister.

SECONDARILY:
- More accurately predict the decay heat prior to fuel transport to CLAB;
- Provide a verification of decay heat calculation codes e.g. Origen and Decay (Swedish code) especially for fuel with long cooling times;
- More accurately predict the total decay heat in CLAB.

By autumn 1998, measurements and calculations on 14 BWR and 31 PWR assemblies with different nuclear data have been performed. Gamma-ray spectra have been obtained using the HRGS technique. In November-December 1998, measurements are planned to be performed on 50 BWR assemblies including the 14 assemblies already measured.

Measurements will, according to current plans, be repeated in certain intervals in the future on the same selected fuel assemblies. Assemblies of different designs may be added to the population of assemblies assigned for measurement. CLAB provides a unique opportunity in the world to follow up the decay heat of individual fuel assemblies during several decades to come. The results will be applicable for design and operation of facilities for wet and dry interim storage and subsequent encapsulation for final disposal of the fuel.

## 2. CALORIMETRIC MEASUREMENTS

### 2.1. Measurement principle

The principle of the method used for the calorimetric measurements in the pools is simple: the temperature raise of a limited and isolated heated volume of water is measured, until thermal equilibrium conditions between the volume and the surrounding pool water are established. The temperature difference at equilibrium, which is a measure of the power input, is recorded.

### 2.2. Equipment

The equipment consists of a box with double walls, see Figure 1, placed in one of the unloading pools in CLAB [1]. The box, which is open at the top and at the bottom is part of the system normally used for detecting leaking fuel. Insulation is provided by the 10 mm wide air-filled gap between the inner and outer walls of the box. There are two high quality resistor temperature sensors mounted inside the box. Another two sensors are placed outside the box in the pool.

The fuel assembly to be measured is placed in the box and a hood is lowered over the top of the box. The hood is filled with air and an air lock is created which prevents water from flowing through the box during the measurement although there is direct contact with the pool water at the bottom. Therefore, no overpressure can be built up inside the box.

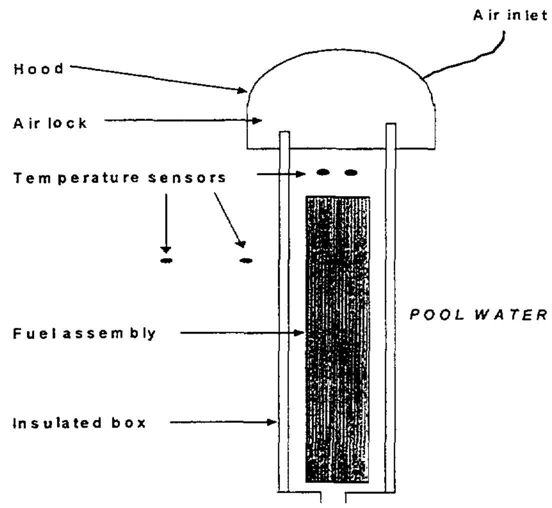



*FIG. 1. Schematic figure of the measurement equipment*

### 2.3. Calibration and correction for heat loss due to gamma radiation

In order to translate the equilibrium temperature difference between the box and pool into decay power, a calibration curve has been established. An immersion heater is used to heat the enclosed water. The heater consists of a skeleton of a fuel assembly and a heating cable, which is attached along the structure to simulate the fuel assembly geometry. Power is provided from a stabilised electrical supply and the power input is measured. Data including the temperature readings are automatically recorded.

There are separate boxes for BWR-fuel and PWR-fuel, and calibration curves for both cases have been established. The calibration measurements have also been used to assess the accuracy of the method.

A fraction of the decay heat from the fuel assembly is lost from the calorimetric measurement box with the gamma radiation that escapes without giving rise to temperature inside the system. Therefore, the measured decay heat values have to be corrected accordingly. Gamma intensity measurements in the close surroundings of the box have been carried out in order to assess this effect. The results are presented in Table I.

TABLE I. GAMMA LOSS

| Fuel type | Gamma loss % |
|---|---|
| 8*8(BWR) | 4,4 |
| 15*15(PWR) | 0,9 |
| 17*17(PWR) | 0,7 |

## 2.4. Measurements on active fuel

### *2.4.1. BWR*

14 BWR assemblies from Ringhals NPP were selected for the measurements. The operator declared burnup varied between 21 and 38 MW·d/kgU, and the decay time between 3.6 and 15.0 years.

In total 30 measurements have been carried out. Seven assemblies were measured twice, one assembly five times and one six times, with the two latter cases used for assessing the reproducibility of the measurements. The standard deviations for these cases were 51 W/tU and 49 W/tU respectively, corresponding to 3.3% and 3.2%. Three measurements were rejected because the airlock of the measurement box was lost, which could easily be seen on the temperature curves. The results of the measurements were corrected for the heat loss due to gamma radiation prior to comparison with the values calculated by means of computer codes (See Table II).

### *2.4.2. PWR*

31 PWR assemblies from Ringhals 2 and 3 NPPs were selected for measurement. The operator declared burnup varied between 19.7 and 51.0 MW·d/kgU, and the decay time between 5.9 and 16.0 years. The results of the measurements were corrected for the heat loss due to gamma radiation prior to comparison with the values calculated by means of computer codes (see Table III).

## 2.5. Calculations

### *2.5.1. Codes*

The decay heat for the actual assemblies has been calculated using Origen 2.1 for BWR and Origen–S for PWR. The Decay code has been used to calculate both BWR and PWR-assemblies. This is a simplified code, which has been developed for calculating the decay heat and is based on ISO standard 10645 first edition 1992-03-01.

### *2.5.2. Power history of the fuel assemblies*

The input data for the decay heat calculations are fuel data and irradiation history of each assembly. The number of cycles is correctly modeled, but the detailed power history within each cycle is not represented. Each assembly has been assigned a mean value for the entire cycle.

## BWR

Table II below compares the BWR-measurements and the calculations. The measured values are shown in column 1. The values in column 2 are corrected to account for the heat loss due to gamma radiation. Calculated values are shown in columns 3 and 5 and ratios between calculated values and corrected measured values are shown in columns 4 and 6. Figure 2 shows a comparison between the measured value and the calculated value by Origen 2.1. The calculations show good agreement with the measurements on the average. Origen 2.1 seems to underestimate the power at long decay times.

TABLE II. COMPARISON BETWEEN MEASURED AND CALCULATED VALUES FOR BWR-ASSEMBLIES

| | | 1 | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|---|---|
| Assembly no | Measure-ment no | Measured value (W/ton) | Corrected value (W/ton) | Calculatd Origen 2.1 (W/ton) | Ratio Calc/corr | Calculated Decay (W/ton) | Ratio Calc/corr |
| 582 | 1 | 652 | 680 | 636 | 0,93 | 750 | 1,10 |
| | 2 | 658 | 687 | 634 | 0,92 | 744 | 1,08 |
| 596 | 1 | 684 | 714 | 670 | 0,94 | 784 | 1,10 |
| | 2 | 692 | 722 | 669 | 0,93 | 779 | 1,08 |
| 710 | 1 | 703 | 734 | 692 | 0,94 | 835 | 1,14 |
| | 2 | 707 | 738 | 691 | 0,94 | 835 | 1,13 |
| 900 | 1 | 726 | 758 | 719 | 0,95 | 857 | 1,13 |
| | 2 | 712 | 743 | 717 | 0,96 | 851 | 1,15 |
| 1136 | 1 | 725 | 757 | 704 | 0,93 | 839 | 1,11 |
| | 2 | 715 | 747 | 702 | 0,94 | 839 | 1,12 |
| 6423 | 1 | 1433 | 1496 | 1511 | 1,01 | 1525 | 1,02 |
| 6431 | 1 | 1509 | 1575 | 1730 | 1,10 | 1774 | 1,13 |
| 6432 | 1 | 1452 | 1516 | 1588 | 1,05 | 1605 | 1,06 |
| | 2 | 1430 | 1493 | 1575 | 1,05 | 1594 | 1,07 |
| | 3 | 1479 | 1545 | 1524 | 0,99 | 1555 | 1,01 |
| | 4 | 1474 | 1539 | 1523 | 0,99 | 1555 | 1,01 |
| | 5 | 1466 | 1531 | 1523 | 0,99 | 1555 | 1,02 |
| 6454 | (1 | 1135 | 1185 | 1488 | 1,26 | 1497 | 1.26)* |
| | 2 | 1396 | 1457 | 1480 | 1,02 | 1491 | 1,02 |
| | 3 | 1451 | 1514 | 1443 | 0,95 | 1469 | 0,97 |
| | 4 | 1424 | 1487 | 1441 | 0,97 | 1469 | 0,99 |
| 6478 | 1 | 1342 | 1401 | 1366 | 0,98 | 1413 | 1,01 |
| 8327 | 1 | 2074 | 2165 | 2239 | 1,03 | 2230 | 1,03 |
| 8332 | (1 | 1138 | 1188 | 1462 | 1,23 | 1532 | 1.29)* |
| | 2 | 1360 | 1420 | 1447 | 1,02 | 1515 | 1,07 |
| 8338 | 1 | 1352 | 1411 | 1452 | 1,03 | 1520 | 1,08 |
| | 2 | 1344 | 1403 | 1440 | 1,03 | 1509 | 1,08 |
| **Average +/- one standard deviation** | | | | | **0.98 +/-0.05** | | **1.07 +/-0.05** |

*These measurements are not included in the evaluations due to irregularities observed during measurement

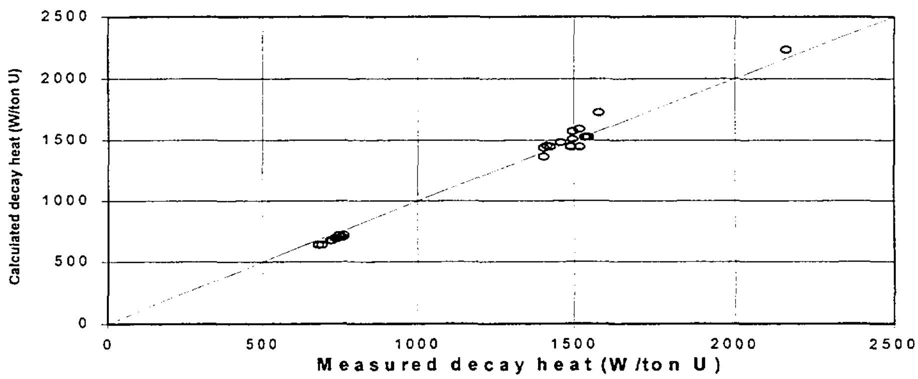



*FIG. 2. Comparison between calculated (Origen 2.1) and measured decay heat*

## PWR

Table III compares the measurements and the calculations on the 31 PWR assemblies[4]. The value in column 2 is corrected to account for the heat loss due to gamma radiation. Calculated values are shown in columns 3, 5 and 7 and ratios between calculated values and corrected measured values are shown in columns 4, 6 and 8.

TABLE III. COMPARISON BETWEEN MEASURED AND CALCULATED VALUES FOR PWR-ASSEMBLIES

| 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 |
|---|---|---|---|---|---|---|---|
| **Assembly** | **Measured** | **Calculated values** | | | | | |
| **no** | **decay heat (W)** | **DECAY (W)** | **Ratio Calc/meas.** | **ORIGEN-S (W)** | **Ratio Calc/meas.** | **Siemens (W)** | **Ratio Calc/meas.** |
| 0C9 | 657 | 651 | 0,99 | 618 | 0,94 | - | - |
| 0E2 | 798 | 769 | 0,96 | 753 | 0,94 | - | - |
| 0E6 | 644 | 681 | 1,06 | 637 | 0,99 | - | - |
| 1C2 | 532 | 542 | 1,02 | 520 | 0,98 | - | - |
| 1C5 | 636 | 651 | 1,02 | 623 | 0,98 | - | - |
| 1E5 | 620 | 654 | 1,06 | 610 | 0,98 | - | - |
| 2A5 | 308 | 341 | 1,11 | 285 | 0,93 | - | - |
| 2C2 | 595 | 626 | 1,05 | 587 | 0,99 | - | - |
| 3C1 | 642 | 616 | 0,96 | 584 | 0,91 | - | - |
| 3C4 | 654 | 651 | 1,00 | 620 | 0,95 | - | - |
| 3C5 | 686 | 651 | 0,95 | 619 | 0,90 | - | - |
| 3C9 | 601 | 617 | 1,03 | 582 | 0,97 | - | - |
| 4C4 | 536 | 554 | 1,03 | 520 | 0,97 | - | - |
| 4C7 | 666 | 650 | 0,98 | 619 | 0,93 | - | - |
| 5A3 | 294 | 332 | 1,13 | 279 | 0,95 | - | - |
| 5F2 | 1097 | 1026 | 0,94 | 1087 | 0,99 | - | - |
| C01 | 502 | 535 | 1,07 | 500 | 1,00 | - | - |
| C12 | 508 | 529 | 1,04 | 493 | 0,97 | - | - |
| D27 | 589 | 583 | 0,99 | 545 | 0,93 | 511 | 0,87 |
| D38 | 549 | 562 | 1,02 | 532 | 0,97 | 498 | 0,91 |
| E38 | 432 | 489 | 1,13 | 449 | 1,04 | 420 | 0,97 |
| E40 | 460 | 497 | 1,08 | 456 | 0,99 | 417 | 0,91 |
| F14 | 466 | 504 | 1,08 | 463 | 0,99 | 434 | 0,93 |
| F21 | 516 | 549 | 1,06 | 509 | 0,99 | 436 | 0,85 |
| F25 | 486 | 523 | 1,08 | 479 | 0,99 | 449 | 0,92 |
| F32 | 917 | 833 | 0,91 | 898 | 0,98 | 848 | 0,92 |
| G11 | 516 | 556 | 1,08 | 503 | 0,98 | 471 | 0,91 |
| G23 | 524 | 564 | 1,08 | 513 | 0,98 | 481 | 0,92 |
| I09 | 664 | 684 | 1,03 | 666 | 1,00 | 622 | 0,94 |
| I20 | 507 | 544 | 1,07 | 498 | 0,98 | 469 | 0,92 |
| I24 | 515 | 545 | 1,06 | 499 | 0,97 | 467 | 0,91 |
| **Average +/-one standard deviatio** | | | **1.03 +/-0.06** | | **0.97 +/-0.03** | | **0.91 +/- 0.0** |

The calculations show good agreement with the measurements on the average. Origen-S seems to underestimate the power with 3 % on average. The Siemens value is a preliminary independent calculation with the Korigen code for verification purpose and similar calculations are planned for the other Siemens assemblies[5].

Figure 3 shows a comparison between the measured value and the calculated value by Origen-S.

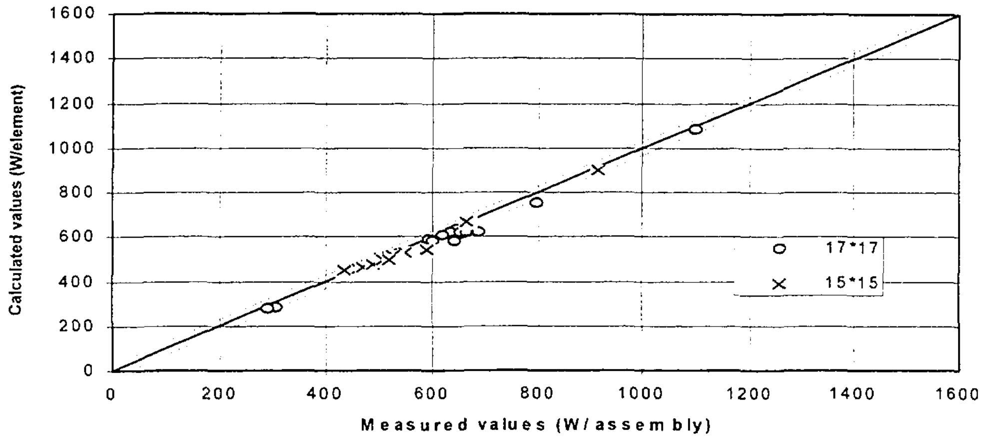



*FIG. 3. Comparison between Origen-S calculations and measured values*

## 3. GAMMA-RAY MEASUREMENTS

### 3.1. Principle of the method

Assuming that the gamma intensity (I) from the $^{137}$Cs activity of a spent fuel assembly can be measured, and that it is proportional to the concentration of the $^{137}$Cs activity in the whole, or part of the fuel studied, one may write the decay heat $P_{137}$ generated by the $^{137}$Cs activity [2]:

$$P_{137} = CI \tag{1}$$

The constant C depends on the various geometry conditions of the gamma intensity measurement array, and has to be determined in a calibration procedure. In practice, the experimental geometry can be expected to be constant, but the value of C has to be determined for each type of fuel assembly. For a given type of reactor, however, the variation in C is not expected to be very large, since the external dimensions necessary have to be the same for all types of fuel and no large variations are expected in the fuel to moderator ratio. This will be verified in the continued measurement programme. The total decay heat P is obtained from eq. (1) by introducing the ratio f:

$$f = \frac{P_{137}}{P} \tag{2}$$

This leads to:

$$P = C\frac{I}{f} \tag{3}$$

The ratio f, or f factor, depends in a complicated way on fuel parameters like burnup (BU), cooling time (CT), initial enrichment (ε) and power history. To apply eq. (3), these parameters must be known and the f factor must be calculated for each fuel assembly by using a computer code, e.g. Origen-2. As an example, the f factor versus cooling time is shown in Figure 4. Note the important fact, that for cooling times over 10-12 years the variation of the f factor is almost linear. The same applies for variations of burnup and initial enrichment.

This procedure may cast some doubt on the usefulness of eq. (3), since the decay heat may be obtained directly from the Origen-2 calculation. In practice, however, it turns out that the f factor is essentially constant over large intervals of the fuel parameters mentioned above. As an example, we show in Table IV the variation in f as various fuel parameters are changed by the stated amount. It should be noted that the calculated f factor corresponding to the fuel parameters shown in Table IV is 0.30.

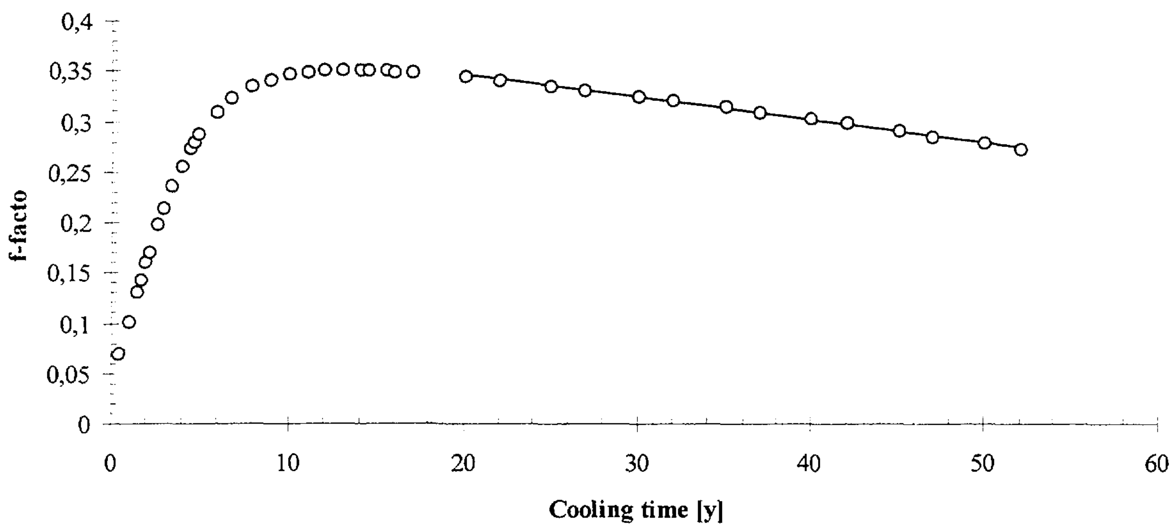



*FIG. 4. The f factor as function of cooling time*

TABLE IV. THE RELATIVE VARIATION OF THE F FACTOR (F = 0.30) AS THE BURNUP, COOLING TIME AND THE INITIAL ENRICHMENT ARE VARIED

| **Fuel parameter** | **Parameter interval** | **Relative variation in f** |
|---|---|---|
| Burnup = 30 GW·d/tU | ±5 GW·d/tU | 4 % |
| Cooling time = 40 y | ±5 y | 7 % |
| Initial enrichment = 2.0 % | ±0.5 per cent unit | 6 % |

In view of these small variations it is not necessary to determine the value of the f factor for each assembly from exact and time consuming calculations with e. g. the ORIGEN code. Sufficiently accurate values may be obtained by interpolation in a table of values of the f factor calculated for some standard values of the fuel parameters.

### 3.2. Equipment

The gamma-ray intensities were obtained by using the gamma-scanning facility at CLAB. A schematic lay-out of the equipment is shown in Figure 5. The fuel assembly to be measured is positioned in the fixture of the elevator located in front of the horizontal collimator. The speed of the elevator may be varied from 0 to about 7 cm/s in order to optimise it with respect to fuel length, scanning time etc.

The detector system is based on a germanium detector. The substantial size of the detector implies a large peak-to-Compton ratio, which is beneficial in order to accurately determine the peak areas of the gamma-ray spectra obtained. With this detector system counting rates of about 100,000 counts/s can be handled at a dead time around 50%. Due to the comparatively large dead time, an adequate correction for this has to be made.

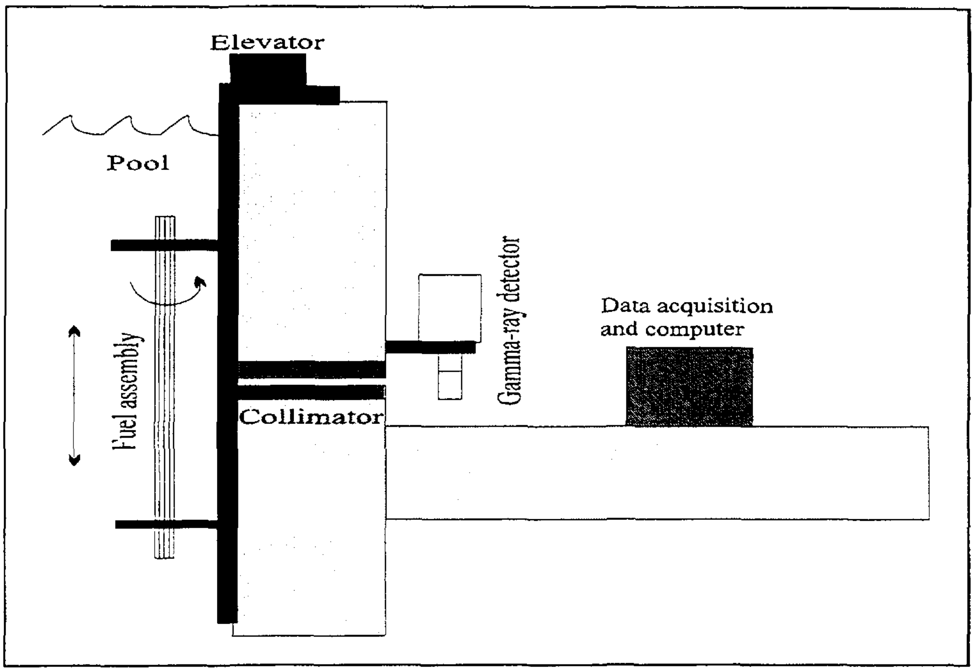



*FIG. 5. Schematic drawing of the gamma scanning facility at CLAB.*

### 3.3. HRGS Measurements

*3.3.1. Correlation between the $^{137}$Cs-intensity and decay heat*

**BWR**

The $^{137}$Cs -intensity for each fuel assembly was corrected for the time elapsed between the calorimetric and the gamma measurements. Due to the relatively short cooling times of the assemblies measured, the gamma-ray intensities were corrected by using exact f factors calculated with ORIGEN-2 and actual power histories (see discussion in section 3.1). By plotting these intensities versus the measured decay heat, Figure 6 was obtained. The line in Figure 6 is a least squares fit to the data corresponding to a slope coefficient of C=(12.08±0.10) cps/W. The individual standard deviation of the data points to the line corresponds to $\Delta P/P = 4.1\%$.

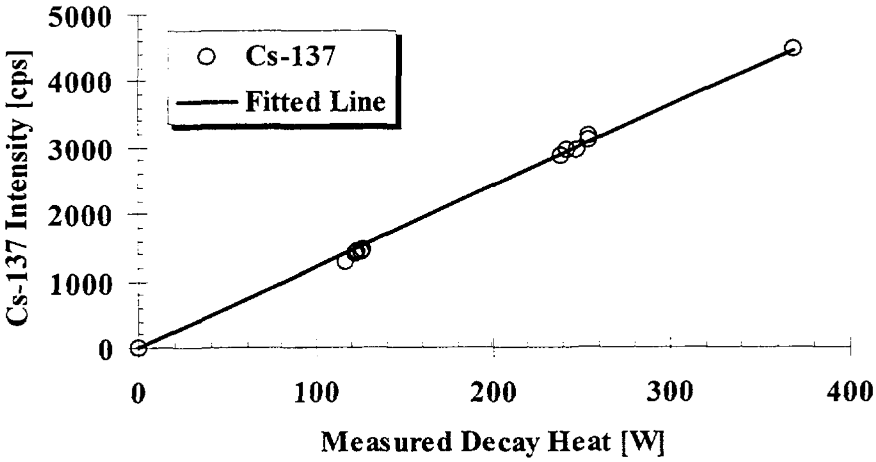


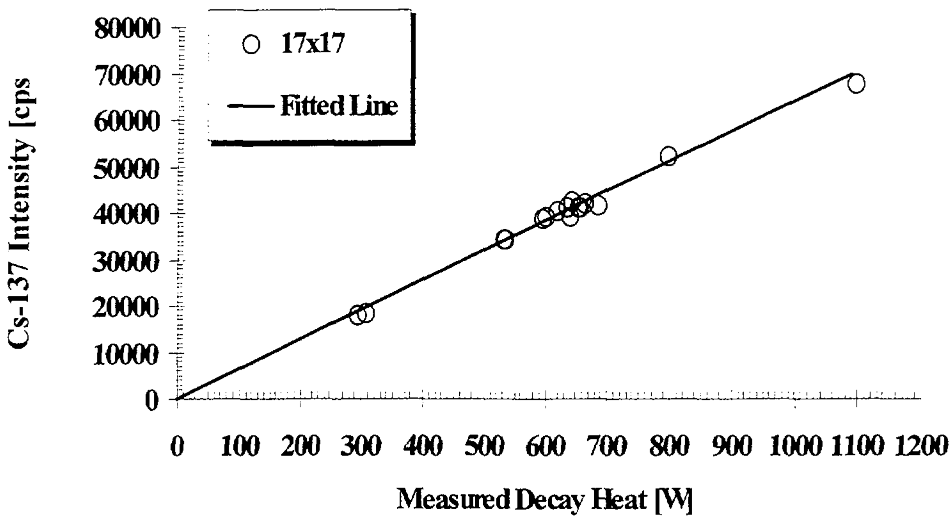

*FIG. 6. Measured decay heat as a function of measured $^{137}$Cs -intensities; The intensities have been corrected for the exact f factor.*

**PWR**

The generally longer cooling times for these assemblies motivated the simplified procedure to calculate the f factors as discussed in section 3.1. The f factors of each assembly were determined by using a function of the form $f = a \cdot BU + b \cdot CT + c$. This function was obtained by fitting a plane in the space spanned by burnup, cooling time and the corresponding f factors calculated by ORIGEN-S and using a limited range in burnup and cooling time and by using representative power histories [3].

The $^{137}Cs$ -intensity for each one of the 36 fuel assemblies was corrected for these f factors and, in addition, for the time elapsed between the calorimetric and the gamma measurements. By plotting the measured decay heat versus these intensities Figure 7 was obtained.

The slopes of the fitted lines in Figure 7 is (63.9 ± 0.5 and 66.7 ± 0.4) (cps/W) for 17×17 and 15×15, respectively. The relative standard deviation of P is 3.4 % and 2.5 %, respectively



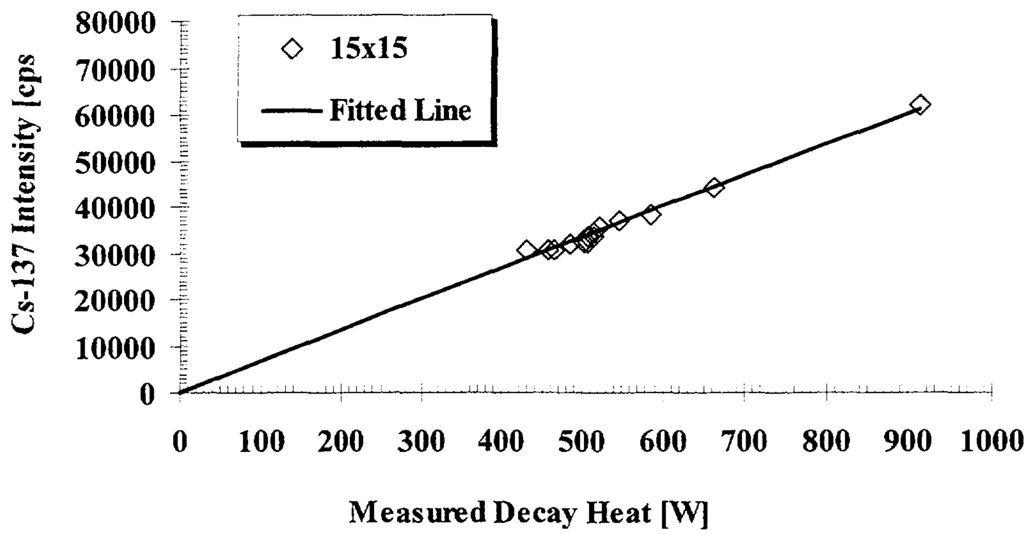




*FIG. 7. $^{137}Cs$-intensity vs. measured thermal power. The intensities are corrected using the actual power history of the fuel assemblies and for cooling time to the time of the calorimetric measurements. Decay heat is corrected for gamma losses*

*3.4.2. Verification of burnup*

The average intensity of the $^{137}Cs$ radiation obtained for each assembly was recorded. In order to check the quality of the gamma scans and to obtain a calibration constant k used for burnup verification, the data were fitted to the formula:

$$I = k \cdot BU \tag{4}$$

where

- I is the average gamma intensity of an assembly
- BU is the operators declared value of the burnup
- k is a least-squares fitted slope coefficient.

The fit was done separately for the 15x15 and the 17x17 assemblies. As shown in Figure 8 a good linear relationship was obtained for both types of assemblies with a standard deviation for individual assemblies of about 2%. The somewhat different geometry of the two types of assembly is seen to result in a slight but significant difference between the two values of the slope coefficient.

*3.4.3. Verification of cooling time*

The measured intensity of the 1,275 keV line from $^{154}Eu$ for the 36 assemblies is plotted as a function of the declared burnup in Figure 9 for 15x15 and 17x17 assemblies, respectively. For both sets of points a function:

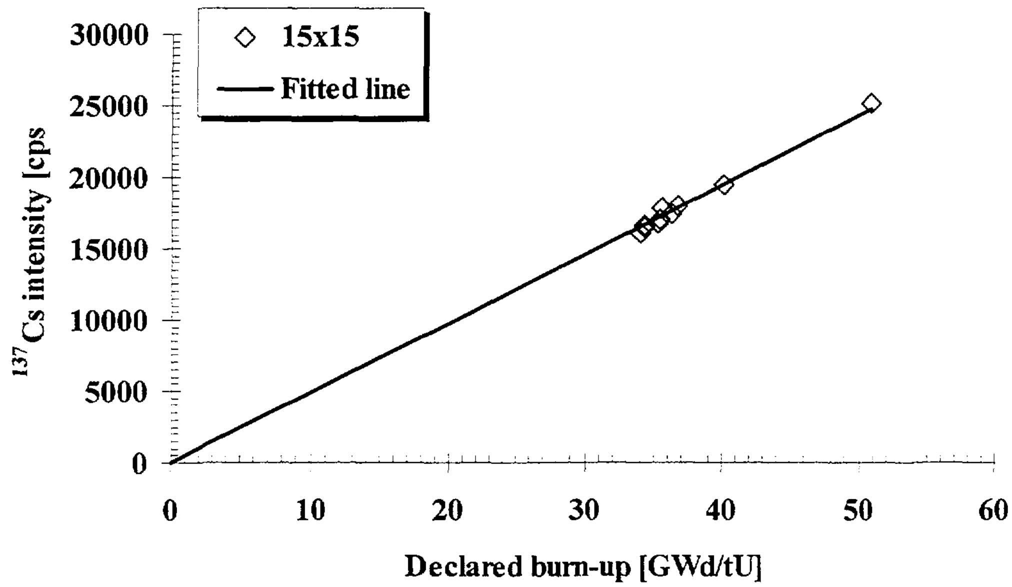
$$I = K \cdot BU_{\kappa} \tag{5}$$

was least square fitted.

The cooling time may be determined from the $^{137}Cs$ and $^{154}Eu$ intensities using the following equation:

$$CT = \frac{1}{\lambda_2 - \kappa\lambda_1} \cdot \ln\left\{\left(\frac{I_{Cs}}{k}\right)^{\kappa} \cdot \frac{K}{I_{Eu}}\right\}$$

The parameters k, K and $\kappa$ are those of the eqs. (4) and (5) and $\lambda_1$ and $\lambda_2$ are the decay constants of $^{137}Cs$ and $^{154}Eu$, respectively. The calculated cooling times are compared with the declared values in Figure 10. The cooling times agree well with standard deviations of 0.5 y and 0.3 y, for 15×15 and 17×17 fuel, respectively. One may expect even smaller standard deviations if corrections for power history are applied.



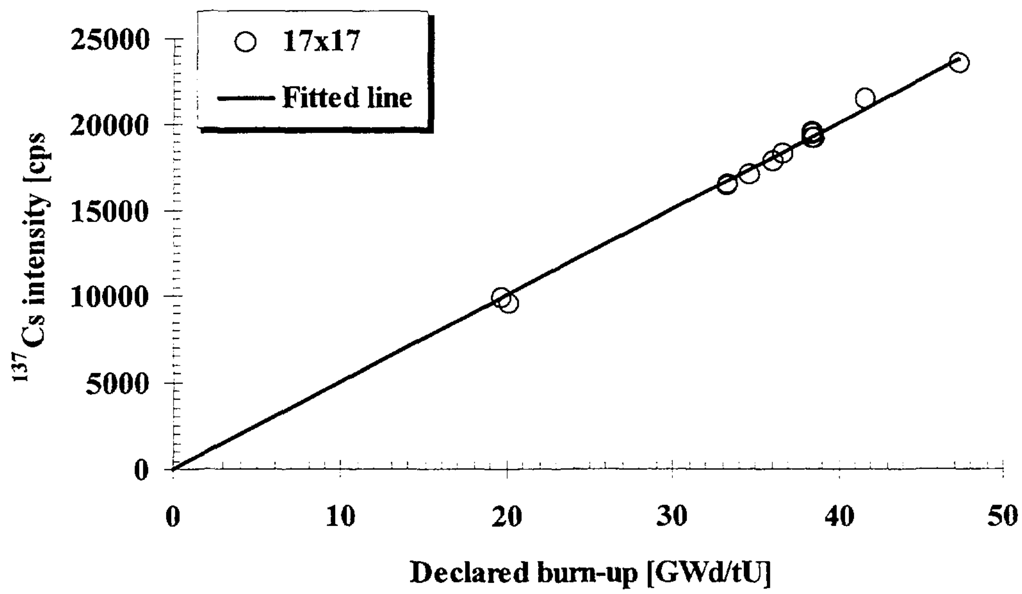
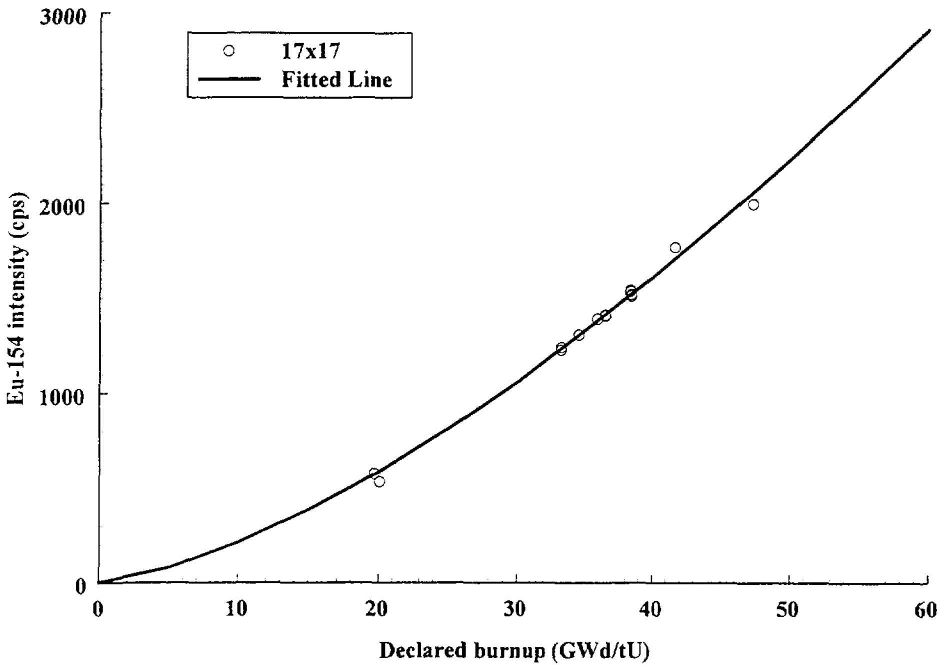




*FIG. 8.* $^{137}$Cs *intensity as function of the declared burnup for PWR-assemblies*



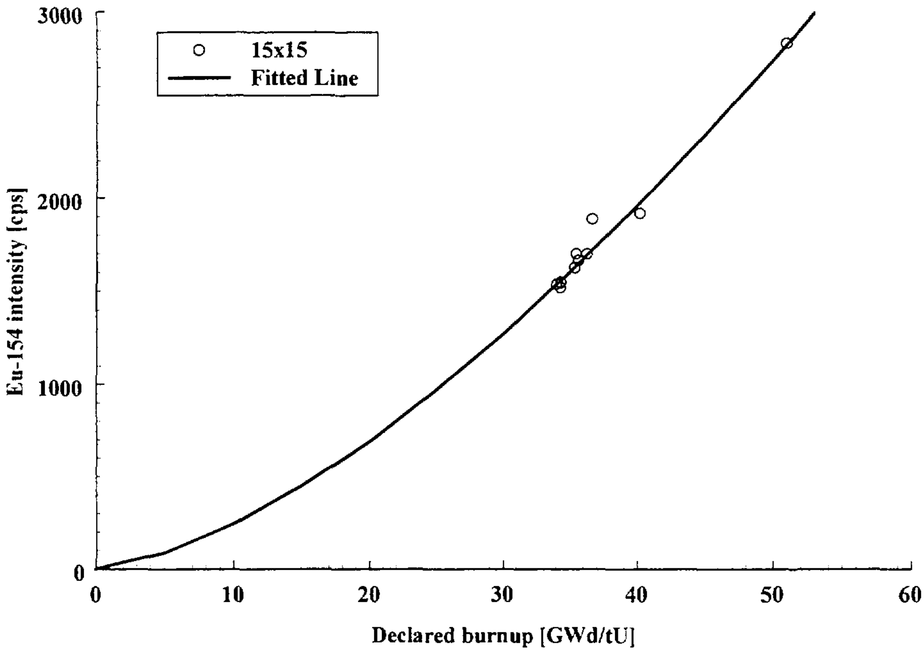




*FIG. 9. $^{154}$Eu intensity as function of the declared burnup for PWR-assemblies*

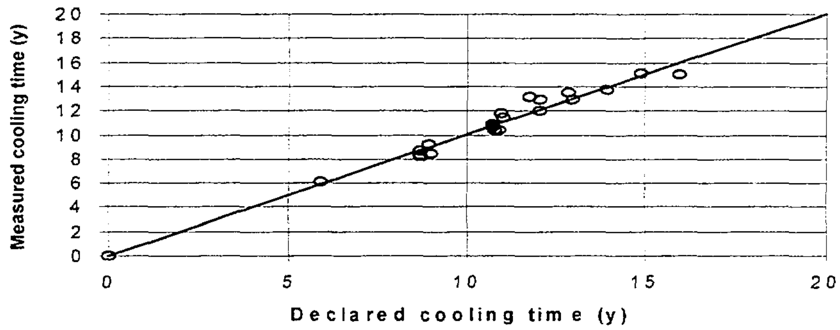



*FIG. 10. Comparison between declared and calculated cooling times for PWR-assemblies*

## 4. CONCLUSIONS

So far the following conclusions referring to the primary objectives can be drawn:

- The calorimetric measurement method used is rugged and has, shown good reproducibility. The uncertainty of the measured decay heat obtained using this measurement equipment was approximately 3% (one standard deviation).

- It was shown that, given that the cooling time exceeds about 10 years, the decay heat can be determined from the measured $^{137}Cs$ gamma-ray intensity within an uncertainty of about 4 % (one s.d.) in the BWR case and about 3,0% in the PWR case. These uncertainties mainly reflect the uncertainty of the calorimetric data.

- It is possible to achieve quick determination of the decay heat by gamma-ray measurements. Typical measuring times are less than 10 minutes.

- A linear relationship between the measured gamma-ray intensity of $^{137}Cs$ and the operator declared burnup was established with a standard deviation of 2%. This can be used to verify the operator declared BU.

- The combination of gamma-ray measurements of $^{137}Cs$ and $^{154}Eu$ was shown to be feasible for determination of the cooling times within 0.5 year (one s.d.) as compared with the operator declared value.

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IAEA-SM-352/41

# EXTENDING DRY STORAGE OF SPENT LWR FUEL FOR UP TO 100 YEARS[1]



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**Abstract**

Because of delays in closing the back end of the fuel cycle in the U.S., there is a need to extend dry inert storage of spent fuel beyond its originally anticipated 20-year duration. Many of the methodologies developed to support initial licensing for 20-year storage should be able to support the longer storage periods envisioned. This paper evaluates the applicability of existing information and methodologies to support dry storage up to 100 years. The thrust of the analysis is the potential behavior of the spent fuel. In the USA, the criteria for dry storage of LWR spent fuel are delineated in 10 CFR 72 [1]. The criteria fall into four general categories: maintain subcriticality, prevent the release of radioactive material above acceptable limits, ensure that radiation rates and doses do not exceed acceptable levels, and maintain retrievability of the stored radioactive material. These criteria need to be considered for normal, off-normal, and postulated accident conditions. The initial safety analysis report submitted for licensing evaluated the fuel's ability to meet the requirements for 20 years. It is not the intent to repeat these calculations, but to look at expected behavior over the additional 80 years, during which the temperatures and radiation fields are lower. During the first 20 years, the properties of the components may change because of elevated temperatures, presence of moisture, effects of radiation, etc. During normal storage in an inert atmosphere, there is potential for the cladding mechanical properties to change due to annealing or interaction with cask materials. The emissivity of the cladding could also change due to storage conditions. If there is air leakage into the cask, additional degradation could occur through oxidation in breached rods, which could lead to additional fission gas release and enlargement of cladding breaches. Air in-leakage could also affect cover gas conductivity, cladding oxidation, emissivity changes, and excessive creep and mechanical property changes. Postulated accident scenarios would be the same for 20 year or 100 year storage, because they are mostly governed by operational or outside events, and not by the cask or fuel. Analyses of accident scenarios during extended dry storage could be impacted by fuel and cask changes that would result from the extended period of storage. Overall, the results of this work indicate that, based on fuel behavior, spent fuel at burnup below ~45 GW·d/tU can be dry stored for 100 years. Long-term storage of higher burnup fuel or fuel with newer cladding will require the determination of temperature limits based on evaluation of stress-driven degradation mechanisms of the cladding.

## 1. EXPERIENCE BASE FROM EXISTING STORAGE WHERE APPLICABLE TO LIFE EXTENSION

Several countries are storing $UO_2$ spent fuel. Detailed performance data are generally not available from these programmes, but no adverse or unexpected performance has been reported. The two major foreign dry storage programmes are in Canada and Germany.

CANDU fuel has been dry stored in over 200 concrete canisters using an air atmosphere starting in 1985, see [2 and 3]. No adverse or unexpected performance has been reported. Both intact and intentional defected spent CANDU fuel have been stored for the last 18 years in a saturated air or helium atmosphere at 150°C. No degradation or additional failure of the intact rods has been detected. The water in the casks became very acidic due to radiolysis of the air and water [4]. Germany has been routinely putting LWR fuel in dry inert storage since 1992 [5] with no problem [6]. No

[1] Work conducted at the Pacific Northwest National Laboratory under contract with EPRI, Palo Alto, CA. The results of this work are reported in *Data Needs for Long-Term Dry Storage of LWR Fuel*, EPRI TR-108757 (August 1998).

surveillance data are available about the condition of the fuel in these casks. In addition, Germany has been conducting full-scale dry-storage cask demonstrations with irradiated PWR and BWR fuel since 1982. Tests lasted up to two years, at which time cladding creep stopped due to pressure, temperature, and decay heat drops in the fuel rod. The United States has over a dozen of licensed dry-storage cask systems [7]. The longest operating dry storage has been in 29 casks at the Surry plant and eight at the HB Robinson plant. Cask loading at these sites was started in late 1985. Monitoring of the operational casks has not indicated any storage operational problems such as leaking casks.

Since 1984, a monitored dry-storage demonstration programme has been conducted by the U.S. Department of Energy (DOE) at the Idaho National Environmental and Engineering Laboratory (INEEL). Krypton-85 indications were found during the three-month testing of the TN-24P cask, when it held unconsolidated fuel. It is thought that the leak in the TN-24P cask occurred when the cask was rotated. A similar finding of Krypton-85 was found in the REA 2023 cask tested by DOE with Cooper BWR fuel at Morris, Illinois. In neither case was the leaker able to be visually identified at the end of the test. The rate of the $^{85}$Kr release along with the lack of visible identification indicates that the breaches were very small. While the tests tell little about the defect mechanisms and the potential for failures in long-term storage, it is important to remember that no gross breaches occurred, and fuel assemblies/canisters were able to be pulled out of the casks with no adverse effects, signs that longer-term storage is feasible.

## 2. NOMINAL STORAGE CONDITIONS

The maximum temperature in the cask occurs after loading and dryout. This maximum temperature is limited by licensing consideration. Thereafter, except for fluctuations due to the external temperature, the temperature in the cask decreases because of the drop in decay heat. For this analysis, the decay curve reported in [8] is used. During a 100-year storage period, the fuel will experience two temperature regimes. The first occurs during the first 10 years when the fuel temperature drops from a maximum of about 380°C, depending on the fuel, to approximately 100°C. The second regime is for the remaining storage time when the temperature is decreasing slightly; it can generally be given a constant upper bound of approximately 100°C.

During dry storage, the spent fuel will be subjected to gamma fields of ~$10^5$ R/h and a neutron flux (>1 MeV) of ~$10^4$ to $10^6$ n/cm$^2$·s [9]. The total dose during 100 year dry storage is anywhere from four (gamma) to seven (neutron) orders of magnitude less than for two to four year residence in the reactor. The gamma field decreases with the age of the fuel and increases proportionately to the burnup. At the maximum expected burnup for LWR fuel (60 GW·d/tU), these dose rates would no more than double those calculated for current cask.

## 3. ANALYSIS OF 100 YEAR FUEL ASSEMBLY BEHAVIOR

There are potentially four different atmospheres in the cask: (1) inert fill atmosphere with some residual water depending on level of drying achieved (this is the most likely since experience to date has shown that in-leakage is highly unlikely [normal condition]); (2) inert fill plus a very small ingress of air due to a small leak in the cask (off-normal); (3) a mistaken air backfill (off-normal); and (4) an unlimited air atmosphere due to a large breach in the cask (off-normal, accident). In the first case where the inert atmosphere stays intact, no oxidative corrosion will take place. Malinauskus [10] has shown that for a small leak in a cask, the supply of oxygen will limit the amount of fuel or cladding oxidation. Although operational history indicates that a large cask leak early in storage is highly unlikely, the potential for degradation of the fuel or cladding in the remaining two cases (mistaken air backfill or large leak) can be evaluated for their potential ramifications.

The only identified potential fuel problem is oxidation of the fuel and consequences of this oxidation, such as fission gas release if the cask leaks during off-normal conditions. As long as the fuel cladding stays intact or the cask does not leak, there is little chance of detrimental fuel oxidation under normal conditions in a well-dried storage cask. Should a rod with a breach be put in storage, there is the potential for fuel oxidation should the cask leak.

With regard to cladding performance, the following aspects have been considered: (1) the maximum additional cladding oxidation that can occur during the additional 80 years of residence in the dry storage canister and its ramifications on emissivity and breach; and (2) the potential for cladding breach due to a stress-driven mechanism such as creep, stress rupture, etc. This includes the potential effect of any cladding annealing due to the initial higher storage temperatures or off-normal temperature excursions. In addition, changes in the mechanical properties of the structural components of the fuel assembly that would affect its ability to maintain critical configuration or impair retrievability are evaluated.

## 3.1. Fuel oxidation

### *3.1.1. Mistaken air backfill, limited air at high temperatures*

Both the exposed fuel and cladding are competing for the available oxygen in a cask, each with a different rate law. The partitioning of the oxygen take-up depends on the number of rods with breaches and the surface area of the total number of rods in the cask. The oxidation of both the fuel and the cladding depends strongly on temperature, so only those parts of the rods whose temperatures are within about 20°C of the maximum temperature will participate in the oxidation.

The amount of oxidation depends on the amount of available oxygen. The total amount of oxygen from the backfilled air depends on the free volume in the cask. For the Castor type cask, this is about 6600 liters or 60 moles of oxygen [11].

Based on the rod dimensions, number of rods per assembly, and number of assemblies in the cask, the rod surface area was calculated. The oxygen uptake as a function of temperature measured by Suzuki [12] was multiplied by the surface area to determine the oxygen usage rate for cladding oxidation.

Due to the different activation energies and rates of oxidation of the fuel and the cladding, the maximum oxidation of the fuel in a limited air environment occurs if the peak cladding temperature is about 250 to 275°C. As the fuel oxidizes to $U_3O_8$, it swells and splits the cladding. The rate at which the fuel is being oxidized is equal to the rate of splitting of the cladding. There appears to be a temperature dependent incubation period before this splitting begins. The incubation time and cladding splitting rates measured by Einziger and Strain, see [13 and 14], were used. In any of the large casks containing many assemblies, the fuel oxidation is equivalent to one or less rods independent of the number of failed rods in the cask. Oxidation takes place in two years or less and results in a decrease of internal cask pressure of about 20%.

Clearly, if a cask has a breached rod with a pinhole, which in-reactor breach statistics indicate will happen, and the cask is mistakenly backfilled with air instead of helium, there will be oxidation of the fuel pellets early in the storage history. The fuel will form grain-sized particulates, splitting the cladding in the process. The consequence of this oxidation is a split in the cladding, which now contains grain-sized oxidized fuel instead of pellets. As the fuel is moved, some of this powder will fall from the cladding into the cask and be available for dispersal. The effect of this oxidized particulate on removing the fuel from the cask is the same whether the fuel is stored for 20 or 100 years; it is contamination that will have to be addressed. There is no apparent reason for this oxidation early in storage to impair continued storage in the same cask.

### *3.1.2. Cask leaks after 10 years (low temperatures)*

It should take fuel about 17 years to reach the $U_4O_9$ plateau at 150°C and about 100 years at 125°C. The fuel remains on the plateau until sufficient $U_3O_8$ forms to start splitting the cladding. Even for the most conservative estimator, fuel oxidation should not be a problem during dry storage after 10 years of storage.

*3.1.3. Radiolysis*

As much as 65g of water or water vapor might be left in a cask after vacuum drying [15]. Upon radiolysis, this water could produce 1.8 moles of oxygen. In the larger casks, this could result in ~10 to 15 g of oxidized fuel, which is probably not enough to split the cladding open. The radiolysis of nitrogen, should a leak occur in the cask, and/or moisture in the canister can lead to the creation of some very aggressive oxidants such as nitric acid and hydrogen peroxide. In stagnant systems with a prototypical radiation field, Canadian tests indicate accelerated oxidation and a significant drop in the pH of the canister water. U.S. tests have had similar effects in unirradiated fuel where 1% nitric acid was added to the system.

**3.2. Cladding oxidation**

In general, the rods attain a transition oxide layer while in the reactor so the post-transition linear kinetics will be operative [16].

The data below 350°C are bounded by a model developed for corrosion in water by Dalgaard [17]:

$$h\ (\mathrm{mm}) = 8.26 \times 10^{2} \times \exp\left(-93\ \mathrm{kJ/mol/RT}\right) \times t \qquad (1)$$

where, in all the above relationships, T is the temperature in degree Kelvin, and t is the time in days.

Even though the equation was developed for water corrosion of Zircaloy, for lack of an adequate database for air oxidation of Zircaloy, the Dalgaard equation was used to predict the amount of oxidation for extended storage.

*3.2.1. Limited air*

In a sealed cask that is accidentally backfilled with air, the thickness of any additionally oxidized Zircaloy is determined by the volume of air present. A maximum additional 6% of the cladding thickness would oxidize and would not be detrimental to performance. The oxidation would be slightly less if there were breached rods in the cask also using oxygen. Of course, the rate of oxidation may be too low to use all the oxygen and achieve these oxide thickness. At 200°C, the time required would be about 200 years.

*3.2.2. Large cask leaks (unlimited air)*

During the first 10 years, the temperature drops from 380 to ~100°C. The amount of cladding oxidation can be calculated by integrating Dalgaard's equation. Less than $5 \times 10^{-2}$ mm of the cladding is expected to be oxidized. At an expected temperature of 100°C for the last 80 years of storage, no more than an additional $7.6 \times 10^{-5}$ mm of cladding is expected to oxidize. The thinnest cladding used is 0.42 mm, so the oxidation during the first 20 years is approximately 13% of the thinnest cladding, and oxidation at 100°C represents less than 0.01% wastage.

*3.2.3. Radiolysis*

Radiolysis will produce 7.2 g of hydrogen, equivalent to 54 ppmw. Even assuming that the original hydrogen level as the rod came out of the reactor was ~130 ppmw, the additional hydrogen would not bring the hydrides to the level where they would cause deterioration of the mechanical properties of the cladding, see [18 and 19]). Radiolysis of any remaining water in the cask after drying should not cause any problems for the extended storage of the fuel. Because there have been very few measurements of the mechanical properties of zircaloy as a function of hydrogen content, this is still an area of controversy.

### 3.3. Potential cladding rupture

There is a requirement to set the storage atmosphere such that gross degradation of the fuel rod is prevented. The NRC does not define gross degradation, but they have shown concern for the ability to recover the fuel from dry storage for further movement or disposal without handling problems or gross contamination. Subsequently, the NRC determined that the most probable mechanism of cladding failure was diffusion-controlled cavity growth (DCCG) in spite of the fact that voids, necessary for void growth, have never been observed in Zircaloy.

#### *3.3.1. Diffusion-controlled cavity growth*

The mechanism was first proposed and analytically developed [20] to determine the time-to-failure. It was reformulated [21] so that for any given temperature profile and duration, the percent of decohesion could be calculated iteratively. In addition, Schwartz and Witte arbitrarily set the acceptable amount of decohesion at 15%. A number of properties of the fuel rod, such as the grain boundary thickness of the cladding, stress on the cladding, and grain boundary diffusion rate, which might change with prolonged service, are required for the calculation.

The maximum temperature limit was set, so at 20 years the maximum decohesion would be 15%. The integral equation can be rewritten so that the limits on time run from 20 to 100 years, and the limits on decohesion run from 0.15 to the new calculated maximum. Since the temperature changes very little after 20 years, a constant storage temperature of 100°C was chosen. Even though there are 80 additional years of storage time, the grain boundary diffusion rate at 100°C is so low that the additional decohesion is $<0.00001$.

#### *3.3.2. Strain limited rupture*

Cladding breach was arbitrarily deemed by Peehs [22] to occur when a rod strain of 1% had been reached. Based on stress due to internal rod pressure, the creep strain was calculated parametrically with the initial maximum fuel rod temperature for 40 years of storage. Due to the cladding strain and a decreasing temperature, the cladding stress decreases to a level where, by 40 years, the additional strain with time is infinitesimal. Hence, based on the creep-rupture mechanism, if the temperature is set to allow 40 year storage, additional storage time without cladding breach would also be permissible. A major uncertainty in this approach is the choice of the appropriate constitutive equation for the creep and thus the actual anticipated creep strain, which can range over an order of magnitude and may or may not cease after sufficient internal pressure drop has occurred [23]. The creep equations need to be resolved properly before the Peehs methodology can be used with confidence for long times. The 1% strain limit also seems excessively conservative. Numerous tests have resulted in spent fuel cladding strain to over 2% without breach, see [24 and 25]. These strain limits were measured at temperatures above 350°C and would have to be confirmed for cooler storage. The Peehs calculations for maximum storage temperature would need to be redone for high-burnup fuel where the temperature profile would be expected to be much flatter with time.

Another concern for long-term storage is the release of helium from decay products in the fuel pellet to the plenum, thus increasing the internal rod pressure. The additional pressure was calculated using the production and release models presented by Cunningham et al. [8]. After 100 years for a rod with 72 GW·d/tU burnup, assuming a linear production rate with burnup, the pressure would increase $< 1$ psi, which is negligible.

*3.3.3. Cladding annealing effects*

The properties of zircaloy cladding are affected by irradiation. However, the radiation fields to which the cladding is exposed during storage are many orders of magnitude lower than those already encountered in-reactor and do not change the effects that take place in the reactor, which usually saturate during the first or second irradiation cycle. For this reason, changes of yield strength, elastic modulus, and fracture toughness due to the radiation field from the fuel are expected to be small during the storage period, see [26 and 27]. Although the strength of the cladding may be increased by irradiation, the temperature may be high enough during the initial part of the storage period to cause some annealing to occur. The result of annealing is to decrease the strength and increase the ductility of the material. It may also assist in relaxing internal stresses, see [26 and 27]. The temperature for 50% recovery of irradiation damage in Zircaloy-2 was measured [28] and determined to be 380°C. Extrapolation of Kemper's [29] data indicated an approximate 40 to 50% recovery in six months at temperatures between 325 and 350°C. Below 200°C, there is virtually no annealing affect on the mechanical properties, see [28 and 30].

**3.4. Transfer to a new cask after 20 year storage**

*3.4.1. Dry-to-dry transition*

During reactor operation, a coating of crud developed from corrosion of the materials in the primary system forms on the fuel rod. The main constituent is $^{60}Co$. The crud tends to be very tenacious on PWR rods and much more flaking and easily dislodged on BWR rods. A definitive compilation on the properties of crud is given in Sandoval, et al. [31].

There is the potential that some fuel assemblies will have to go through a dry-to-dry transfer in order to be stored for more then the licensed 20 to 40 years. During that transfer, there is the possibility that with the lid not yet secure, the cask could tip over and release spalled crud to the atmosphere. The extent of the release will depend on the amount of activity released, density of crud on the surface of the rods, amount of the surface that is actually covered with crud, amount of crud that spalls when the cask tips over, and crud that becomes airborne. NUREG-1536 [32] specifies the surface density for $^{60}Co$ on fuel rods as 140 $\mu Ci/cm^2$ for PWR rods and 600 $\mu Ci/cm^2$ for BWR rods. The maximum spot density is the maximum concentration of radionuclides found anywhere on the rod. The average crud density on a rod is at least factor of 10 less. $^{60}Co$ has a half-life of 5.26 years. For fuel put in dry storage, five or more half-lives will have passed when a dry transfer is made with a consequent reduction in the $^{60}Co$ activity of a factor of 30. The densities quoted by NRC are the high end points in the distribution. Any particular assembly might have a substantially lower maximum spot density. Accounting for half-life decay or the use of a realistic crud density for any particular set of assemblies to be stored does not decrease conservatism, but reduces the activity by a factor of 300.

Estimates of crud spallation from rods stored at 230°C ranged from 1.6 to 4.8%. These measurements are probably reasonable for static dryout up to a temperature where rod ballooning occurs. Once ballooning over 6% starts to occur at 400°C, then additional crud will spall. In any case, the measurements are all within the 15% spallation figure for static storage given by the NRC. When a cask tips over, the rods are subjected to an impact. There are no measurements of the adherence of crud to the rods during an impact. For lack of data, the Sandoval evaluation used a spallation factor of one for accidents. Once the crud spalls, only a small fraction will be available for gaseous transport. The remainder will settle quickly on other surfaces within the cask. The degree of settling will depend on the particle size distribution of the crud. No data are available for crud that has been removed from a rod as a result of impact.

*3.4.2. Wet transfer through the pool*

Cask reflood must be managed in a manner that prevents adverse thermal shock of the fuel and adverse pressurization of the cask.

3.4.2.1. Effect of wet transfer on pinhole breaches

When a failed rod is placed back in the reactor pool, the water is at atmospheric pressure and ~50°C, while the uncooled rod temperature is between 100 and 150°C. As the rod cools, its internal pressure will drop, which might fill up to 25% of the internal rod void volume with water to equilibrate the pressures. If this does happen, steam could form, and a mixture of water and steam could be in contact with the fuel. The steam would soon subside as the rod cooled down, leaving the fuel in contact with water at ~50°C. At these temperatures, many studies [33] have shown that the leaching of spent fuel is very slow. Considering that the fuel is in the pool for transfer purposes only, the duration would be short; i.e., <30 hours, and leaching of the fuel would be negligible. Even if the fuel stays hot for a short time, Canadian studies of fuel oxidation at 150°C in a saturated humidity atmosphere show limited surface oxidation of the grain boundaries to $U_4O_9$, and only after many years.

3.4.2.2. Effect of transfer on intact cladding

When the cladding at 125°C is submersed in the pool water at 25°C, the thermal stress on the cladding is given by

$$\sigma = E \times \Delta T \times \alpha \quad (2)$$

where

$\sigma$ is the thermal stress across the cladding,
E is the elastic modulus equal to $14 \times 10^6$ [34],
$\alpha$ is the coefficient of thermal expansion, and
$\Delta T$ is the temperature differential across the cladding.

The largest value for the thermal expansion coefficient for either Zircaloy-2 or -4 in the range of 0 to 1000°C is $6.48 \times 10^{-6}$/°C. Using the conservative assumption that one side of the cladding has completely cooled and the other side not at all, the temperature differential is 100°C. The thermal stress is calculated to be 66 MPa. This stress is small compared with the yield stress, so no permanent deformation is expected as a result of exposure to cold water.

3.4.2.3. Effect on crud spallation

Although the thermal shock induced by exposure of cladding to cold water is insufficient to cause permanent deformation in the cladding, it is possible that a significant amount of crud spallation could occur. The amount of spallation depends on the thermal transfer characteristics of the crud, its mechanical properties (strength, bulk modulus, and coefficient of thermal expansion), and its adhesion to the cladding. Sufficient data are not available at this time to determine the propensity for crud spallation to occur, either for PWR fuel or BWR fuel, which will have to be separately evaluated due to the difference in their crud adhesion properties.

### 3.5. Accident evaluation

If conditions are set within a cask so that fuel rods do not fail under normal operating conditions, an additional 80 years of storage either within the same cask or after transfer to a new cask should not further degrade the fuel rods. The two major concerns for the fuel are oxidation due to the formation of a cask leak or further fracturing of the fuel pellet fragments into a fine powder due to the impact. The considerations for the cladding are breach due to impact, oxidation due to a large leak, crud spallation, development of incipient cladding cracks that could lead to breach at a later time, breach due to rapid cooling during a flood, and eutectic formation, melting, and breach due to rapid heating during a fire.

An accident is a short-duration event. As such, many of the models of fuel rod and fuel assembly responses developed by Rashid [10] for transportation are applicable. The assumptions made in that analysis and the basis of the assumptions appear plausible for both transportation and storage accident events. In some instances, the conditions for the evaluations such as for flood, fire, and cask tipover due to impact are directly applicable. Other calculations, such as for the puncture drop and 9-m fall, will have to be related to the actual energy impacted onto the cask during storage-related events such as a tornado, wind storm, or missile attack. The g-forces during transportation appear to be considerably greater than the seismic standard earthquake's 0.12g, and thus the results of the analysis may also be transferable. For fire, such phenomena as rod burst, cladding melting, and eutectic formation between the cladding and the assembly support structure are considered and found not to be possible. For the impact events, degradation phenomena such as rod fracture, rod rupture, and structural analysis of the assembly are considered. For the rod evaluation, the three main materials or rod inputs to the evaluations are the stress in the rod, cladding flaw state, and fracture toughness of the materials. Only fracture toughness data from Zircaloy-2 over a temperature range of 25 to 250°C was used. Calculations using data for irradiated Zircaloy-4 should be conducted.

### 3.6. Transportability of the fuel

After interim storage, be it 20 or 100 years, the fuel will have to be transported to a repository for final disposal. At the time the storage system was licensed for 20-year storage, the fuel needed to be suitable for transport after 20 years. As discussed, the fuel or cladding damage does not increase appreciably over the last 80 years of storage. Hence, transportation requirements can be met after either 20 or 100 years.

### 3.7. Burnup effects

For operational efficiency, the utilities are driving their fuel to higher burnups. The current average in the neighborhood of 35 to 45 GW·d/tU could go up to 55 to 65 GW·d/tU. This is being approached in two ways: 1) extending the burnup of current designs until reactor performance is no longer acceptable, and 2) development of newer designs and use of materials that are more corrosion resistant, have greater dimensional stability, and increased margins of performance. The impact of higher fission gas inventory, rim-effect on oxidation, more hydrides in the cladding, and significantly larger oxide layers in higher burnup fuels on extended dry storage are being evaluated. Interestingly, the breach rate of the fuel has not increased appreciably with the increased burnup [35]. The three areas where higher burnup has the potential to affect the long-term dry storage of spent fuel are stress-driven cladding-failure lifetimes, thermal performance of the rods, and radionuclide inventory of the source term.

Higher enrichments combined with longer irradiation times will result in the fuel containing a higher radioisotope inventory. This will reflect in cask licenses to accommodate increases in heat load, partial cask loadings, or longer cooldown times so the cask can handle the heat load and shielding requirements.

Many of the degradation mechanisms for the cladding are stress driven. Extra corrosion of the Zircaloy cladding results in cladding thinning and a higher cladding stress. Manufacturers have countered this trend by introducing newer types of more corrosion-resistant claddings, such as Zirlo and Zircaloy-2p.

The internal pressure in the rod can occur from increased production of the noble gases in the fuel due to the higher burnup and increased release of the gas from the fuel matrix to the plenum. The production of xenon, which is the main fission gas, is proportional to the burnup [36] and thus the amount of fission gas produced by a PWR rod at 50 GW·d/tU will result in a greater rod pressure at higher burnups even with the same gas release. In addition, there are strong indications that the gas release increases at the higher burnups. This increased release is observed in high-burnup fuels that have formed a rim of fine grains near the pellet surface. This region of the fuel tends to exhibit much higher gas release.

The mechanical properties of the Zircaloy saturate after approximately one-cycle of irradiation, so these properties should not change with the higher burnup if other changes in the cladding did not occur. Extending the burnup of standard Zircaloy cladding results in additional oxide formation. About 10 to 20% of the concurrently produced hydrogen dissolves in the cladding that precipitates predominately as circumferential hydrides upon cooling. The ultimate and yield strengths are not affected strongly by the hydrogen content until above 1500 ppm, see [37 and 38]. As the hydrogen content increases, the Zircaloy becomes progressively more brittle at higher temperatures [39]. A high hydrogen content could reduce the allowable strain before a cladding breach or reduce the ability of the cladding to survive an accident impact. The mechanical properties of the new cladding types are still proprietary. The ability of these new alloys to withstand accidents and behavior under dry storage conditions has to be evaluated.

## 4. CONCLUSION

Overall these results indicate that, based on fuel behavior, spent fuel at burnups below ~45 GW·d/tU can be dry stored for 100 years. This length of storage will not adversely affect normal transfers and transport activities during, or at the end of, the storage period. Long-term storage of higher burnup fuel or fuels with newer cladding will require the determination of temperature limits based on evaluation of stress-driven degradation mechanisms of the cladding. Post-irradiation mechanical properties of the newer high burnup cladding and fuel may need to be taken into account.

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# REGIONAL SPENT FUEL STORAGE FACILITY (RSFSF)

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**Abstract**

The paper gives an overview of the meetings held on the technology and safety aspects of regional spent fuel storage facilities. The questions of technique, economy and key public and political issues will be covered as well as the aspects to be considered for implementation of a regional facility.

## 1. INTRODUCTION

Spent fuel storage is a common problem in all countries with nuclear reactors. Whatever strategy is selected for the back-end of the nuclear fuel cycle, the storage and transport of spent fuel will be important services. As of today about 130,000 tHM of the 200,000 tHM spent fuel arisings from power reactors are stored world-wide. About 70,000 tHM were reprocessed so far. From research reactors more than 62,000 fuel assemblies are stored world-wide .Final disposal facilities for spent fuel and high level waste are not available and are not expected to be implemented in the near future.

Safe management of spent fuel from power and research reactors involves the technology, resources and licensing procedures, so that the exposure to ionizing radiation of the operational personnel and the general public is controlled, and the environment is protected in accordance with national regulations and international consensus.

Most countries with power reactors are developing their own national strategy for spent fuel management, including interim storage. However, several countries with a small nuclear power programme or only research reactors, face the serious problem of extended interim storage and disposal of their spent nuclear fuel. The high specific costs for the construction of away-from-reactor extended interim storage facilities and/or geological repositories for the relatively small amounts of spent fuel accumulated in such countries, is obviously not reasonable and, therefore, from an economical point of view, access to a regional interim storage facility and/or repository for their fuel would be an ideal solution.

It is interesting to note that *de facto* Regional Spent Fuel Storage Facilities (RSFSFs) exist in several countries. The word "regional" is used in the broad sense of the word that is understood as a multinational Spent Fuel Storage Facility (SFSF) covering more than one country. However, there are certain advantages associated with the co-operation among different countries in the same geographical region.

In Western Europe, commercial considerations provided the incentive for the development of spent power reactor fuel management systems. Eurochemic was one of the most significant early projects for a multinational arrangement. COGEMA and UKAEA-BNFL are effective involved in interim storage of spent power reactor fuel from a number of countries, while awaiting reprocessing. In the former USSR, arrangements seem to have been largely politically motivated. The service offered by the USSR was mainly centred around their own national interest, such as the fuel leasing arrangements concluded with the former East Block countries. The arrangements continued until 1990 and were de facto suspended by legislation. Meanwhile, new arrangements are under investigation.

Research reactor fuel from all over the world is at present stored in wet interim storage pools at the Receiving Basin for Off-Site Fuels (RBOF) facility at DOE's Savannah River Site. Proliferation concerns weighed heavily in this case. Some research reactor fuel is stored at Dounray, Scotland, waiting for its reprocessing.

The safety and economic benefits from the implementation of regional spent fuel storage facilities are very attractive in terms of reduction of the number of spent fuel storage facilities world-wide, enhanced economy due to the scale of storage construction, and easier safeguarding to ensure non-proliferation. However, there are still various problems to solve as to find operators of such facilities with governmental support and to convince countries of proliferation concern to participate. The time is ripe for serious discussion of such regional facilities and to begin planning for the day when neither take-back programmes nor the reprocessing option might be available.

The IAEA has made a start by convening two meetings of experts to collect and evaluate information on a regional spent fuel storage facility. The first one was held in December 1997, the second one from 28 September to 1 October 1998.

The main objective of the meetings was to discuss the different technologies and safety aspects of regional spent fuel storage facilities for power and research reactor fuel, requirements regarding logistics, geology and climate, the question of the final destination of the spent fuel, the linkage of regional storage to regional disposal, economic and legal questions of regional storage (profit center or cost sharing), and public and political issues that would influence the location and acceptance of a regional spent fuel storage facility.

It is recommended that the following specific regional issues be considered, i.e. the lack of research reactor fuel take back arrangements of Russian origin fuel and the lack of spent fuel storage facilities required for research reactor decommissioning.

## 2. THE REGIONAL SPENT FUEL STORAGE CONCEPT

In order to make regional storage attractive to prospective customers, the hosting country may also offer other services in additional to storage, as shown in Figure 1. The spent fuel management options catered for in this regional spent fuel management system are as follows (Fig. 1):

In the case where material is returned to the customer (**Option 1**):

- Spent fuel;
- High level waste (HLW) after reprocessing.

In the case where material is not returned to the customer:

- Ongoing storage of spent fuel and/or HLW (**Option 2**);
- Disposal of spent fuel and/or HLW (**Option 3**).

**Option 1**

A hosting country that wishes to embark on a regional storage project should decide in advance what the final destination of the spent fuel would be. That is, whether the spent fuel is to be returned to the country of origin at the end of the storage period, or whether the material is to be disposed of in the hosting country.

**Option 2**

Another approach is to store the spent fuel on an ongoing basis without making provision for final disposal. The ongoing storage option may require repackaging of the stored materials during the long-term storage period envisaged.

**Option 3**

At the end of the interim storage period the spent fuel can be disposed of in a *final* disposal facility.

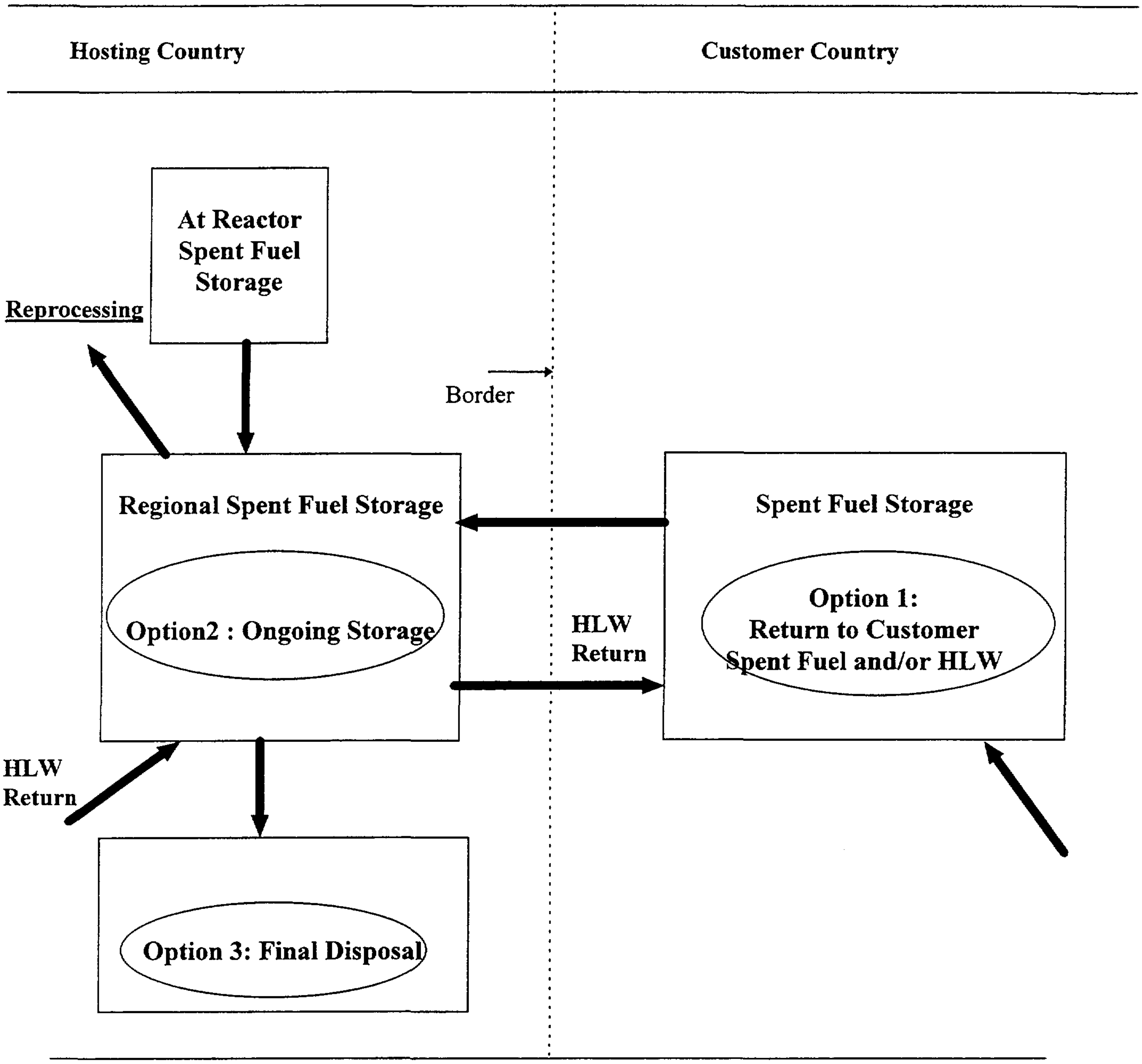


*FIG. 1. Example of a typical regional spent fuel management*

## 3. FEASIBILITY OF RSFSF

### 3.1. Technical feasibility

#### *3.1.1. Spent fuel inventory*

Spent fuel is located in all countries with nuclear power plants (NPPs). As of today 130,000 tHM of the 200,000 tHM spent fuel arisings from power reactors and more than 62,000 fuel assemblies from research reactors are stored world-wide. Up to 2010, the amount of stored spent fuel will raise to 225,000 tHM. The situation and its development can be seen in Fig. 2.

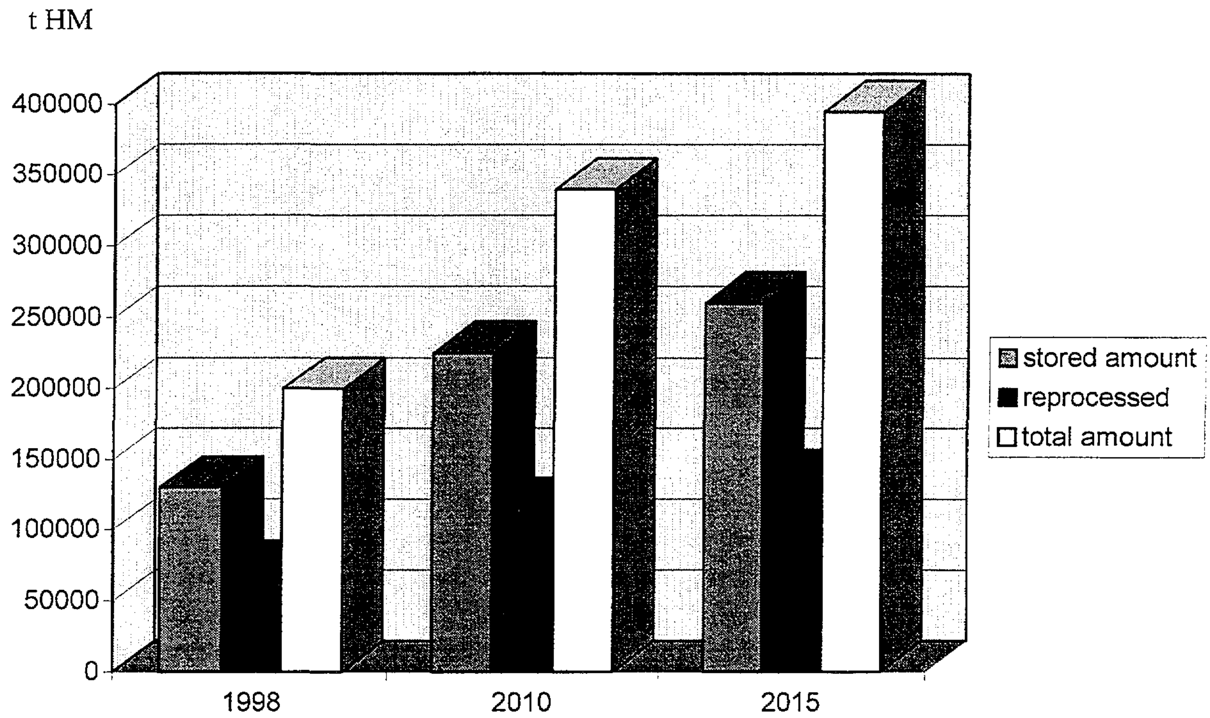


*FIG. 2. Stored, reprocessed and total spent fuel arisings*

### *3.1.2. Storage technology*

In principle two different technologies can be considered: wet or dry storage facilities. Both technologies are proven.

**Wet storage**

Wet storage pools are common at a NPPs and at reprocessing plants, but exist also as stand alone facility. A general advantage of wet storage systems is the fact that stored fuel can be easily retrieved, controlled and checked. In the storage pools, a relatively large quantity of spent fuel can be stored at the same time. A general disadvantage of a wet storage system, is the need for active systems for cooling and cleaning the water, as well as for operator supervision and support systems. Cooling may only be disrupted for a short time span of several days. From cleaning the pool, secondary waste will be produced constantly.

**Dry storage**

Dry storage is used successfully in several countries for centralised intermediate storage, especially using metal casks and concrete silos. A general advantage of dry storage is the ease of implementation in the case where casks are used. Dry storage is easy to operate, as little or no active systems are needed. Dry storage can accommodate spent fuel as well as vitrified waste. The capacity can be easily adapted to the needs. Dry storage facilities usually have a high degree of safety, even against rare events such as an air crash.

**Additional technical aspects**

For handling of fuel to conditioning, repackaging, sealing and any treatment in connection with incidents and accidents, it is advisable to have a service unit (hot cell etc.) within the storage facility.

**Recommendation for a regional storage technology**

Dry storage systems are preferred for a regional solution for the following main reasons:

- no active systems;
- low maintenance;
- easy to operate and to adapt to the needs.

- open for all kind of fuel
- less secondary waste
- simplicity of accident prevention.

Further studies are required, however, concerning the economy of the chosen technology, taking into account the same level of safety and radiation protection.

*3.1.3. Fuel acceptance criteria*

All types of fuel should be accepted in the RSFSF, i.e. spent fuel from power reactors, research rectors and residues from reprocessing as e.g. vitrified high level waste. All types of power reactors should be addressed such as PWR, BWR, RBMK, WWER, HTGR, CANDU. Fuel from research reactors are more divers and care must be taken esp. to fuel leakage due to corrosion. Generally, there should be an acceptance also for leaky fuel elements. The acceptance criteria should be easy to fulfil by a potential customer as the attractiveness must be high.

*3.1.4. Long-term stability of systems and stored fuel*

The exact duration of the intermediate storage is not fixed yet as this will be done in the licensing process and in the commercial arrangements between service provider and customer. It can be imagined, that besides a straight forward storage of less than 50 years ( e.g. several European storage facilities are licensed for 40 a ), there might be the need for ongoing storage over an even longer period. Due care should be given to the long term stability of all its components and especially those components which are safety related including the fuel itself. The fuel can develop cracks, due to high burn up brittle fraction could occur. Especially research reactor elements can have corrosion pits from their pool storage period. The possible effects should be foreseen in the safety concept of the whole facility and addressed.

*3.1.5. Conditioning of stored fuel*

**Preparation before storage**

The preconditions for the fuel to be accepted by the RSFSF should be minimal. The way of fuel pre-preparation may be subject of special arrangement between customer and storage facility. The drying and He-filling of storage casks are such measures, which are useful to avoid corrosion.

**Repackaging, consolidation and retrievability**

It is very useful if not necessary to have facilities on the RSFSF site, which enable a repackaging of fuel or at least of storage units. This will give flexibility in handling incidents or accidents under due radiation protection. A special kind of repackaging is consolidation. This will save space for storage and thus be cost-saving. Intermediate storage means per se also having the possibility to retrieve the stored fuel at a certain time, e.g. to reprocess it. The retrievability is given normally with both systems, wet and dry.

*3.1.6. Siting*

**General remarks**

All site-related factors likely to affect the safety of the facility have to be evaluated. On the other hand, all impacts of the facility to members of the public and the environment also have to be addressed. Finding a suitable place for a RSFS is not very problematic from technical point of view.

**Geology and climate**

The geologic and climatic conditions in the host country will be very specific, so no general recommendations can be made. However, if possible, the most favourable site with the lowest geological and climatic influence should be chosen.

*3.1.7. Safety*

**Safety criteria and standards**

The RSFS shall have an internationally accepted safety standard. For that reason it shall be in accordance at least with the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management and the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. There are basically four safety objectives to achieve at any time during the whole storage period:

- safe enclosure of radioactive material;
- shielding;
- under-criticality;
- safe removal of residual heat.

The technology and the design should be aimed at optimising the radiological impact according to the ALARA principle.

**Safety assessment**

Before construction of the RSFS a systematic safety assessment should be carried out. It has to cover the whole lifetime of the facility with due care for the safe enclosure of the fuel over that time.

**Accident and incident analysis**

A very important part of the safety analysis is the accident and incident analysis. The safety objective is, that even in case of accidents or incidents there is no inadequate situation for people and environment.

*3.1.8. Licensing*

The RSFS has to be licensed according to national regulation of the host country. To make sure, that this licensing procedure leads to a high safety standard of the RSFSF, the host country should have experience in that field or, if not, co-operate with a customer country for that reason.

*3.1.9. Infrastructure aspects*

**Transport**

Transportation is the major logistic item for a storage facility. It can be assumed, that all transport goes on according to the IAEA Regulations for the Safe Transport of Radioactive Material. Transport is generally safe and reliable. No major transport accident was observed with spent fuel in the past. For convenience of handling it is strongly recommended, that the RSFS can be arrived by rail. Normally transport casks are heavy ( about 80 - 120 tons), and this poses no problem for transport by rail.

**Linkage to final disposal**

The intermediate storage is connected with the question of what happens to the fuel after it. One of the possible options is final disposal. The basic element of linkage to a final repository will be the possibility of conditioning the fuel, whether on the spot of the RSFSF or elsewhere.

*3.1.10. Safeguards and physical protection*

The advantage of a few regional facility in comparison to many facilities in different countries is obvious: the safeguards control would be easier. There is no doubt that from a technical point of view appropriate measures can be taken at any potential site for a RSFS. The only (legal) precondition is, that the host country has signed the Convention on Physical Protection and the Treaty on Non-proliferation. This is preferable for customer countries too.

*3.1.11. Research & Development*

The storage periods envisaged by the RSFSF concept are in the order of magnitude 50 - 100 a or even longer. It is very important to evaluate the behaviour not only of spent fuel but also of components essential for safety of the facility for such long periods. There is proven evidence that storage over approximately 50 years can safely be done and several national licences have been granted for that period. Considerable information has been collected on fuel behaviour esp. during wet storage and to a smaller extent for dry storage too in the IAEA BEFAST-programme, which started in 1981 and was finished in 1996 (see TECDOC-944). There are still ongoing IAEA activities within the SPAR programme (Spent Fuel Performance Assessment and Research). However, it seems to be still a need for information especially on long term effects with high burn-up (60 - 75 GW·d/tHM) and also for new cladding material specially designed for high burn-up.

**3.2. Economic feasibility**

*3.2.1. General remarks*

There generally exists a need to reduce the cost of nuclear power generation. This objective can be achieved in a big part of utilities or countries, by closing the back end of nuclear fuel cycle in a cost effective manner. Basically exist three options of nuclear fuel cycle:

- direct disposal {once through};
- reprocessing {closed cycle};
- deferral of the decision.

In any case, spent fuel has to be temporarily stored for a limited period, depended on the chosen option. In determining whether to establish a regional SFSF, the cost and liabilities to all affected partners must be weighted against the benefits. The economic considerations would normally be the driving force for a regional solution.

*3.2.2. Scope of service*

The scope of service offered by the provider to the potential customer is interim storage, on the terms agreed between the customer and the service provider. In addition to the interim storage service offered, the service provider can offer other services, i.e.:

- repacking and consolidation;
- reprocessing ,
- final disposal.

*3.2.3. Storage system, location vs. cost of service*

The cost of interim storage services will depend on the cost of systems used for interim storage facility. Cost of the facility is deeply depending on the technical solution, and licensing procedure of the facility, however safety standards must never be compromised. In any case, storage system including storage location must go through optimisation based on the cost benefit analysis.

*3.2.4. Liabilities, transfer of the title, financing sources and conditions.*

Transfer of the title of spent fuel could take place:

- when the fuel is physically transferred, in which case the service provider accept full responsibility;
- at some future date, depending on the contractual arrangements.

Financial provisions for future liabilities of the host country have to be seriously considered in the process of establishing a regional storage facility. A very useful way is collecting financial sources through a fund, specially created and offered to the spent fuel storage and associated purposes.

### 3.3. Institutional feasibility

#### *3.3.1. Organisations*

Regional approach to the storage of spent fuel would require the involvement of relevant institutions. On a national level governmental and regulatory bodies as well as spent fuel producers and operators will take part in a process. On an international level institutions like IAEA, OECD/NEA, EURATOM, etc. may be involved.

#### *3.3.2. Legal aspect*

**Safety, safeguard and licensing**

For a regional spent fuel storage primarily, the laws and regulations of a host country will apply. Therefore, it is important that the hosting country has well established national legal framework and mature regulatory system. Internationally recognised safety principles, standards and practices should be applied. Regulations of host and customers countries should conform to relevant international conventions and treaties.Spent fuel and high level waste stored in a regional spent fuel storage is subject to international safeguards regulations. For safeguard control and inspections the relevant international organisations like IAEA or EURATOM are applied. For licensing, the laws and regulations of the hosting country will apply. It is advisable that the licensing system is well defined and transparent.

**Ownership of spent fuel**

There are three options regarding the ownership of spent fuel stored in such a facility:

- the ownership of fuel remains with the customer; after the storage period expires the fuel is returned to the owner;
- transfer of ownership to the host country is delayed and can take place at some later time, depending on contractual arrangements;
- ownership of fuel is immediately transferred to the host country; no return of fuel is foreseen.

If the take back of fuel is agreed among the parties the contract between the host and the customer country requires strong commitments on both sides. In such a case an international assurance that the agreements will be respected may be required.

**Liability**

Closely related to the ownership of spent fuel are future liabilities of the host and customers' countries. In case the ownership of spent fuel stays with the customer's country the liabilities of the host country can be covered by adequate financial provisions and agreements among partners. Storage fees should cover all expenses for storing, for unforeseen future events (i.e. future repackaging) and damages. When the transfer of ownership to the host country is included in the arrangement the service provider takes the full responsibility for the fuel. The liabilities of the host country include also future disposal of spent fuel. Such long-term responsibilities would definitely include the state of the hosting country regardless if the regional storage operator is private enterprise.

### 3.4. Ethical feasibility

The ethics of regional spent fuel storage depends to a large extent on the conditions under which the *storage burden* is shifted from individual customer countries to a single hosting country. These conditions determine whether or not the regional solution would be ethically justified. In order to treat the ethical issues systematically, two different approaches are used. The first approach is to consider the present generation's *obligations* both to itself and to future generations. The second approach is to weigh up the *consequences* of this generation's actions both with regard to itself and to future generations.

The IAEA Safety Fundamentals (Reference) incorporate, to a large extent, the obligations that apply to spent fuel management in general:

- *Human health* and the *environment* must be protected.;
- *Future generations* must not be unduly burdened as a result of the establishment of a regional spent fuel storage system in any particular country;
- *Third party* countries, which do not directly participate in the regional spent fuel storage arrangement, but which form part of the same region, must not be unduly burdened by this project;
- *Equity or balance* must apply among the participating countries.

### 3.5. Political feasibility

The political issue is clearly of considerable importance in the regional spent fuel storage concept. The state in the hosting country needs to put its blessing on the entire enterprise before it could be implemented. In the first place, when addressing the issues, it is important to keep in mind the *typical* perceptions of nuclear power that exist on the part of the public at large and politicians in particular. The spent fuel issue cannot be divorced from the nuclear issue. In some cases opposition to nuclear power is reflected in those areas where the industry is *perceived* to be most vulnerable, such as in the case of nuclear waste management.

What is usually uppermost in the mind of the public and therefore also of politicians in general is the *safety* of nuclear facilities. When dealing with the safety issue, it should be remembered that the public does not always appreciate the difference between the different types of nuclear installations.

Of great importance in promoting the regional concept at the political level, is to stress the *advantages* of the concept. It is uncertain which benefits would weigh most heavily with politicians: that is, enhanced safety, non-proliferation or commercial considerations. The weight of the benefit would clearly depend on the policies of individual countries.

The chances of successful implementation of the concept would be greatly enhanced in the case where there already exists a strong *bond* between the prospective participants to the regional project. Such a bond could for instance be a strong economic tie between countries falling in the same geographical area. The European Union is a good example of such a bond.

It is of course likely that the strongest opposition would emerge from within the political circles of the hosting country. Of considerable interest is where the initiative for the whole enterprise comes from. Namely, whether it comes from a private entrepreneur or from the institutional side within the hosting country.

## 4. BENEFITS AND RISKS

The benefits related to RSFSF cover almost all the aspects of the projects. Nonetheless there are some challenges and implications that must be faced in developing the project. These challenges should be known from the beginning or at least predictable.

### 4.1. Benefits

The techniques to implement a SFSF can be considered proven technologies. However, as the conceptual design of the systems used differs one from the others, some advantages can come from sharing the experiences done so far by means of pool of experts. Furthermore it is reasonably accepted that the limitation of storage sites lowers radiological risks associated with nuclear waste as well as environmental impacts.

The economic benefits are foreseen for hosting country as well as for the customer countries, because of the economy of scale. The host country, is expected to have economic advantages, in terms of funds from customer countries and/or profit on the operation of the facility. Important economic benefits to the local community of host country are expected too (employment, local infrastructure, economic incentives). The customer countries can have advantages in terms of unit cost of spent fuel stored.

The development of an international framework for the implementation of a regional spent fuel storage may have an important reflection for future regional disposal initiative (testing of international treaties feasibility, development of international framework for future co-operation, etc.). The existence of an international framework and treaties signed by the countries joining RSFS assures the transparency of back-end fuel cycle and limits therefore the possibility for nuclear proliferation.

### 4.2. Risks

The economic risks for hosting and customer countries could be:

- hosting country may fail in recovering its investment if customers leave the group in the development phase of the project or they don't provide the flux of spent fuels foreseen, making the unit cost of storage rises;
- customer countries may lose their investment if the host country fails in gaining the licence for constructing and operating the facility.

The stability of institutions and legal framework, could not be maintained in a long-term period, being the life period of SFSF much longer than life that many institutions, especially at international level, have been experienced so far. Public acceptance is crucially important and it could be the weak point in the process of project development. Furthermore the effectiveness of international treaties could be compromised by modifications in political relations among partners (host, customers and third parties) as well as changes in national borders.

## 5. IMPLEMENTATION

There are basically two ways in which a regional spent fuel storage service can be initiated. In the first place it can be initiated as an entirely private enterprise. In the second place it can be launched on the basis of a public or government undertaking with or without private participation.

### 5.1. Free enterprise approach

In the case of a free enterprise approach, it is assumed that a private spent fuel storage *service provider* in the hosting country decides for purely commercial reasons to provide a regional storage service to interested customers. That is, the service provider should stipulate whether or nor he is prepared to accept *ownership* of the spent fuel delivered to the regional storage facility. As a private company, the service provider cannot be expected to ensure long-term repository or even ongoing storage control. Therefore, in the final instance it is unavoidable for the state in the hosting country to assume such a long-term responsibility.

### 5.2. Institutional approach

On the basis of the above-mentioned difficulties associated with the long-term storage or final disposal of spent fuel, it is most likely for the state of the hosting country to initiate the regional storage project. The incentive for such a hosting country could be commercial, but would rather tend to be based on non-proliferation and safety considerations. In this case, the state of the hosting country may appoint a private operator to take charge of the storage operation, whilst the state assumes liability for all transactions concluded with prospective customers. Where the hosting country takes institutional responsibility as mentioned above, the hosting state would automatically assume the long-term obligations that a purely private service provider would not be able to undertake. That is, the state would accept title of the spent fuel with a view to ongoing storage or final disposal in the hosting country.

6. RECOMMENDATIONS

The experts recognised the large benefits of regional spent fuel storage analysed in terms of the technical, economical, institutional and ethical/socio-political aspects involved and recommended to stimulate the debate on regional spent fuel storage and other steps in the spent fuel management process. IAEA should invite Member States to investigate the possibility of regional solutions to spent fuel problems within the global context.

Spent fuel storage as such is a well established and proven technology. Concluded that there are potentially a number of countries that could fulfil the site criteria for hosting a regional spent fuel storage facility, the discussion among Member States with regard to the implementation of such a regional solution should be encouraged.

The experts recognised that the political and public relations issues involved in regional spent fuel storage might present difficulties. There are still several issues that need to be discussed further, notably the institutional and socio-political questions. Any solution set up to expedite a regional storage project should facilitate transparency and the Agency should stress the benefits of safeguards application and the pursuance of non-proliferation objectives within a regional context. IAEA should undertake an analysis of the global spent fuel situation in order to assist with the identification of potential regional focal points. The institutional responsibilities for regional storage over extended periods need to be analysed by both the hosting and the customer countries.

7. CONCLUSIONS

It appeared from the preceding discussions that the regional spent fuel storage concept is entirely feasible, in principle. However, one should be aware of the many political and public acceptance issues that may arise in opposition to a regional concept. Never the less, there are many benefits in a regional solution like the obvious economies of scale achievable with regional spent fuel storage facilities. It is also clear that storing spent fuel in a few safe, reliable and secure facilities will facilitate safeguards and physical security and reduce the risk of proliferation, especially for highly enriched uranium fuel from research and test reactors.

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# CLOSING THE GAP BETWEEN SPENT FUEL STORAGE AND FINAL DISPOSAL IN A MULTINATIONAL MANAGEMENT SYSTEM

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**Abstract**

In this paper, a multinational spent fuel management concept is proposed. The management concept is based on a service agreement between countries, which intend participating in a common spent fuel (SNF) management venture. Accordingly, one of the participants in this venture would act as the hosting country, while the others fulfil the role of customer countries. The hosting country would agree to accept SNF from customer countries under specific conditions, as required by the service agreement. The service agreement should cover a sufficient number of options that customers can use, such as storage, reprocessing or disposal. The service offering should be flexible enough to accommodate diverse customer requirements. Typically, the first step in the multinational management process is the storage of the SNF delivered to the hosting country. The final step being the disposal of the material in a deep geologic repository. This paper explores the ways and means of closing the gap between the first and last steps in the management process.

## 1. INTRODUCTION

In this paper, a multinational spent fuel management concept is proposed. The management concept is based on a service agreement entered into between countries, which intend participating in a common spent fuel (SNF) management venture. Accordingly, one of the participants in the venture would act as the hosting country, while the others fulfil the role of customer countries. The hosting country would agree to accept SNF from customer countries under specific conditions as required by the service agreement. The service agreement should preferably cover a number of options that customers can use such as storage, reprocessing or disposal. The service offering should be flexible enough to accommodate diverse customer requirements.

Typically, the first step in the multinational management process is the interim storage of the SNF delivered to the hosting country. The final step is the disposal of the material in a deep geologic repository. This paper explores the ways and means of closing the gap between the first and last steps in the multinational management process

## 2. THE MULTINATIONAL MANAGEMENT SYSTEM

Various studies [1, 2, 3, 4 and 5] have been done on multinational or regional radioactive waste and spent fuel management concepts. The proposal outlined in this paper, flows from work that has been carried out by an International Working Group [6] on multinational management systems. The aforementioned proposal is, however, taken somewhat further than the original concepts put forward by the Working Group.

The proposed multinational SNF management system, or MMS, is shown in Figure 1. The various options available to customers participating in the MMS are schematically illustrated in the above figure. The SNF management options catered for in the MMS are:

- ◊ SNF return to customer after interim storage;
- ◊ Long-term or perpetual SNF storage;
- ◊ SNF reprocessing;
- ◊ Semi-final disposal of SNF involving retrieval;
- ◊ Final disposal of SNF.

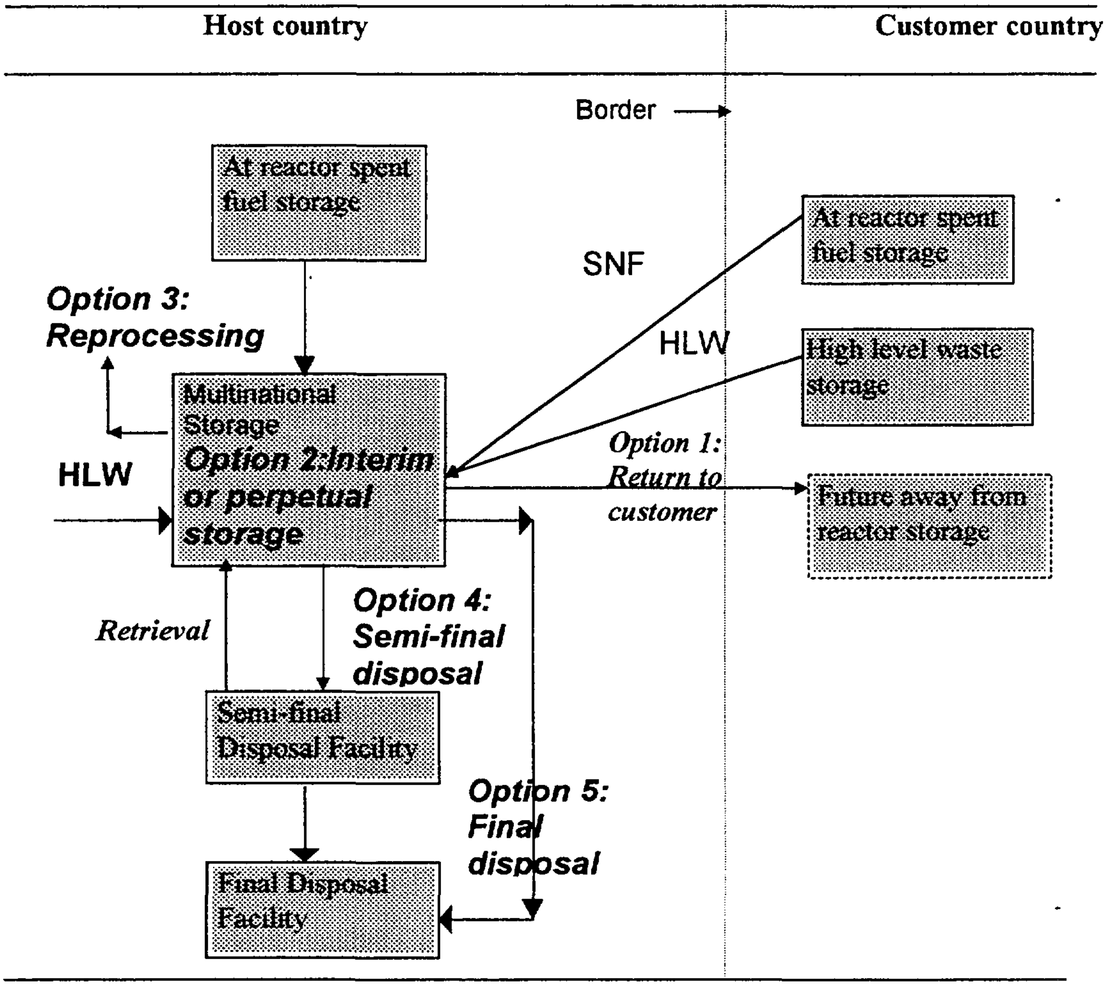


*FIG. 1. A multinational spent fuel management system*

## 3. ANALYSIS OF THE OPTIONS IN THE PROPOSED MULTINATIONAL MANAGEMENT SYSTEM

### 3.1. Technical considerations

The technology base for SNF transportation and storage is internationally well established. On the other hand, despite a great deal of repository research and development work world-wide, final *disposal* of SNF in deep geologic repositories still needs to be adequately demonstrated. In the case of the USA and Germany, repository sites have already been selected and are presently being developed. These projects are intended only for domestic SNF and HLW disposal. Target dates for the commissioning of these facilities are still tentative, but these repositories are generally not considered to be realizable before 2010. Other countries, such as Sweden, have accumulated a considerable body of know-how, but still lack approved sites.

Since there are not any deep geologic repository systems operating anywhere in the world, it is unlikely that hosting countries would step forward any time soon to make available their territories for multinational *disposal* purposes. On the other hand, multinational SNF storage is an established business in the sense that firms involved in reprocessing, already store large quantities of SNF on behalf their customers. It is therefore reasonable to assume that the motivation for hosting a multinational *storage* operation would be very differently from the motivation to host a multinational *disposal* operation. A hosting country that wishes to embark on a regional storage project should decide in advance what the final destination of the SNF would be. That is, the hosting country has to decide whether the SNF is to be returned to the country of origin at the end of the storage period, or whether the material is to be disposed of in the hosting country. The final destination issue could be resolved, for instance, by locating the regional storage facility on a site that has been pre-selected for regional disposal.

Alternatively, in stead of disposal, the SNF can be stored indefinitely, or "in perpetuity", without making provision for final disposal. The latter option may require repackaging of the stored materials during the long-term storage period envisaged. The establishment of regional storage facilities on the basis of perpetual storage of SNF has already been proposed in the case of the US Fuel and Security initiative. The perpetual storage option is shown in Figure 2.

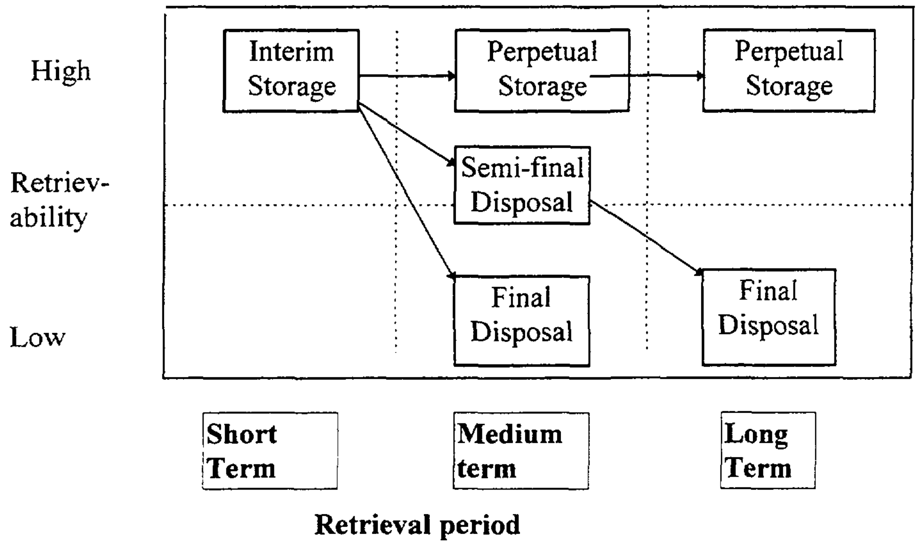


*FIG. 2. Retrievability of Spent Fuel*

At the end of the interim storage period the SNF can be disposed of in a *final* or *semi-final* repository. Semi-final disposal has the advantage of postponing the "final stage" of the disposal process by making provision for SNF retrieval from the repository. By pursuing this route, possible political objections to the "unproved" nature of the final disposal process could be overcome. The US repository at Yucca Mountain in Nevada, for example, will be kept open for a period of fifty years after completion of the repository emplacement operation. Semi-final disposal could be carried out either in a near surface or in a geologic repository. In the case of a near surface repository being used SNF would have to be retrieved *in any event* in order eventually to transfer the material to a geologic repository for final disposal. Whereas, if SNF were put into a geologic repository in the first place, the SNF need not be retrieved if the repository could be finally closed after emplacement. If such closure is not permitted for whatever reason, the material might have to be recovered from the geologic repository. These options are shown in Figure 2.

### 3.2. Economic considerations

Prolonged or perpetual storage offers an interesting economic alternative to final disposal in the sense that the expenditure associated with disposal is postponed indefinitely. However, where perpetual storage is the chosen approach, financial provision needs to be made for ongoing monitoring and maintenance costs. Furthermore, provision should also be made for the possible repackaging of the SNF in order to ensure that the integrity of the spent fuel containment is maintained. This latter operation is likely to be expensive.

Where the expenditure for disposal operations is indefinitely postponed there may also be an economic advantage in terms of discounted cash flow considerations. The advantage depends on

whether positive real discount rates prevail throughout the entire period under consideration. Unfortunately, real discount rate estimates for the next fifty-to-a-hundred years are of little value, and so are the project *present value* estimates based on these assumptions.
The most significant economic advantage of multinational waste management is the benefit of economy scale. The larger volume of material likely to be handled in a multinational system, compared to the usually smaller volumes of national management systems, would clearly reduce the unit cost of SNF management. Economy of scale is one of the strongest and obvious arguments for a regional approach to SNF management.

The economic implications of semi-final disposal, in comparison with the more conventional final disposal approach, need to be further investigated. The optimum cost of disposal would clearly be achieved in the case where the repository is kept open for the shortest period possible during the emplacement activity. Provision for the combined effect of SNF retrieval from, and the upkeep of the repository over the extended open period after cessation of repository operations, could be expected to add substantially to the overall costs of SNF management. Semi-final disposal should therefore be avoided unless politically imperative for the successful implementation of the multinational system.

Considerable economic advantages could be gained if all the management facilities associated with the SNF system were located on the same physical site. Such collocation would also have the added advantage of avoiding potential transportation problems on public roads between sites .

### 3.3. Financial considerations

In financial terms, the multinational SNF management project is characterised by an uneven cash flow situation. Typically, customers make payments for services at the beginning of the project, while the majority of the expenditure on facilities tends to occur toward the end of the project. The challenge, therefore, is to balance the non-coinciding revenue and expenditure patterns of the project. Balancing could be achieved by means of a waste management fund.

Various arrangements are possible whereby payments for service are made. In the extreme, a single payment can be made right at the beginning of the project for the entire SNF management service, including final disposal of the material. Alternatively, payments can also be made in installments on an annual basis for specific services rendered, such as interim storage for example. Arrangements between hosting and customer countries would therefore clearly depend on circumstances.

In the case where SNF is to be returned to the customer at the end of the storage period, payment might be required at the beginning of the storage service. The customer, in this case, retains full ownership of the material and also remains finally liable throughout the storage period. The hosting country is clearly the custodian of the material being kept in storage. Expenses incurred by the hosting country in providing the interim storage service are basically confined to the construction and operation of a storage facility.

In the case where the SNF is to be stored in perpetuity, ownership of the material should be transferred to the hosting country simultaneously with the transfer of the material. At this point, the customer should be called upon to pay in full for the service at the beginning of the storage period, since the perpetual storage option clearly involves a very long-term commitment on the part of the hosting country. Financial provision needs to be made for maintenance during perpetual storage, including future repackaging of the material. Such financial provision would best be made by way of the above-proposed waste management fund administered by the hosting country. The initial capital outlay required by the hosting country for perpetual storage should not significantly differ from the capital requirements for temporary storage.

In the case where semi-final disposal is adopted, the customer would tend to view the situation as being identical to that of perpetual storage. This view is justified because the customer will have to be relieved of his SNF responsibility in both cases. The hosting country would typically require full payment on transfer of the material for a comprehensive management service. The hosting country should make adequate financial provision for the timely construction of a conditioning plant and a repository where they do not yet exist. As stated above, it would be preferable for the hosting country to create a waste management fund to ensure compliance with all future responsibilities.

From the point of view of the customer countries it is immaterial whether the repository is a *semi-final* disposal facility (one which makes provision for the retrieval of material), or a *final* disposal facility (one which does not make provision for retrieval). It would be entirely up to the hosting country to satisfy its own regulators that the repository is safe, regardless of whether the repository needs to be operated as a semi-final or a final disposal facility.

### 3.4. Institutional considerations

The responsibility for the regulatory (i.e., safety) and safeguards (i.e., security) aspects of a multinational SNF management system needs to be clearly divided between the different countries participating in the joint venture. This approach is consistent with the principle of territorial sovereignty. The institutional requirement is that each participating country has to take full responsibility for the safety and security of the material as long as the latter physically resides in its territory. As the safety and security of the material only implies the exercising of physical control, ownership is unaffected.

The shipment of material on the high seas from customer to hosting country has to be carried out in accordance with internationally accepted principles, including the applicable international conventions and regulations. In the case where the material is moved through the territory of a consenting non-participating country, the latter will have to accept full regulatory and safeguards control of the material during the transfer operation. There is no need for a *supra-national* body assuming overall responsibility for the multinational management system, as the responsibility rests collectively with the individual countries. Transport operators will be employed to move the material from one country to another across international borders. International SNF transport has already been established in the reprocessing field, and should therefore not pose any technical or institutional difficulties in the case of a multinational management system.

It is important for the hosting and customer countries to introduce the legislative and /or administrative regulations for the implementation of the SNF storage and disposal. As this material will typically be stored for 40 to 50 years in the hosting country, adequate guarantees are required for contractual compliance. The same guarantees also need to be provided for the further steps of the SNF management process, such as conditioning and final disposal. Provision for a semi-final or final disposal system should be embodied in the institutional framework of the hosting country.

### 3.5. Political considerations

The successful implementation of the multinational waste management system requires a high level of political acceptance. Political acceptance is necessary at the national and the international level. At the national level, political acceptance is particularly important within the hosting, customer and affected third party countries. At the international level, political acceptance is necessary from the leading nations of the world. Furthermore, organisations such as the IAEA embodying international technical and institutional interests would also be required to put their stamp of approval on the undertaking.

When dealing with the political situation obtaining inside the hosting and customer countries, there are three levels that need to be considered, i.e., the *national* level, the *regional/provincial* level and the *local* level. Achievement of political and public acceptance at all three levels is essential. The strategy to be adopted in approaching government at these different levels is clearly country specific. In most democratic countries, the strategy would strongly rely on an open and effective public information campaign. Public debate should be carefully managed. Of primary importance is the need to demonstrate to the public what the level of risk would be, and how society will be protected against the potentially harmful effects of nuclear waste. Effective communication is required on sensitive issues, such as disclosure of the final destination envisaged for the SNF and the safety and security of the SNF in transit. Furthermore, the long-term safety of the multinational final repository is of great concern to the public, notably in view of the fact that an operational facility of this type does not yet exist anywhere in the world.

It is important to inform the public about the overall objectives of the multinational project to ensure that speculation about "undisclosed" or "hidden motives" is reduced to a minimum. This approach is particularly relevant to management strategies such as *storage in perpetuity* and *semi-final disposal.* The international and national political establishments should be involved in the development of these concepts right from the beginning, rather than being confronted with a *fait accompli* at an advanced stage of the project. The public should be made aware of the benefits of the project and the risks associated with storage and disposal of the materials.

### 3.6. Ethical considerations

If we assume that it is ethically justified for a country to dispose of its *own* SNF in a deep geological repository, the only remaining ethical issue is whether or not it would be justified for a country also to dispose of *anothe*r nation's SNF in such a repository.

Taken on purely rational grounds, the storage or disposal in any particular country of spent fuel originating from other countries should not pose an ethical problem. The difficulty with the free exchange of SNF among countries, so it seems, stems from the underlying supposition that SNF should be regarded as belonging to a "special category" that should not be traded internationally as a commodity. In this view, SNF is therefore to be considered a substance that should be exclusively dealt with by the generator of the material. The storage of SNF of international origin by reprocessors, however, is acceptable, where the SNF is waiting for reprocessing. But where the reprocessing link is missing, such storage is considered problematical and it has not yet been undertaken. SNF is either a waste or a resource depending on whether a future use is foreseen for the material. Accordingly, SNF has a negative value only after its owner has declared it as a waste, in which case SNF would be treated like HLW. Otherwise, SNF can be reprocessed and the useful constituents removed from it for future use.

Although the practice of nuclear waste *exchange* on the basis of radioactivity equivalence is already widely accepted internationally, the outright *sale* of nuclear waste (including SNF where declared as such) on the other hand appears to be ethically/politically problematic.

This restrictive approach is commercially untenable, as there "would always be a taker for waste in whatever form, provided the price is right". Although an expected market value for multinational SNF management services could theoretically be defined, there are nevertheless ethical conditions that should be satisfied before the concept could be implemented. The *criteria* that would ethically justify one country taking over another country's waste, could be summarised as follows

- *Equity or balanc*e between hosting and customer countries;
- *Burden* on future generations (mainly with regard to the hosting country);
- *Protection* of the interests of future generations;
- *Protection* beyond the national borders of the hosting country;
- *Transparency* in terms of safety and security issues.

## 4. THE TRENDS

The trends as far as the closing of the spent fuel storage/disposal gap is concerned could be summarised as shown in Table I.

## 5. SUMMARY AND CONCLUSIONS

An attempt has been made in this paper to examine the gap that exists between the storage and disposal activities in a multinational spent nuclear fuel management system. It appears from the analysis that storage, as an intermediate step in the overall spent fuel management process, is an attractive point at which the process could be *temporarily* terminated. Final disposal, as opposed to storage, is presently still in a developmental phase and should be seen in a different light to storage. However, in spite of the fact that progress with regard to repository development has been disappointingly slow - mostly due to non-technical reasons - there is nevertheless presently still little physical need for direct disposal. It therefore appears that, where reprocessing is postponed, storage, whether for interim or long term purposes is currently the indicated approach. These arguments favouring long-term storage of SNF in general could also be seen as favouring long-term regional storage.

TABLE I. THE SPENT FUEL STORAGE/DISPOSAL INTERFACE

| ASPECT | FACTORS *PROLONGING* STORAGE PERIOD | FACTORS *REDUCING* STORAGE PERIOD |
|---|---|---|
| **Technology** | • Longer ***cooling*** periods required for higher burnup UOX & MOX fuel<br>• ***Reliability*** of long term storage technology | • Need to ***demonstrate*** successful closing of the back-end<br>• Need for spent fuel ***repackaging*** during perpetual storage |
| **Economics** | • Advantage of ***discounted cash flow*** at positive real interest rates<br>• ***High cost*** of reprocessing<br>• ***Uncertainty*** about future value of Pu | • ***Commitment*** by certain nations to reprocess<br>• Existing reprocessing ***contracts*** |
| **Politics, Ethics & Public Acceptance** | • Public ***opposition*** to disposal<br>• Perpetual storage is an ***alternative*** to final disposal | • ***Protection*** of future generations<br>• ***Reducing*** the burden on future generations |

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IAEA-SM-352/44

# CONDITIONING OF SPENT FUEL FOR INTERIM AND FINAL STORAGE IN THE PILOT CONDITIONING PLANT (PKA) AT GORLEBEN

XA9951812

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**Abstract**

In 1994, due to the change of the nuclear law in Germany, the concept of direct final disposal for spent fuel was developed as an equivalent alternative to the waste management with reprocessing. Since 1979, tests for the direct final disposal of spent fuel have been conducted in Germany. In 1985, the State and the utilities came to an agreement to develop this concept of waste management to technical maturity. Gesellschaft für Nuklear-Service (GNS) was commissioned by the utilities with the following tasks: to develop and test components with regard to conditioning technology, to construct and operate the pilot conditioning plant (PKA), and to develop casks suitable for final disposal. Since 1990, the construction of the PKA has taken place at the Brennelementlager Gorleben site. The PKA has been designed as a multipurpose facility and can thus fulfil various tasks within the framework of the conditioning and management of spent fuel assemblies and radioactive waste. The pilot character of the plant allows for development and testing in the field of spent fuel assembly conditioning. The objectives of the PKA may be summarized as follows: to condition spent fuel assemblies, to reload spent fuel assemblies and waste packages, to condition radioactive waste, and to do maintenance work on transport and storage casks as well as on waste packages. Currently, the buildings of the PKA are constructed and the technical facilities are installed. The plant will be ready for service in the middle of 1999. It is the first plant of its kind in the world.

## 1. INTRODUCTION

In Germany the State and the utilities came to an agreement in 1985, to develop the concept of direct final disposal of spent fuel to technical maturity. The State has in this context been assigned the task of constructing a final repository and of executing the necessary development work.

GNS is in charge of PKA project management and has set up a project team in Hannover. The PKA is being constructed on the Brennelementlager Gorleben (BLG) site. This company, owner and operator of the Gorleben intermediate-storage facility, is the building owner and future operator of the PKA. BLG is a wholly owned subsidiary company of GNS. In April 1990, BLG commissioned a syndicate, consisting of NOELL/Rueterbau and Steag Kernenergie, with the construction and cold commissioning of the plant.

## 2. OBJECTIVES OF THE PKA

The PKA has been designed as a multipurpose facility and can thus fulfil various tasks within the framework of the conditioning and management of spent fuel assemblies and radioactive waste. The pilot character of the plant allows for development and testing in the field of spent fuel assembly conditioning. The throughput of the plant is limited to 35 t of heavy metal per year, because there is only one shift.

The objectives of the PKA may be summarized as follows:

- Conditioning of spent fuel assemblies for interim and final storage;
- Reloading of spent fuel assemblies and waste packages;
- Conditioning of radioactive waste;
- Maintenance work on transport and storage casks as well as on waste packages.

## 3. THE TECHNICAL CONCEPT OF THE PKA

Conditioning is taken to mean in this context, the packaging of spent fuel assemblies in casks suitable for final disposal and for long-term intermediate storage. There are 3 different methods for conditioning of spent fuel (Fig. 1):

a) The reference way, which is resulting in the self-shielded POLLUX cask;
b) The way of the future to the not self-shielded Fuel-Rod-Kokille 3 (BSK 3) without cutting the fuel rods;
c) The optional way to the short not self-shielded POLLUX canister with cutting of the fuel rods.

To begin, similar procedural steps will be followed for all methods of conditioning as follows:

a) Delivery of the spent fuel assemblies in transport casks or transport and storage casks;
b) Unloading of the casks and storage of the spent fuel assemblies in an on-site buffer store;
c) Detachment of the head and base plates and removal of the fuel rods in layers.

From this point on, different procedural steps are followed for each of the three methods.

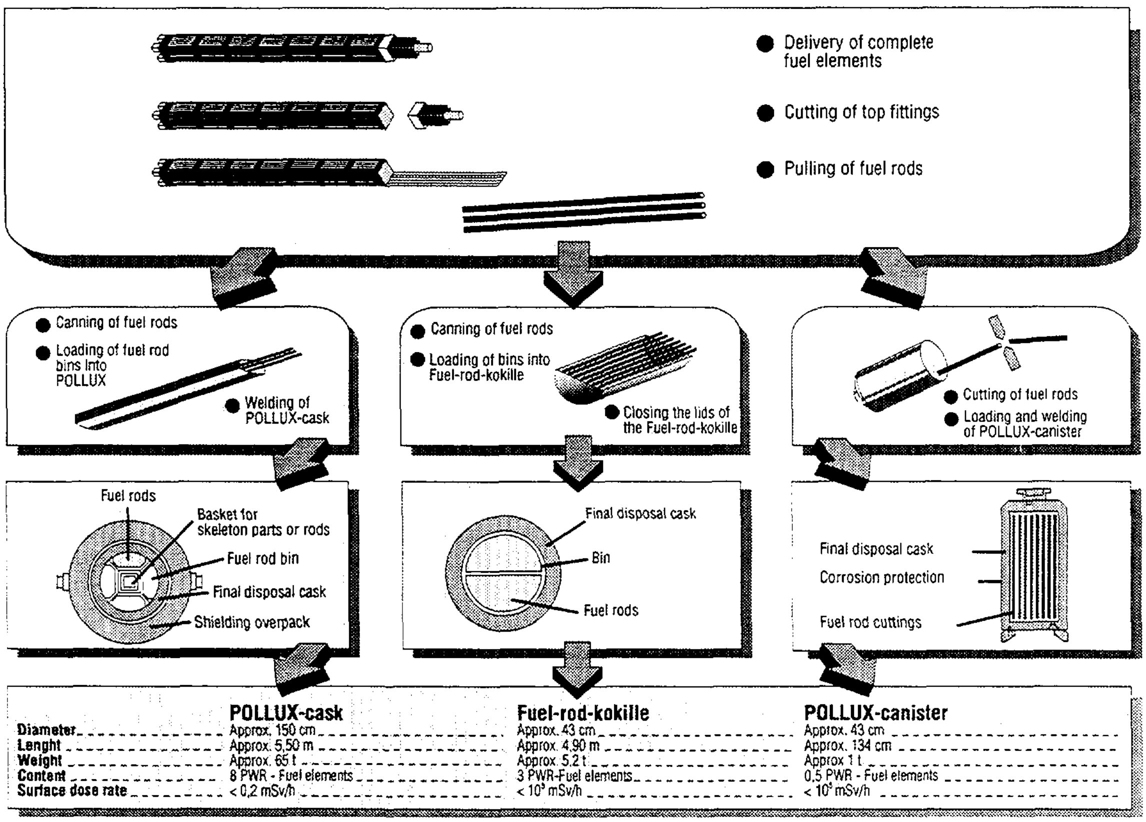


*FIG. 1. Conditioning methods in the PKA*

### 3.1. The POLLUX-way

The POLLUX has a capacity of 8 or 10 pressurized water reactor (PWR) fuel assemblies or 24 or 30 boiling water reactor (BWR) fuel assemblies. There are four bins and one basket in each POLLUX cask. The weight of the cask is ≈ 65 t.

The steps of this conditioning procedure are as follows:

a) Loading complete fuel rods into bins (from two PWR or six BWR fuel assemblies to one bin);
b) Compacting the structural parts of the fuel assemblies and loading the compressed parts into the basket;
c) Loading the bins and the basket into the POLLUX cask;
d) Closing of the POLLUX cask, i.e. screw on the primary lid, welding on the secondary lid and screwing in the shielding lid.

As an alternative of this procedure, the basket may also be loaded, e. g., with the fuel rods from two PWR fuel assemblies.

### 3.2. The way to the „Fuel-Rod-KOKILLE BSK 3"

The steps of this conditioning procedure are as follows:

a) Loading complete fuel rods into bins (from three PWR or nine BWR fuel assemblies to one bin);
b) Compacting the structural parts of the fuel assemblies and loading the compressed parts into drums;
c) Loading the two bins into the BSK 3;
d) Closing of the BSK 3, i.e., screw on the primary lid and welding on the secondary lid;
e) Loading the BSK 3 into an overpack cask (for example, into a CASTOR-type transport and storage cask).

### 3.3. The way to the POLLUX canister

The steps of this conditioning procedure are as follows:

a) Cutting the fuels rods into segments of ≈ 1 m;
b) Loading the fuel rod segments into POLLUX canisters (from 0.5 PWR fuel assembly or 1.5 BWR fuel assemblies to one canister);
c) Closing the POLLUX canister, i.e., insert the primary plug and weld on the secondary plate;
d) Loading the canisters into an overpack cask (for example, into a CASTOR-type transport and storage cask).

The structural part of the spent fuel assemblies are also compacted and loaded into drums or other containers.

### 3.4. Conditioning for the interim storage

For a better efficiency during the interim storage period of spent fuel, the PKA is also able to fulfil the following functions:

- to reload fuel assemblies from smaller to bigger interim storage casks; or
- after some years of cooling time consolidation of fuel assemblies for further storage in interim storage casks.

### 3.5. Fuel assembly conditioning procedure

Using the procedure of packaging LWR spent fuel assemblies in POLLUX casks as an example, the technical possibilities of the plant can be described in a clearcut manner (Fig. 2). The transport cask is lifted into the position A2 at the cask hall, here the cask is prepared for unloading. Then it is lowered into the trolley, taken into the cell area, and docked to the unloading cell (B1).

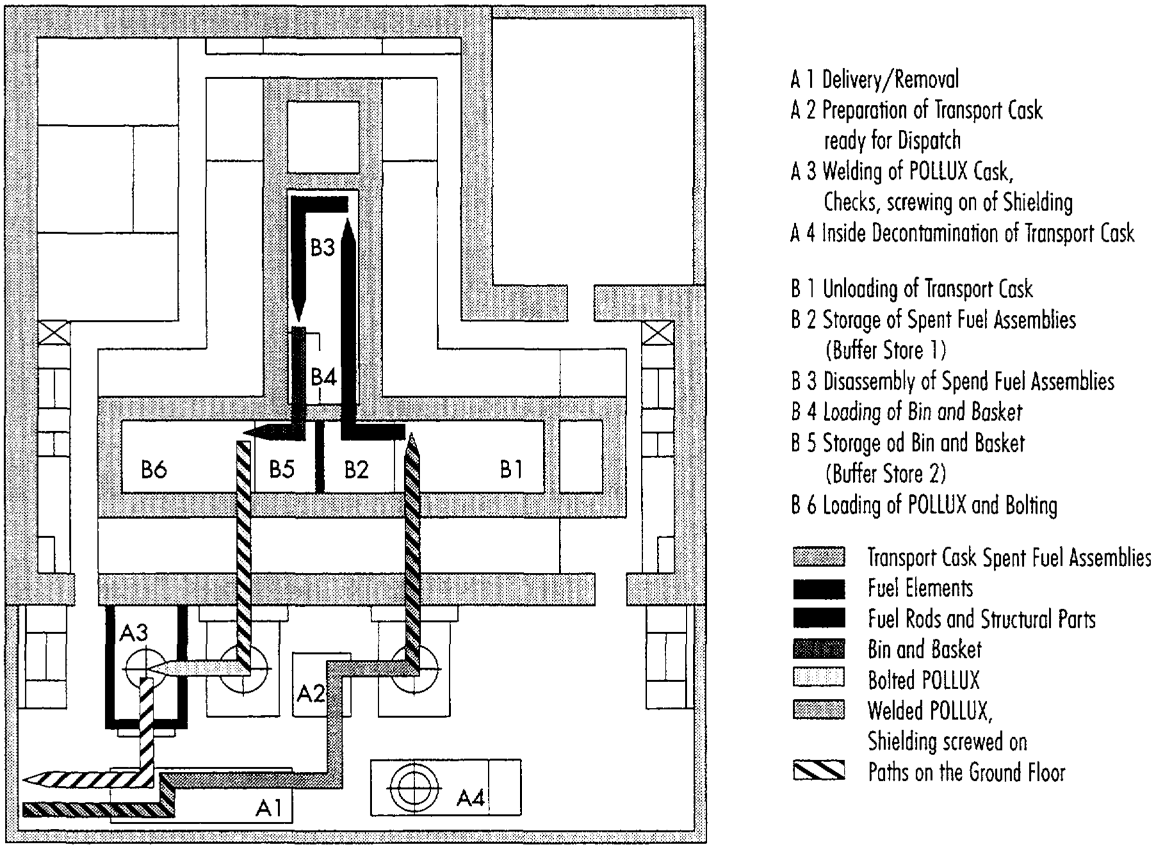


*FIG. 2. Loading of LWR spent fuel assemblies into POLLUX casks*

The spent fuel assemblies are transferred from the transport cask by the cell crane to buffer store 1 (B2). Then, the spent fuel assembly is drawn into the disassembly cell and placed horizontally on the disassembly table (B3). Here the fuel assembly is positioned ready for the detachment of the head plate and the removal of the fuel rods. The fuel rods removed are pushed into the bins of the POLLUX cask and the compressed structural parts are pushed into the basket by way of a pushing device. To this end, the bins and the basket have been docked inside a loading caisson (B4) to a lock. The loaded bins and the basket are transferred to the loading cell (B6) and taken to buffer store 2 (B5) by way of the cell crane.

Simultaneously, the POLLUX cask to be loaded is transferred and docked at the loading position B6. The filled bins and the basket are loaded by cell crane into the POLLUX cask. Then, the first lid is screwed tight and the cask is filled with helium. The final conditioning of the POLLUX cask is done at a workstation (A3) in the cask hall by direct handling. Here the secondary lid is inserted and welded tight to the inner cask, and the weld is ultrasonic checked. Following the screwing on of the shielding lid, the POLLUX package is complete. Now the POLLUX is ready for transport to an interim or final storage.

### 3.6. Cask maintenance

By way of its hot cell concept the PKA provides the opportunity to do maintenance work on all currently available transport and storage casks, both empty and loaded.

The following work may thus be done:
a) the exchange of primary lid gaskets of transport and storage casks;
b) the cleaning and maintenance of both empty and loaded casks;

c) reworking, such as the opening and repairing of POLLUX casks or of transport and storage casks that have been provided with a welded-on rabbeting lid.

### 3.7. Conditioning facilities

The main facilities, especially those to be used to condition spent fuel assemblies and radioactive waste, are located within the cells (Fig. 3). Some of these facilities, particularly those to be used for conditioning, have been developed especially for use in the PKA and have been tested in cold test stands.

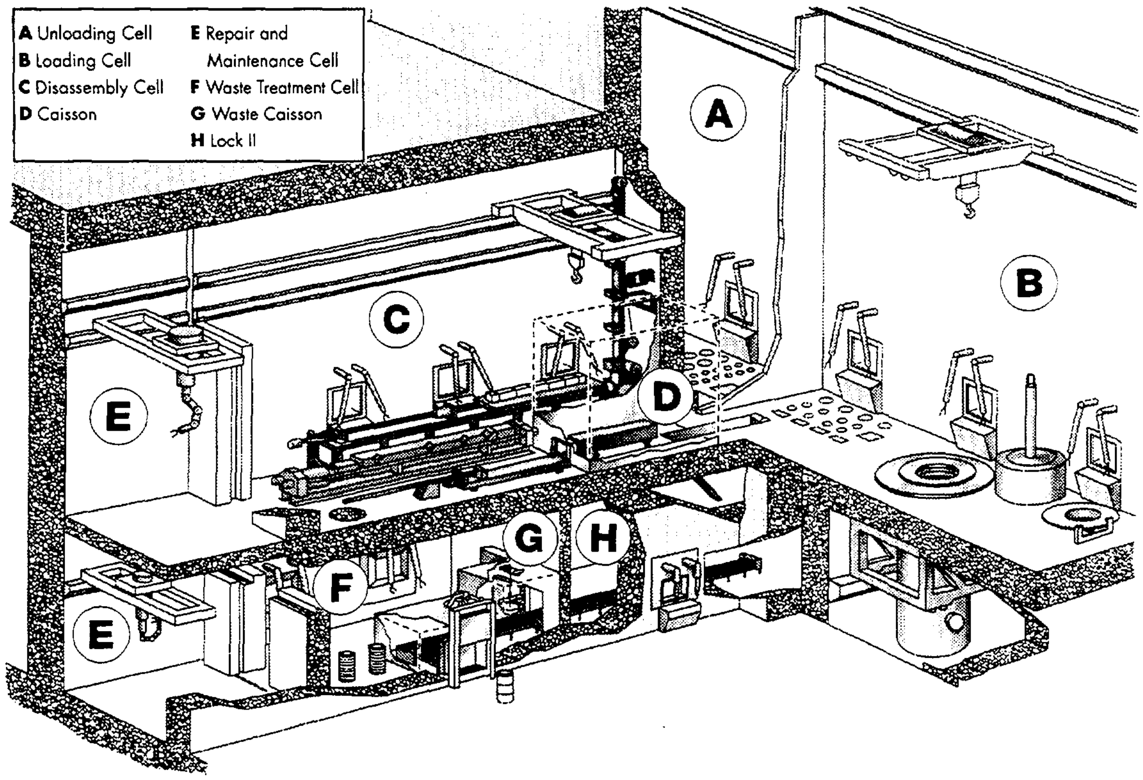


*FIG. 3. Cell area in the PKA*

The main facilities are:

a) In the loading and unloading cell (A and B)
   - the cell docking facility for the loading and unloading of various types of casks;
   - the lid removal equipment, to allow uncontaminated removal of the cask's primary lid;
   - buffer stores 1 and 2 with storage positions for 12 PWR spent fuel assemblies and 25 BWR spent fuel assemblies;

b) In the disassembly cell (C)
   - disassembly equipment, consisting of a disassembly table horizontally trammable, head detachment device, drawing tools, and through vibration gear;
   - compacting device for compacting the structural parts of a complete spent fuel assembly;
   - pushing device to allow horizontal pushing of the fuel rods into a POLLUX bin;
   - loading caisson (D) and equipment (turntable and dual-lid system) to allow the uncontaminated docking of the fuel rod bins and canisters as well as the basket for structural parts;

c) waste treatment cell (F): equipment for cutting and compacting solid radioactive waste.

## 4. THE FINAL DISPOSAL CASK POLLUX

Fig. 4 shows the basic design of the final disposal cask. The cask consists of the shielding cask with an screwed-in lid and the inner cask with bolted primary and welded secondary lid. The material to be stored is inserted in the final disposal cask in bins. Fig. 5 shows in detail the constructive set-up of the final disposal cask.

The cylindrical wall and bottom of the inner cask is extruded in one piece made of fine-grained steel 15 MnNi 6.3. The thickness of the cylindrical wall is designed according to the mechanical and shielding requirements and is 160 mm. This inner cask with contents has a weight of ~31 t when closed. The primary lid of the inner cask is made of the same material as the base body. It assumes the sealing function prior to and during the welding of the secondary lid. A plate made of neutron-moderating and absorbing materials (carbon/boron mixture) is attached on the underside of the primary lid.

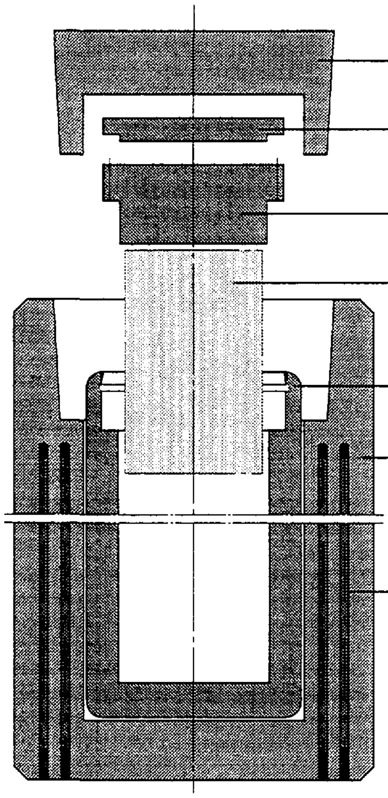


*FIG. 4. Concept of the POLLUX cask*

The secondary lid is designed as a welded lid and made of the same material as the base body. The ~50-mm-thick welded connection is produced by the narrow-gap welding procedure and forms the leak-tight and permanent barrier for transport, interim storage, and final disposal of the spent fuel. The base body of the shielding cask is cast in one piece made of ductile cast iron (GGG 40). The wall thickness of the base body is designed according to the requirements on the shielding and is 265 mm on the side wall. The weight of the shielding cask is ~34 t. The prototype cask for cold handling in the PKA is fabricated and is already used for this purpose.

To simulate the transport and accident conditions, five drop tests, which cover all possible accidents, were performed with a full-scale prototype of the final disposal cask and „finite element" calculations have been carried out. The drop tests consist of:

a) Horizontal 9-m drop with shock absorbers onto the sidewall;
b) Vertical 9-m drop with shock absorbers onto the top end;
c) 9-m drop with shock absorbers onto the cask corner (lid area);

d) 5-m drop onto a concrete foundation without shock absorber onto the sidewall;

e) 5 m-drop onto a concrete foundation without shock absorbers onto the cask bottom.

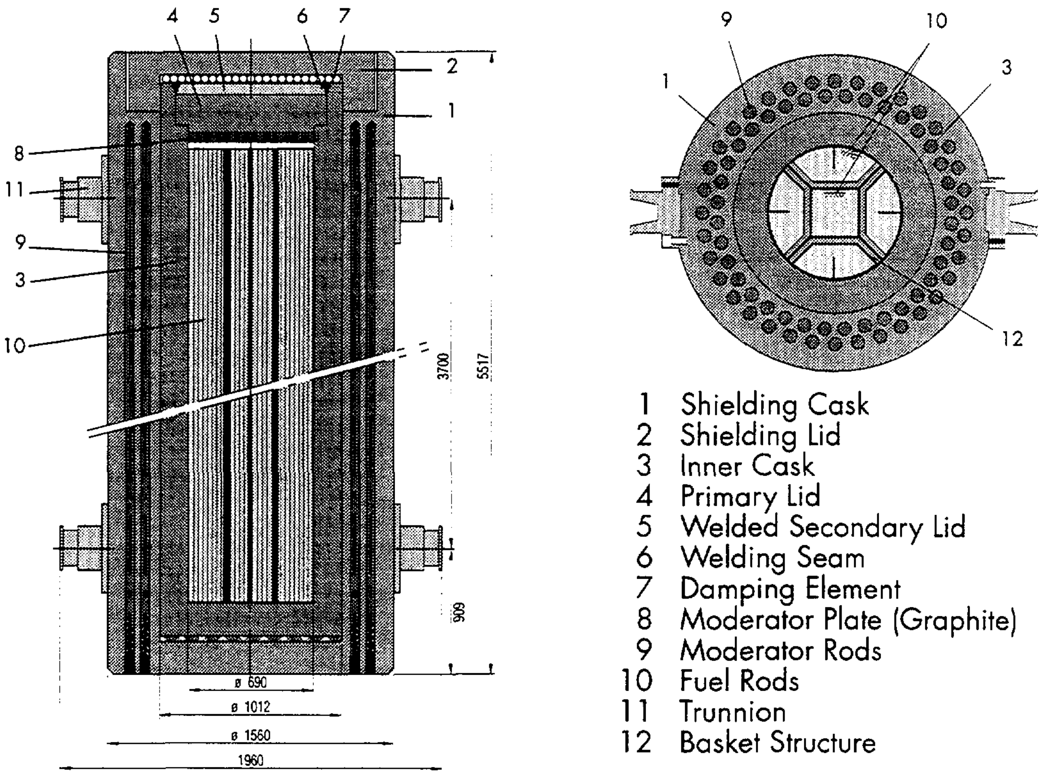


*FIG. 5. Final disposal cask POLLUX*

## 5. THE "FUEL-ROD-KOKILLE BSK 3"

An improved new design is the so-called "Fuel-Rod-Kokille BSK 3" (Fig. 6). It is designed for the accommodation of consolidated fuel rods. The following design properties characterize this final disposal container:

a) The outer shape and the diameter correspond to the high-active waste (HAW) canisters for vitrified high-level radioactive wastes from reprocessing. This makes possible a common handling and final disposal of both waste canisters and fuel containers in bore hole storage;

b) The capacity of the inner cavity for the BSK 3 is designed for the accommodation of fuel rods from three PWR fuel assemblies or nine BWR fuel assemblies in two bins;

c) The resulting wall thickness of 50 mm of the Fuel-Rod-Kokille are sufficient to keep the forces from the rock pressure under control in the final disposal according to current knowledge. Should these requirements increase, a filling of the empty space of the canister with a suitable material would also be possible;

d) The closure of the Fuel-Rod-Kokille after loading is in accordance with the approved welding technique, for the POLLUX cask;

e) After loading and welding , seven BSK 3 are inserted for interim storage in a modified CASTOR HAW 20/28 transport and storage cask originally designed for HAW glass containers.

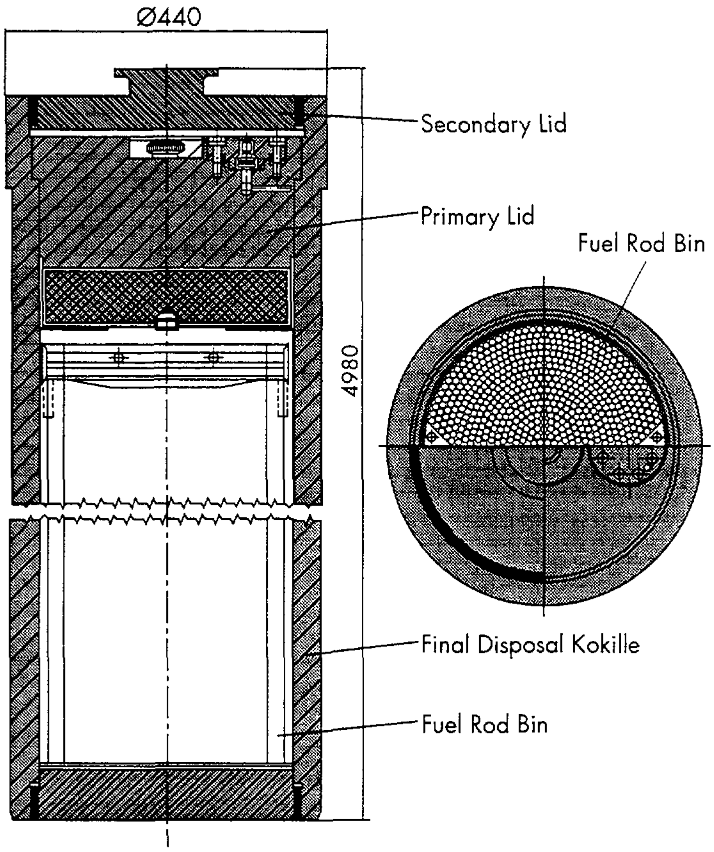


*FIG.6. Fuel-Rod-Kokille BSK 3*

## 6. CONSTRUCTION OF THE PKA

The building of the PKA are completed; at present, the technical facilities are installed with a license based on the atomic law. In the middle of 1999 the plant will be ready for hot operation. Fig. 7 shows a view into the dismantling cell and Fig. 8 the main building of the PKA.

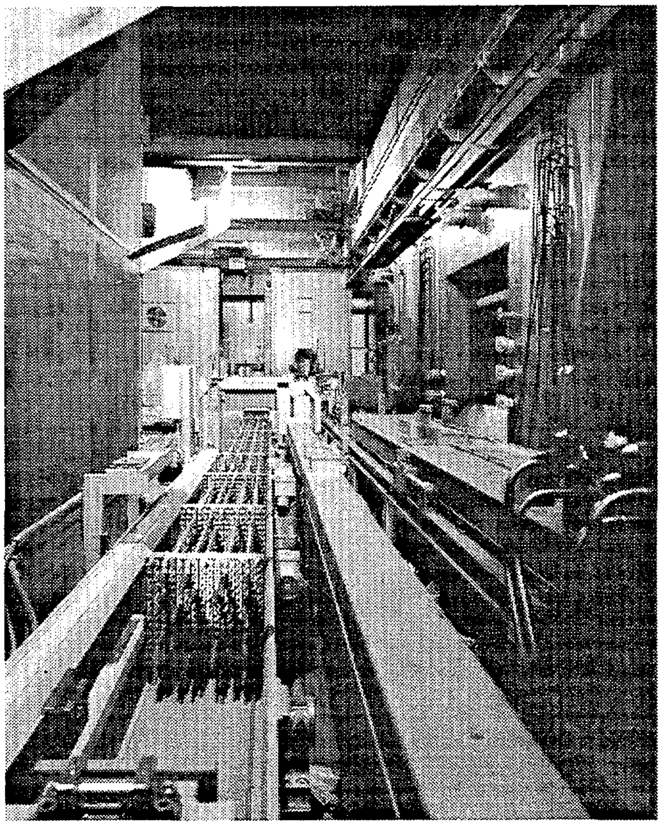

*FIG. 7. Dismantling cell in the PKA*

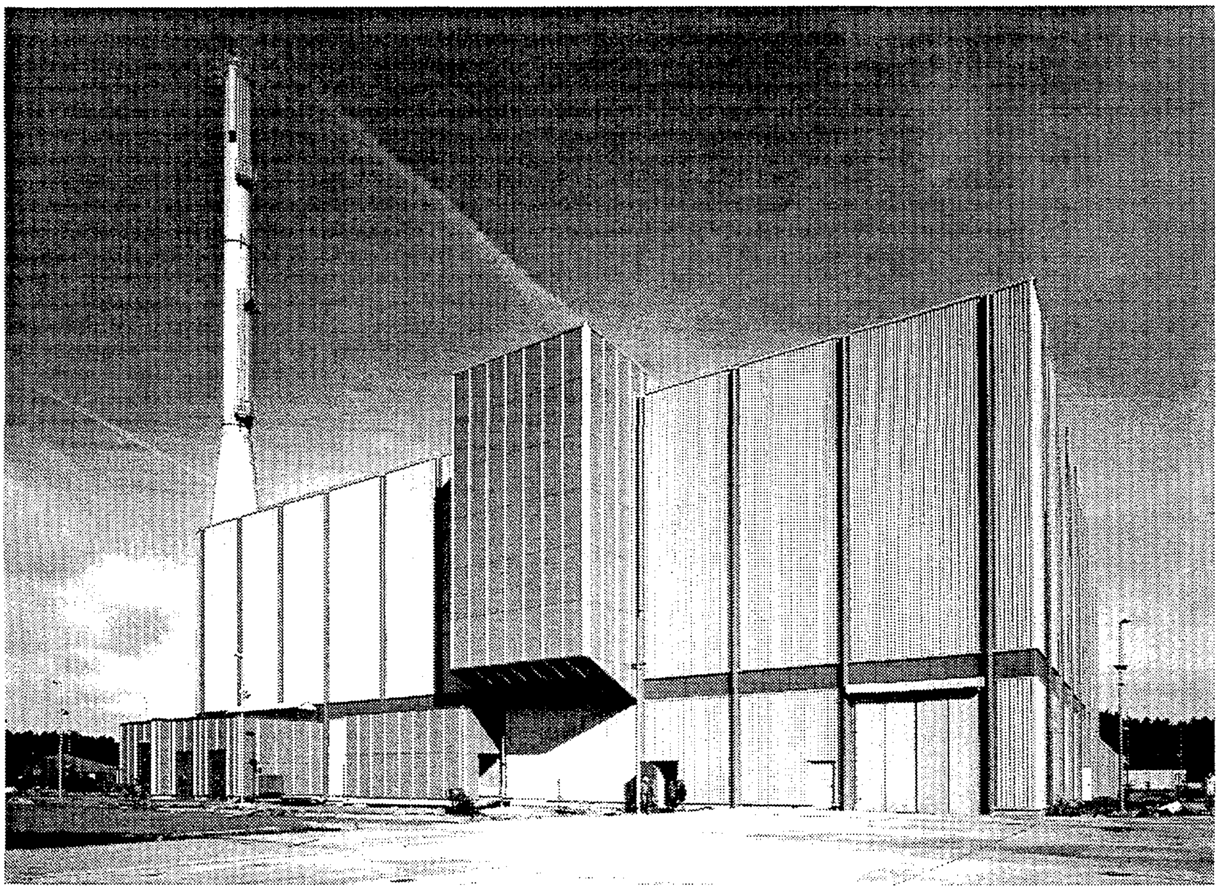

*FIG. 8. Main building of the PKA*

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XA9951813

# THE SWEDISH PLANS FOR ENCAPSULATION OF SPENT NUCLEAR FUEL

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**Abstract**

In Sweden, spent nuclear fuel will be encapsulated in disposal canisters before deposition in a deep repository. Each canister consists of an outer copper canister for corrosion resistance and a cast iron insert for mechanical strength. Trial manufacturing has shown that full-size disposal canisters can be produced according to specifications. An encapsulation plant is planned to be built as an extension to the CLAB interim storage where the fuel is stored today. In the plant, the fuel will be placed in disposal canisters which will be welded and tested before transfer to the deep repository. The crucial sealing operations will be tested and demonstrated in the Canister Laboratory. The welding trials in the laboratory are scheduled to start in late 1998.

## 1. INTRODUCTION

In Sweden, there are 12 nuclear reactors in operation at 4 different sites. These reactors produce around 50% of the electricity used in Sweden. By the year 2010 the nuclear program will have generated approximately 8000 tons of spent nuclear fuel. During the past 25 years a comprehensive system for radioactive waste management has been developed. The system is owned and managed by the Swedish Nuclear Fuel and Waste Management Co (SKB). SKB is owned by the Swedish nuclear power utilities.

At present, the spent fuel is stored at a central interim storage facility (CLAB) located near the Oskarshamn Nuclear Power Plant, see Figure 1. CLAB consists of a receiving building at ground level and a storage building in a rock cavern, approximately 25 meters below ground. The fuel assemblies are stored in water pools in the storage building, in special storage canisters. The construction of a second storage building started in September 1998. The new storage pools are scheduled to be in operation by the year 2004.

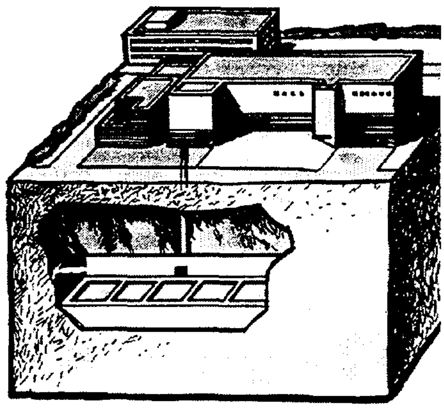

*FIG. 1. The central interim storage facility for spent fuel (CLAB) in Oskarshamn*

After 30-40 years of interim storage, the spent fuel will be encapsulated in corrosion resistant disposal canisters. Encapsulation of the fuel will take place in an encapsulation plant which is planned to be built as an extension to CLAB. The encapsulated fuel will be transferred to a deep repository where the canisters will be deposited approximately 500 meters down in the Swedish bedrock, surrounded by bentonite clay, see Figure 2.

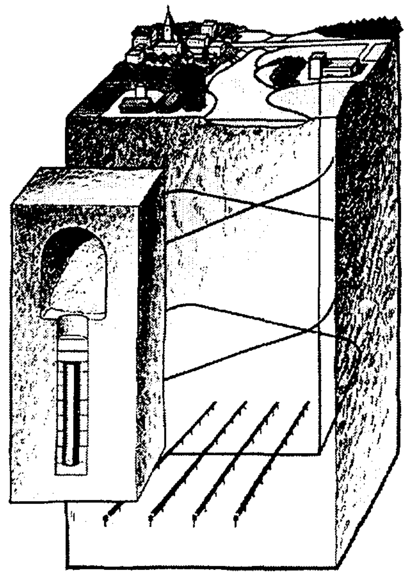

*FIG. 2. In the repository each canister is placed in a deposition hole and is surrounded by bentonite*

## 2. DISPOSAL CANISTER

The most important function of the disposal canister is to isolate the spent fuel in the deep repository. In order to retain integrity in the repository the canister must have:

- initial integrity;
- corrosion resistance;
- mechanical strength.

The design of the disposal canister is determined by these requirements and by the different safety requirements during encapsulation, transport and deposition. In addition, it must be possible to manufacture approximately 200 canisters per year according to specifications.

Throughout the years, SKB has studied several different canister alternatives. Due to the oxygen-free environment which prevails at the repository level, copper has been determined to be the most suitable canister material for corrosion resistance, see [1,2]. To improve the mechanical strength, the canister contains a cast iron insert with channels for the fuel assemblies. Each insert can hold either 12 BWR or 4 PWR fuel assemblies. The design of a disposal canister for BWR fuel is shown in Figure 3. In the current design, the wall thickness of the copper is 50 mm and the minimum wall thickness of the insert 50 mm. The canister diameter is 1050 mm, the length 4850 mm and the weight, with spent fuel, approximately 27 tons.

In order to meet the corrosion requirements the wall thickness of the copper needs to be at least 15 mm, see Ref [3]. When the requirements on mechanical strength, manufacturing and handling are considered as well, a 30 mm wall thickness appears to be suitable. If the thickness of the copper is decreased, the wall thickness of the cast insert needs to increase, to ensure that the dose rate will not be so high that it could affect the other barriers in the deep repository system.

To develop technology for manufacturing canisters, SKB has tested several different methods. In Figure 4 the first trial manufactured full-size canister is shown. Full-size copper cylinders have been manufactured using three methods: extrusion, pierce and draw, and forming from rolled plate and electron beam welding the two halves together. All three methods have shown satisfactory results although additional development work is required. Canister inserts have also been cast in full scale. Just as for the copper cylinders, the initial trials have been satisfactory but development work on manufacturing methods needs to continue throughout the coming years.

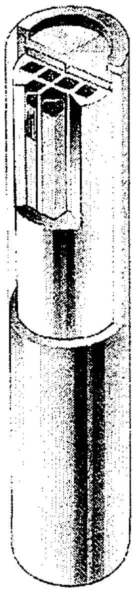

*FIG. 3. Copper canister with cast iron insert for BWR fuel*

The requirement on initial integrity of the canister puts high demands on the technology for sealing and testing the canisters. Together with TWI in England, SKB has developed the electron beam welding equipment which is planned to be used for sealing the canisters in the encapsulation plant. For non-destructive testing of the welds, the main methods are ultra sonic and X-ray testing, although other methods are being studied as well.

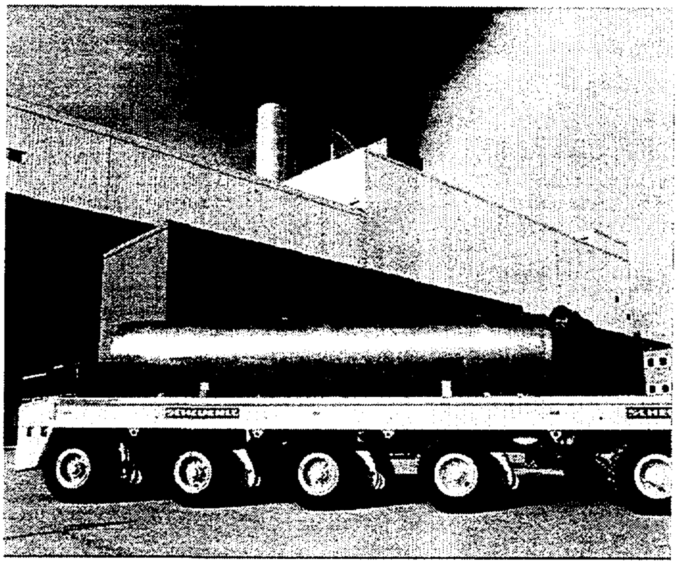

*FIG. 4. The first trial manufactured canister in full scale was delivered to SKB in February 1996*

Before the deep repository is in full operation SKB needs to build a factory for producing empty canisters for delivery to the encapsulation plant. SKB has so far conducted a study on the possible layout of such a production facility. The facility is designed to produce more than 200 canisters per year and will contain equipment for forming copper cylinders, machining, quality control and final assembly of disposal canisters.

## 4. ENCAPSULATION PLANT

### 4.1. Siting

The encapsulation plant can be sited at CLAB, the deep repository, an existing nuclear facility or somewhere else. SKB currently plans to build the encapsulation plant right next to CLAB since that provides several advantages, see [2]. The advantages of siting the encapsulation plant at CLAB are:

- Transportation to the encapsulation plant will be easier since the storage canisters with fuel can be transferred directly without using transport casks;
- Several existing service systems in CLAB, e.g. cooling systems, water purification systems and electrical power supply, can be extended to also service the encapsulation plant;
- There is access to other nuclear engineering infrastructure such as dosimetry and waste handling;
- The experience of spent fuel handling and operation and maintenance of associated service systems possessed by the personnel at CLAB can best be put to use if the facilities are on the same site;
- Transportation to the deep repository will be easier since the spent fuel is encapsulated in sealed canisters;
- The encapsulation plant can be accommodated within SKB's CLAB property site. The environmental impact will therefore be minimal since new land does not have to be used, and no new roads or cooling water installations will be needed.

Before an application for construction of the encapsulation plant will be submitted SKB plans to also study the design of an encapsulation plant at a site away from CLAB. The time when the application will be sent in is dependent on the progress of siting the deep repository. SKB expects to submit the application in the year 2002, at the earliest. It is planned to take about six years between start of construction and encapsulation of the first fuel.

### 4.2. Plant design

The encapsulation plant shall be designed to produce high quality disposal canisters with spent fuel. Safety of personnel and people in the surrounding area shall be taken into account in the design of the plant. The plant shall be designed for an annual output of 200 disposal canisters. At a first stage only spent fuel will be encapsulated but preparations shall be made for the later addition of equipment for final conditioning of core components.

Design work on the encapsulation plant started in 1993 and has resulted in a Basic Design which will form the basis of the application for construction of the plant. BNFL Engineering Ltd has designed the encapsulation process and ABB Atom AB the auxiliary systems.

The main parts of the plant will be an encapsulation building, where the spent fuel will be encapsulated, and a storage building for filled transport casks. The plant will be approximately 65 x 80 meters in size and about 25 meters high, which is equivalent to the height of the existing receiving building at CLAB, see Figure 5.

Before the application for construction is compiled, SKB plans to review the Basic Design further. The plant design also needs to be updated as experience is gained from the Canister Laboratory which is described in Section 5.

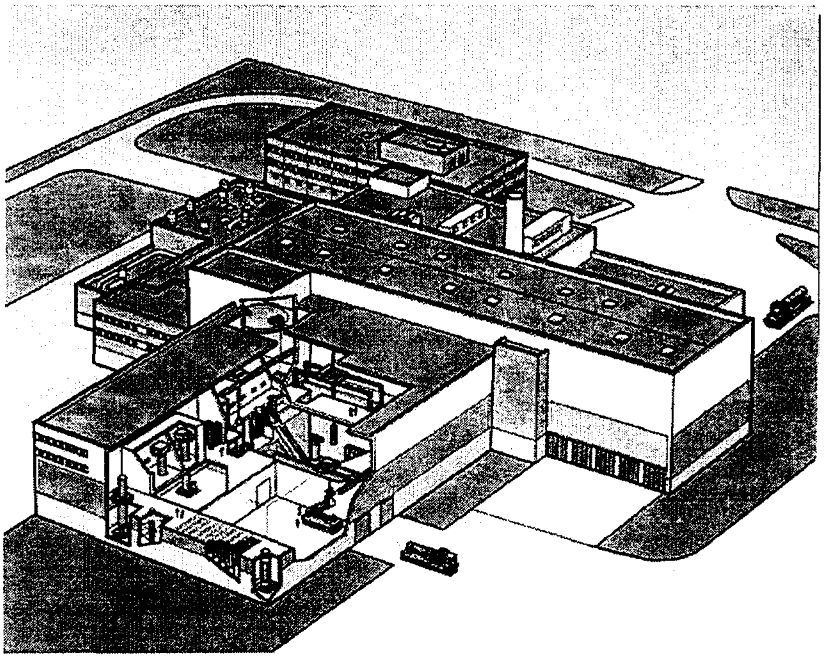

*FIG. 5. The encapsulation plant built as an extension to CLAB*

### 4.3. Encapsulation process

When empty disposal canisters arrive at the encapsulation plant from the canister factory they are checked thoroughly before they are introduced in the encapsulation process. The spent fuel assemblies are transferred in their storage canisters from the storage pools in CLAB to pools in the encapsulation building, using the existing fuel elevator. A fuel handling machine above the pools is used to identify and transfer the assemblies to a transfer canister. A transfer canister is similar to a storage canister but can only hold 12 BWR or 4 PWR fuel assemblies, i.e. the same number of assemblies that a disposal canister can hold. During transfer in the pools, the fuel can pass a fuel monitoring station where gamma measurements are made in order to calculate e.g. burnup and residual power. When a transfer canister is filled, an inclined elevator brings it up, out of the water, see Figure 6.

At the top of the inclined elevator the transfer canister is first allowed to drain after which it is lifted over to a handing cell with a cell crane, see Figure 7. In the handling cell, the transfer canister is placed in one of two drying station. There, the fuel is dried with hot re-circulating air. After drying, the assemblies are lifted, one by one, to a disposal canister which is docked to the handling cell from below. The connection between the disposal canister and the handling cell is air tight so that the outside of the canister is not contaminated and the air in the cell does not escape and cause airborne activity in other parts of the plant. A steel lid is bolted to the insert before the canister is transferred for further processing. The empty transfer canister is brought back, via the inclined elevator, to the pools where it again will be filled with fuel.

At the handling cell, the disposal canister is standing in a shielded frame which is used for transfer of the canister between the different stations in the encapsulation building. The shielded frame is lifted and transferred using a remote controlled air film transporter. From the handling cell the shielded frame with the disposal canister is taken to a station where the air in the insert is exchanged with argon to limit the amount of air in the canister. The air is removed and the argon filled through ports in the steel lid. Before the canister leaves the station the tightness of the steel lid is checked. At the following station, the welding station, the canister is docked to a welding chamber within the station. When the canister is connected, the chamber is vacuumed down and a copper lid is placed on the copper canister. The lid is then sealed to the canister using electron beam welding. During welding the canister is rotated and the welding equipment is fixed.

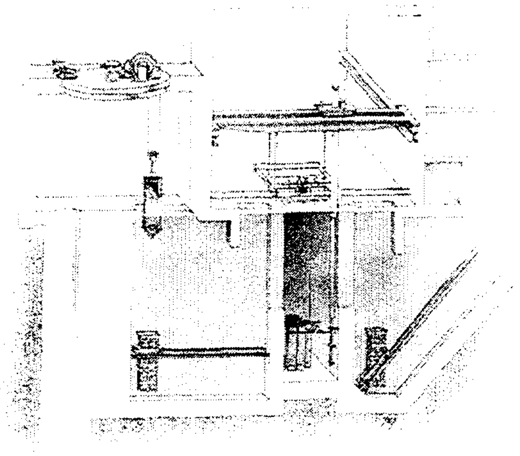

*FIG. 6. Fuel elevator and pools in the encapsulation building*

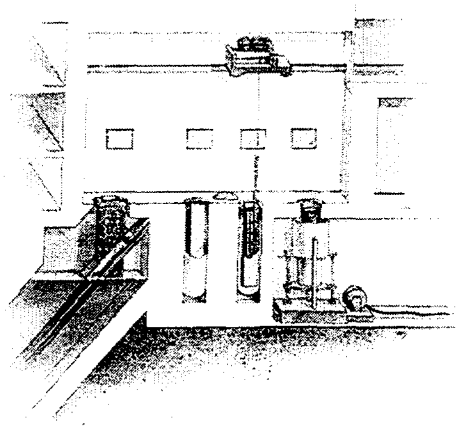

*FIG. 7. In the handling cell the fuel assemblies are dried and transferred to a disposal canister*

After welding, the canister is transferred to a separate station for non-destructive testing and machining. Non-destructive testing is performed using both ultra sonic and X-ray techniques. If the weld contains defects which are repairable, the canister is brought back to the welding station for re-welding. In case a weld can not be repaired the copper lid is removed, the steel lid unbolted and the fuel, finally, unloaded in the handling cell.

When a canister has passed the non-destructive testing it is lifted out of the shielded frame, using a remote controlled shielded handling machine, and is transferred to a station for monitoring and, if necessary, decontamination. The canister is lowered down into the station where smear tests are taken on the entire outer surface to monitor that it is clean. The station is equipped with high pressure water in the event that a canister needs to be decontaminated.

The final step in the encapsulation process is to load the canister into a transport cask. This is done with the same machine that lifted the canister out of the shielded frame. When the transport cask is filled it is fitted with a lid. The cask is then lifted, with an overhead crane, to a transport frame which the cask is lowered onto. With a transport vehicle the frame with the filled cask is transported to a storage for filled casks where it awaits shipment to the deep repository.

## 5. CANISTER LABORATORY

In order to test the crucial sealing operations, SKB has built a Canister Laboratory in the town of Oskarshamn. In the laboratory the electron beam welding and non-destructive testing operations will be tested and developed further. The equipment is installed in three different stations: a welding station, a station for X-ray testing and a station for ultra sonic testing and machining. Other important parts of the process which will be tested in the Canister Laboratory include transport and handling equipment for canisters and equipment in the handling cell. The first welding trials are planned to be performed during late 1998.

The Canister Laboratory will from now on be SKB's center for development of encapsulation technology as well as a center for educating personnel to the encapsulation plant. Results from the Canister Laboratory will be submitted to support the application for construction of the encapsulation plant.

## 6. CONCLUSIONS

The many years of research and development at SKB has resulted in the current design of the disposal canister and the encapsulation plant. Full-scale trials have shown that canisters can be manufactured and sealed with high quality. However, additional development work on designs and manufacturing methods are required. In this work, the Canister Laboratory will play an important role.

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# CLOSING SESSION

**Chairman**

**P.H. DYCK**
IAEA

**Panel**

**F. TAKÁTS**
Hungary

**W. LAKE**
United States of America

**J.I.A. VOGT**
Sweden

**F.C. STURZ**
United States of America

**M. PEEHS**
Germany

## SUMMARY OF THE SYMPOSIUM

### 1. INTRODUCTION

Continuous attention is being given by IAEA to the collection, analysis and exchange of information on spent fuel storage. Its role in this area is to provide a forum for exchanging information and to co-ordinate and encourage closer co-operation among Member States in certain research and development activities that are of common interest.

Symposia on this topic have been organised about once every four years since 1987. The purpose of the Symposium was to exchange information on the state-of-the-art and prospects of spent fuel storage, to discuss the world-wide situation and the major factors influencing the national policies in this field and to identify the most important directions that national efforts and international co-operation in this area should take.

The Symposium consisted of several oral sessions and one poster session. The oral sessions addressed four major topics:

- National programmes;
- Technology;
- Experience and licensing;
- R&D and special aspects.

The sessions were chaired by Messrs. V.B. Ivanov (Russian Federation), F. Takáts (Hungary), W. Lake (USA), J. Vogt (Sweden), L.F. Durret (France), F.C. Sturz (USA) and M. Peehs (Germany). The highlights of the Symposium are given below.

One hundred twenty five participants from 35 countries and 4 international organisations, who attended the presentations of papers and the poster session, reflected the world-wide interest in these important topics covered in this Symposium.

### 2. NATIONAL PROGRAMMES

It is noted that there continues to be world-wide growth in the generation of electric power using nuclear energy as its source. It is further noted that the rate of growth of nuclear energy generation has essentially levelled in Europe and North America while it has increased significantly in Asia. Although these trends have some impact on spent fuel management, including storage, the world-wide spent fuel production rate continues at about 10,800 t HM/yr.

About 130,000 tHM spent nuclear fuel was stored around the world at 1 January 1998. Over 70% (93,100 tHM) of this amount is stored in at-reactor pools in 32 countries, while the rest is in away-from-reactor (AFR) facilities, either wet or dry. Presentations from 20 countries in the session on National Programmes, and additional papers in the other sessions, covered 23 countries describing the technologies used to store more than 88% of the world total spent fuel to be stored.

There are three major categories for classifying spent fuel management policies and practices. These include a closed fuel cycle which involves reprocessing of spent nuclear fuel, a once-through fuel cycle which , of course, ends with disposal of the spent nuclear fuel, and a "wait and see" approach.

There are several countries that continue strong and extensive reprocessing programmes. These countries not only reprocess their own spent fuel, but also provide reprocessing services to other countries. New reprocessing programmes are being developed in Asia. The once-through cycle leads to disposal of spent fuel. In some cases, direct disposal is being pursued, while other countries are committed to a planned storage period preceding disposal.

One can view the decision to either reprocess or dispose as two ends of a spectrum. There is a wide range of options between these two ends which have been called the "wait and see" approach. The "wait and see" strategy should not be viewed as avoiding a decision. It is a choice that allows for developing technologies to mature, it can accommodate evolving national policies, and provide the time to address public acceptance issues. Wait and see can be used where reversibility is desired.

There are several aspects of the "wait and see" approach that are not positive. Although wait and see could avoid financial risks associated with pursuit of technologies or approaches that do not meet their initial expectations, it could lead to missed opportunities, accelerated programme activities needed to "catch-up", and other cost increasing measures. Furthermore, one who chooses to wait and see could be perceived as indecisive, avoiding a difficult decision, or passing an issue on to future generations.

Wet storage remains dominant, even as the use of dry storage concepts increases. Wet storage is essential for cooling newly-discharged fuel, and will continue to be the method of short term storage used in connection with reprocessing. The industry has an extensive experience base in wet storage with an excellent performance record.

Dry storage is being used increasingly as more long-term storage of spent nuclear fuel is done. Dry storage may prove to be a cost-effective activity. In addition, it can easily accommodate multipurpose systems (e.g., storage/transport, storage/transport/disposal).

Although at-reactor and on-site storage are common, many are considering the use of away-from-reactor storage concepts. There will always be storage at-reactors to allow for cooling. There could be increased at-reactor or on-site storage while interim and final spent fuel management solutions are being set. The centralized, away-from-reactor, storage options are expected to be more cost effective than the more dispersed, on-site storage approaches, which leads to the existence of many small storage facilities.

The national choices for spent fuel management and storage reported at the Symposium were numerous. Decisions were evidently based on thoughtful and complete considerations of national needs and conditions. Because the national needs and conditions tend to be unique there is not a universally "BEST" approach.

It should be understood, that the choice of how spent fuel is managed should not and need not affect safety, but may affect the effort expended to achieve the required level of safety. National regulations are applied to spent fuel management activities in a uniform way. The IAEA has an important role in assuring uniformity in National Regulations through development of its Safety Series Publications.

There were a number of common issues and needs that were raised during presentations and discussions. Public involvement and acceptance of spent fuel management activities was seen as an issue of increasing importance. Resource limitations are a common constraint on countries with smaller nuclear programmes. Because of the increases in communication and information exchange in the world today, events and actions by any one nation tend to affect the others. We need to all work together.

Two recommendations come to mind when the above issues and needs are considered. First, we must continue to exchange information, data and experience (from licensing to operations) on technical and public acceptance matters. Two, those who can, should consider providing financial and technical assistance to those with smaller nuclear programmes who are in need of such assistance. Any such support would be a wise investment in the future.

## 3. TECHNOLOGY

The presentations on dry storage largely focused on the specific needs of different utilities and organizations whilst ensuring compliance with the stringent safety requirements applicable in the different countries. It was generally recognised that casks are needed to provide for both storage and transportation requirements. This flexibility is of great importance to meet requirements with regard to design work, licensing procedures and manufacturing work. Furthermore, cask designs have to accommodate different fuel types, including MOX fuel, higher burn-ups and specific needs of individual power plants. Another example is early consolidation and encapsulation of spent fuel in disposal canisters in Germany. Current cask designs are based on proven and cost effective technology.

In the case of the vault storage systems, a concept, caters not only for the storage of spent fuel but also for the storage of a variety of other types of radioactive wastes. Also in this case the design is based on proven flexible and safe technology. In general, it appeared that there is a great consciousness of the need for optimisation and flexibility of storage designs in meeting requirements, within the constraints of the regulatory systems applicable in the different countries.

It was noted that, where reprocessing is not practised, dry storage designs have been developed. However, in the USA the concern of transportability is recently more important, as utilities are now expecting shipment of their fuels from reactor pools to a centralized storage facility. It was also mentioned that the transportability of the fuel is viewed favourably in terms of public acceptance. The dual purpose system already licensed or being licensed in the USA show that such solutions offer greater flexibility than storage only systems.

The situation is different in Europe where reprocessing and wet storage has been implemented by a majority of utilities. Therefore, transfer casks are largely available and AFR interim storage systems have been designed in using the transport containers as storage modules.

Utilities have reracked their reactor pools and increased fuel burnup in order to reduce the volumes of spent fuel to be managed. A concept of rod consolidation, which has been tested in the 1980's could be of interest to manage larger quantities of spent fuel (in the same volume) on reactor sites. Also of interest was discussion of plans in the UK to store AGR fuel, in pools for up to 80 years.

## 4. EXPERIENCE AND LICENSING

In three sub-sessions, eight papers discussed regulatory and operational experiences with interim spent fuel storage. They described regulatory process and oversight, burnup credit analysis and measurements, and operational performance.

The first sub-session focused on regulatory aspects. One paper described the development of a regulatory process used to license modifications to existing facilities to increase capacities. Two other papers described regulatory oversight issues related to quality assurance (QA) and quality control (QC) associated with dry cask storage design, fabrication and operation. These papers discussed a variety of problems encountered, the corrective actions, and the regulatory actions taken to assure safety. These papers stressed the need for continued vigilance by cask designers, fabricators and users to assure reliable and safe interim storage.

In the second sub-session three papers discussed analysis and measurement techniques related to the use of burnup credit for criticality analysis. Several measurement techniques and methodologies for the characterisation of spent fuel assemblies have been developed. These techniques include passive and active neutronic methods as well as gamma-spectrometric methods, in order to:

- determine the fissile inventory for safeguards;
- verify the operator's declaration consistency;
- use burnup credit in storage, transport and disposal operations.

A method for applying burnup credit to the criticality safety design analysis for PWR pool storage reracking was presented. Of particular importance was the prediction and validation of the isotopic composition or depletion analysis through experimental results. At present, fresh fuel reactivity is used for spent fuel in criticality analyses. Burnup credit would offer an increased packing density in storage racks and therefore an increased capacity.

The two papers in the third sub-session discussed operating experiences with wet and dry storage. Both papers described evidence of better than expected performance. Dry cask storage testing performed in the United States was presented. Participants expressed continued interest about the importance of cladding performance, and in particular, future plans for testing to provide further evidence of fuel and cask integrity after many years in dry storage.

## 5. R&D AND SPECIAL ASPECTS

The information presented during the Session on "R&D and Special Aspects" can be grouped into 3 parts:

a) Spent Nuclear Fuel (SNF) behaviour and properties;
b) SNF treatment technologies;
c) International co-operation aspects.

As a general observation it can be stated:

- no R&D was reported on wet storage. This underlines its position as a well established technology and all open questions related to wet storage have been satisfactorily answered;
- all contributions to the SNF dry storage discussed the storage performance of SNF for extended storage requirements such as increased fuel burnup, storage periods exceeding today's licensing limits or improved cladding material. The contributions demonstrate the continuing growth and acceptance of dry storage technology that has occurred over the past twenty years.

### 5.1. SNF behaviour and properties

Additional data on the creep of unirradiated Zr1%Nb to broaden the data base had been reported. This material shows acceptable creep behaviour. The assessment of SNF dry storage performance with increased burnup concludes only circumferential creep in the temperature range > 300°C needs to be analysed. There are no other significant defect mechanisms in operation as SNF burnup increases above 50 GW·d/tHM. Results presented show that spent nuclear fuel dry storage is feasible for all spent fuel on the market - even with increased burnup. The assessment of dry storage of SNF for periods exceeding the presently licensed storage times resulted in an optimistic forecast. It is expected that dry stored SNF that has remained intact for 20 years will continue to perform for up to 100 years as storage conditions become less onerous. Data to support the assessment of SNF with increased burnup was reported at the symposium.

### 5.2. SNF treatment technologies

To date there has been little reported data on the decay heat of SNF after longer periods of storage. In the Swedish CLAB facility the decay heat of long stored SNF has been determined by calorimetry and gamma ray spectrometry and compared to calculated data. It was concluded that for SNF of longer storage periods the decay heat can be calculated if a detailed EOL data base describing the in-service operation is available. Furthermore, it could be shown that the decay heat can be satisfactorily determined by $\gamma$ intensity measurements.

In two countries the final conditioning of SNF for disposal in geological formations is available. The Swedish experts had finalized their concept by demonstrating its technical feasibility. The technology will be developed in a pilot laboratory. In Germany the conditioning process is fully developed and the first installation on a technical-scale is nearly completely erected and will be commissioned in 1999.

### 5.3. International co-operation aspects

Countries with small nuclear programmes are candidates to join a regional spent fuel storage approach. This approach has already been applied for research reactor fuel with good success. Since time is ripe, also for commercial fuel, the IAEA started a tiative for interested countries.

Not just interim storage but a complete service including conditioning and also final disposal is suggested for an international approach. However detailed assessments are required in the areas of technology, economy, financing, institutional aspects, political aspects and ethic considerations. Until the outlined issues have been resolved the potential for such an approach can not be judged positively.

## 6. CONCLUSIONS

There are three major categories for classifying spent fuel management policies and practices. These include a closed-fuel cycle which involves reprocessing of spent nuclear fuel, a once-through fuel cycle which ends with the disposal of the spent nuclear fuel, and a "wait and see" approach. One can view the decision to either reprocess or dispose as two ends of a spectrum of options. It should be noted, however, that countries, which choose originally the reprocessing option, envisage the final disposal of high burnup and MOX spent fuel. The "wait and see" strategy should not be viewed as avoiding a decision, but as a means of evaluating the possible options and maintain the retrievability of the spent fuel.

Messages retrieved from the Symposium are that the primary option for spent fuel will be interim storage for the next decades, the duration of interim storage becomes longer than earlier anticipated and the storage facilities will have to be capable for receiving also spent fuel from advanced fuel cycle practices (i.e. high burnup and MOX spent fuel).

It was noted that the handling and storage of spent fuel is a mature technology and meets the stringent safety requirements applicable in the different countries. However, it is performed in a flexible and dynamic way, continuously adapting to changes in nuclear policy and progress in technology, for example transportability of spent fuel, application of burnup credit and utilisation of advanced fuel types.

Wet storage remains dominant, even as the use of dry storage concepts increases. Wet storage is essential for cooling newly-discharged fuel, and will continue to be the method of storage used in connection with reprocessing. The industry has an extensive experience base in wet storage with an excellent performance record. Dry storage is being used increasingly, as more long-term storage of spent nuclear fuel is done. Dry storage may prove to be a cost-effective activity that can easily accommodate multipurpose systems (e.g., storage/transport, storage/transport/disposal).

Possible Agency initiatives could be described as:

- To assist in providing a technical reference for country reports to be delivered for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (i.e. establish common understanding of various technical issues of spent fuel management, in general, and spent fuel storage, in particular);
- To assist Central and East European Countries with problems related to the storage of spent fuel and establishing adequate spent fuel storage facilities;
- To assist in the evaluation and research of the long-term behaviour of fuel and storage components in order to realise the anticipated long storage periods; data
- To continue the exchange of information, data and experience (from licensing to operations) on spent fuel storage technologies and public acceptance matters; and,
- To organise peer reviews in the subject of spent fuel storage and management.

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# THERMAL DESIGN AND TESTING OF A DRY SPENT FUEL STORAGE FACILITY

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## 1. INTRODUCTION

Till 1989, the Embalse Nuclear Power Station - a CANDU plant - has been storing the irradiated fuel in a wet storage pool. But even when there are no technological problems with the wet storage, it requires maintenance and produces secondary wastes. For this reason the CNE decided the design, testing and construction of a dry spent fuel storage facility.

## 2. THE GENERAL DESCRIPTION

### 2.1. The facility description

This facility consist of concrete vertical modules grouped in a storage site placed near the existing spent fuel storage bay. Every module (Fig. 1) has a capacity for 5 steel canisters, each one containing fuel baskets with 60 fuel bundles. The canisters are closed and filled with dry air. The dimensions of the fuel bundles, formed by 37 fuel rods, are 10 cm. diameter and 50 cm. length. The concrete module wall (lateral shielding) thickness is 80 cm and the internal diameter is 110 cm. Ordinary concrete was used.

### 2.2. Thermal design

The thermal design target was to avoid the release of fission products and protect the shielding capacity of the concrete wall. For the first condition, in order to avoid the cladding corrosion, a maximum cladding temperature of 180°C was settled. For the second one, in order to avoid the concrete cracking, a absolute maximum temperature of 110°C and a maximum temperature drop through the wall thickness of 60°C were settled.

The temperature levels, for the same other conditions, are directly related to the decay heat generated by the spent fuel elements. And the decay heat generated is a function of the fuel burnup and the decay time in the spent fuel storage bay. For this study it was assumed with a conservative criterion an outlet fuel burnup of 7,500 MW·d/t and an average irradiation power of 475 kW/bundle. The decay heat was calculated with the code "ORIGEN".

So, the two above mentioned targets implies as a final target to define the minimum decay time in the spent fuel storage bay, before its transference to the dry spent fuel storage facility.

### 2.3. Test facility

To accomplish these objectives a test facility was designed and built. The fuel rods corresponding to 18 fuel bundles in the central canister were simulated with electrical heated rods the other fuel bundles were simulated with cylindrical heaters of fuel bundle diameter (see Fig. 2). The other 4 canisters were simulated also with cylindrical heaters. 24 Thermocouples were installed in the concrete in order to determine the radial temperature distribution and other 32 thermocouples were installed to measure the "fuel cladding" temperature distribution.

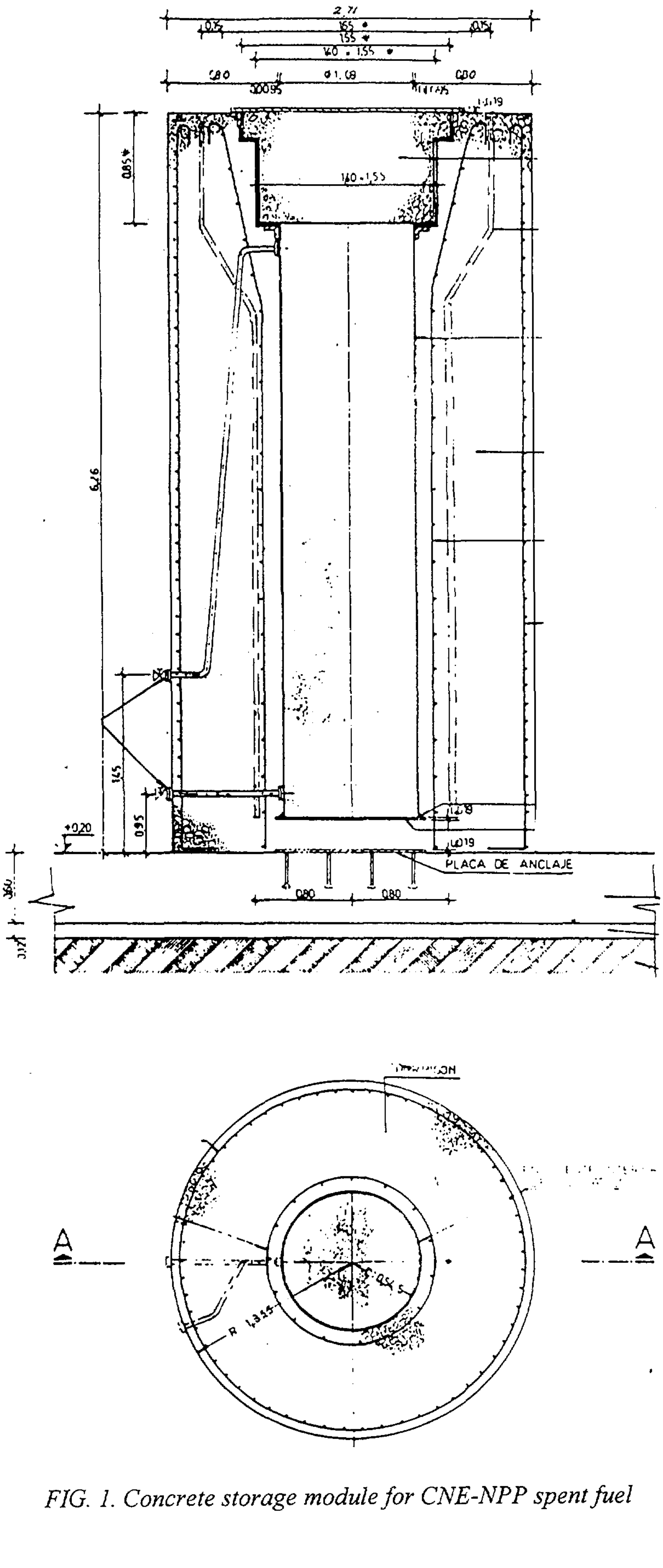


*FIG. 1. Concrete storage module for CNE-NPP spent fuel*

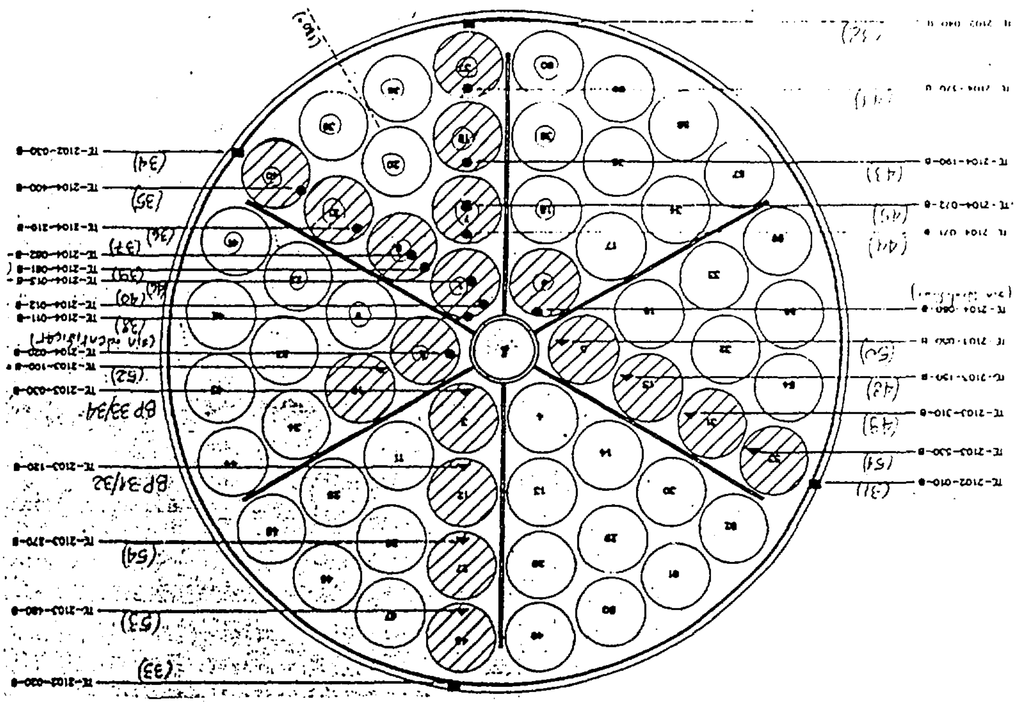


*FIG. 2. Test facility*

## 2.4. Test description

The prototype was placed in the plant site, near the place where the facility would be constructed. The test was performed, without interruption, over a period of one month and a half during summer time. The measurements were repeated each hour, 24 hours a day. The power was increased slowly. The measurements were done for three power levels with the results presented in Table I. The maximum power level corresponded to spent fuel elements with 4.5 years in the spent fuel storage bay.

TABLE I. RESULTS OF MEASUREMENTS

| POT (Watts) | Tmax (H°)[a] °C | ΔTmax (H°)[b] °C | K (H°)[c] W/m/°C | Tmax (F.R.)[d] °C |
|---|---|---|---|---|
| 2,228 | 59.5 | 30.1 | 1.9 | 97 |
| 3,171 | 78.7 | 40.1 | 2.0 | 123 |
| 4,753 | 101.6 | 61.4 | 2.0 | 160 |

[a] Maximum concrete temperature
[b] Concrete wall thickness temperature difference
[c] Concrete thermal conductivity
[d] Maximum fuel rod temperature

## 2.5. Field measurements

After these tests, two instrumented concrete modules were loaded. The first with 7 years, and the second with 6.5 years decayed fuel elements. With the concrete temperature drop measured it was possible to calculate the fuel power and to compare it with the "ORIGEN"values (Table II). From the Table it appears that the ORIGEN code over predict the power.

TABLE II. MEASURED AND CALCULATED FUEL POWER

| Decay Time (years) | "Measured" Power (Watts) | "ORIGEN" (Watts) |
|---|---|---|
| 7 | 3,050 | 3,180 |
| 6.5 | 3,250 | 3,420 |

## 3. CONCLUSIONS

The concrete thermal conductivity obtained for the module, through the power and temperature field measurements, were in good coincidence with the one obtained for a concrete sample prepared at the laboratory. From test and field measurements, we can assume as a conservative conclusion that the minimum acceptable decay time for the fuel elements in the spent fuel bay is 5 years. These studies were satisfactorily complemented with $\gamma$ ray abortion measurements to verify the shielding concrete capacity before and after the heating experience and also with concrete deformation measurements. At the moment our Regulatory Authority accepted a minimum decay time of 6 years for the spent fuel. There are 80 concrete modules, 54 of which are loaded.

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# WWER SPENT FUEL CRITICALITY, DEPLETION AND SHIELDING STUDIES



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The purpose of this paper is to present the results of WWER spent fuel inventory studies, in applying well-known depletion codes, as well as a criticality evaluation of the WWER spent fuel casks by Monte Carlo codes. Some results of WWER spent fuel shielding calculations are given too.

With the aim to evaluate the fuel nuclear inventory after spent fuel discharge, the point depletion codes ORIGEN-S (as a part of the SCALE-4.3 modular code system [1] ) and NUKO [2] have been implemented and applied. Using the ORIGEN-S code, the fuel (U and Pu) and waste (minor actinides and fission products) characteristics, such as isotope concentrations, decay heat and radiation sources for given initial composition and irradiation history, can be estimate.

The spectral and burnup calculations were performed using the NESSEL-4 code [3]. The fission products in lower concentrations and the minor actinides were calculated by the NUKO code. This code processes spectrum dependent one-group cross sections and solves the system of differential equations for the nuclide chain to be considered.

Studies on nuclear inventory changes for WWER-440 fuel assemblies, which reached an average batch burnup of about 30 MW·d/kgU during a three-year core lifetime, have been carried out by using the ORIGEN-S and the NUKO codes. The fuel assemblies with an initial enrichment of 3.6% have been irradiated in Unit 1of the NPP Kozloduy (a WWER-440) during fuel cycles 16, 17 and 18. The cycle lengths have been 274, 339 and 203 full power days (FPD), respectively.

Some important preliminary results for WWER-440 irradiated fuel assemblies, such as fuel nuclide concentrations ( $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{242}$Pu), minor actinide concentrations ( $^{237}$Np, $^{241}$Am, $^{243}$Am, etc.), fission product concentrations ($^{99}$Tc, $^{133}$Cs etc.) for the end of fuel cycles16, 17 and 18, as well as after 1, 2 and 5 years cooling time, will be presented.

The results obtained, although preliminary, give qualitative presentation of some important WWER spent fuel and waste post-irradiation characteristics. Using these results it could be concluded that the both codes ORIGEN-S (with appropriate WWER cross section sets) and NUKO could form a basis for spent fuel storage safety studies, including the burnup credit analysis in WWER spent fuel storage facilities.

The subcriticality of fuel storage facilities has to be guaranteed in every step of the nuclear fuel cycle. This includes both the fresh and depleted fuel in transport and storage facilities as well as the final disposal of spent fuel. The criticality safety analysis is needed also in the case of introduction of new fuel types or storage facilities as well as by modifications of old ones. To perform criticality safety studies computer code systems applying the Monte Carlo method are usually used.

A criticality safety evaluation of a WWER-440 and WWER-1000 spent fuel cask test model was performed, applying two world wide known Monte Carlo computer codes: SCALE (Version 4.3) and MCU [4] . The first model is based on the SKODA WWER440/84 transport and storage cask [5] with 84 WWER-440 fuel assemblies (3.6% enrichment). The other one is based on the Russian design [6]. On the basis of the results obtained, it can be concluded that for the WWER fuel cask test models the basic criticality safety criterion, namely the effective multiplication factor $k_{eff}$ less than 0.95, is satisfied quite well.

The MCNP code [7] has been used for dose calculation in WWER-1000. A comparison of the results based on the 175 multigroup neutron constants library and the point energy presentation library of ENDF/B-6 has been performed.

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XA9951816

# STUDY ON INCREASING SPENT FUEL STORAGE CAPACITY AT JURAGUÁ NPP

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## 1. INTRODUCTION

The current Cuban nuclear programme includes two water moderated and cooled WWER-440 type reactors. These reactors, now under construction, are 417 MWe each. After being discharged from the core, the spent fuel assemblies are placed in the racks at the storage pool. By design, this pool can store up to 2,094 fuel assemblies (or 252 t HM) without additional neutron absorbers and with a reserve margin of 25%.

The delay in the decision about the final disposal of the spent fuel, will lead to longer interim storage. The re-racking of the storage pools has demonstrated to be an economical and feasible option to increase the storage capacity within the plant. In essence, the method is to introduce a new rack with a smaller separation between the fuel assemblies which are placed in hexagonal boron steel shell to assure the required sub-criticality level ($k_{eff} \leq 0.95$). With this method it has been possible to approximately double the capacity of the WWER storage pool [1, 2].

The re-racking of the storage facility led to the analysis of the new conditions for criticality, shielding, residual heat removal and mechanical loads over the structures. In the following, a summary is given of the studies carried out on the criticality and dose rate changes in the vicinity of the storage pool of Juraguá NPP.

## 2. DESCRIPTION

The WWER fuel assemblies are composed by 126 fuel pins with an hexagonal Zr+2.5% Nb shell. The wrench diameter is 14.40 cm and the wall thickness is 0.21 mm. The fuel pins consist of 0.76 cm diameter $UO_2$ pellets in a 0.065 cm thickness Zr+1% Nb cladding. The fuel pin lattice pitch is 1.22 cm [3, 4]. The spent fuel assemblies discharged from the reactor core are transferred to the racks in the storage pool conforming a hexagonal lattice. A general overview of the storage pool is shown in Fig. 1. Note the positions of points $P_1$, $P_2$ and $P_3$ for which the radiation dose is computed.

## 3. CRITICALITY ANALYSIS

Based upon preliminary evaluations of [1 and 2], our calculations were performed under the following suppositions:

- Maximum WWER fuel enrichment: 3.6 w/o;
- Storage pool filled with clean water at 20 °C;
- Boron steel shell of 3.0 mm thickness;
- Boron content of the shell: 1 w/o;
- Storage mesh pitch: 16, 17, 18 and 19 cm.

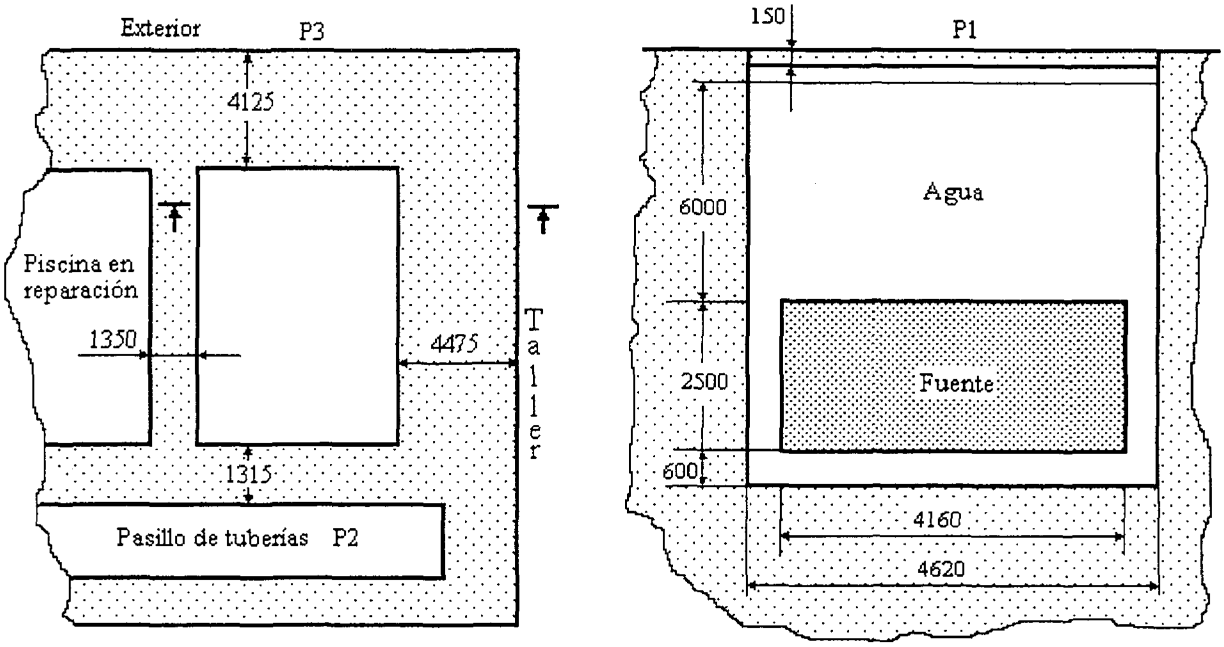


*FIG. 1. Layout of the spent fuel storage pool. Plant view (left), section (right)*

The WIMS-D/4 spectral code [5] was used for calculation of the neutron multiplication factors $k_{inf}$ and $k_{eff}$. The storage pool rack dimensions were considered through geometrical bucklings. The values of reflector saving $_x = 8$ cm, $_y= 8$ cm and $_z = 7$ cm were taken from measurements at the ZR-6 critical facility [4].

Calculations were carried out by modelling the super-cell of the fuel assembly within the shell by the option CLUSTER DSN, based on the discrete ordinates and cylindrical geometry [6]. To describe the spectral effects in the super-cell, 36 energy groups were taken. The original 22.5 cm lattice pitch without boron steel shell was taken as reference. Tables I and II show the results of the evaluations.

TABLE I. MULTIPLICATION FACTORS VS. LATTICE PITCH

| Pitch, [cm] | F | $k_{inf}$ | $k_{eff}$ |
|---|---|---|---|
| 16.0 | 1.98 | 0.94401 | 0.93675 |
| 17.0 | 1.75 | 0.82934 | 0.82325 |
| 18.0 | 1.56 | 0.74226 | 0.73696 |
| 19.0 | 1.40 | 0.67753 | 0.67275 |

B-steel shell: 15.0 cm Ø int. wrench and 3.0 mm thickness.

TABLE II. MULTIPLICATION FACTORS VS. BORON STEEL SHELL THICKNESS

| Thickness, [mm] | $k_{inf}$ | $k_{eff}$ |
|---|---|---|
| 3.0 | 0.94401 | 0.93675 |
| 2.5 | 0.95070 | 0.94345 |
| 2.0 | 0.96222 | 0.95494 |
| 1.5 | 0.98189 | 0.97449 |

B-steel shell: 15.0 cm Ø int. wrench. Lattice pitch: 16.0 cm.

F denotes the densification ratio defined as:

$$F = V_0/V \tag{1}$$

where, V is the volume associated to each irradiated fuel assembly (storage lattice cell). Index 0 denotes the normal pitch lattice.

## 4. SHIELDING ANALYSIS

The goal of the shielding calculation of the storage pool is to evaluate the doses due to the gamma radiation. The neutron source density resulting from ($\alpha$, n) reaction as well as spontaneous fission are negligible. To solve the neutral particles transport equations, it is necessary to know the source intensity (Q) and the nuclear densities (N). These magnitudes were calculated for the normal lattice ($Q_0$, $N_0$) using the ORIGEN 2 depletion code [7], which simulates the fuel irradiation in the reactor core and its subsequent decay in the storage pool.

$Q_0$ and $N_0$ were determined for 3.6 w/o enriched fuel, considering 3 years of continuous irradiation in the reactor core, a burnup of 28,600 MW·d/t HM and 2 years of cooling in the at reactor pool.

Under the assumption that the radiation source is mainly due to the irradiated fuel for a compacted lattice, Q and N can be written as:

$$Q = F\,Q_0 \qquad (2)$$

$$N = F\,N_0 \qquad (3)$$

In these expressions, Q and N are homogenized in the cell volume. Note that Equation (3) only applies to the nuclear densities of the materials of the irradiated fuel. The $\gamma$ source distribution in 18 group representation is shown in Table III.

To solve the transport equation, the ANISN code [8] was used and combined with the CASK microscopic cross section library [9]. ANISN uses the discrete ordinates method in multi-group approximation to solve the one-dimensional neutral particles transport equation. The CASK library is widely used for shielding calculations, it considers 22 neutron and 18 $\gamma$ groups. The scattering differential cross sections are expressed by the third order Legendre expansion, $P_3$.

Three interesting points were evaluated in the AFR spent fuel storage building. The $S_{12}$ $P_3$ approximation of ANISN, with an appropriated spatial mesh in one dimensional plane geometry, was used. Figure 1 depicts the three points considered: $P_1$: Floor of the irradiated fuel elements storage. $P_{1a}$: for variable water level (from 1 to 5 m) without upper lid. $P_{1b}$: for 6 m water level with upper lid. $P_2$ : Pipeline gallery. $P_3$: Contiguous pool (under repairing). Tables IV and V present the relative dose values (d) referred to the normal lattice.

TABLE III. 18-GROUP GAMMA SOURCE DENSITY OF THE NORMAL LATTICE

| Energy Group | $E_{min}$ (MeV) | $E_{max}$ (MeV) | $Q_\gamma$ ($cm^{-3}s^{-1}$) | Energy Group | $E_{min}$ (MeV) | $E_{max}$ (MeV) | $Q_\gamma$ ($cm^{-3}s^{-1}$) |
|---|---|---|---|---|---|---|---|
| 1 | 8.00 | 10.00 | 6.10E-02 | 10 | 1.00 | 1.33 | 3.03E+08 |
| 2 | 6.50 | 8.00 | 3.98E-01 | 11 | 0.80 | 1.00 | 1.62E+09 |
| 3 | 5.00 | 6.50 | 2.43E+00 | 12 | 0.60 | 0.80 | 4.15E+09 |
| 4 | 4.00 | 5.00 | 2.30E+00 | 13 | 0.40 | 0.60 | 5.31E+09 |
| 5 | 3.00 | 4.00 | 1.22E+05 | 14 | 0.30 | 0.40 | 5.73E+08 |
| 6 | 2.50 | 3.00 | 9.92E+05 | 15 | 0.20 | 0.30 | 1.03E+09 |
| 7 | 2.00 | 2.50 | 6.57E+07 | 16 | 0.10 | 0.20 | 2.58E+09 |
| 8 | 1.66 | 2.00 | 2.34E+07 | 17 | 0.05 | 0.10 | 3.82E+09 |
| 9 | 1.33 | 1.66 | 1.67E+08 | 18 | 0.00 | 0.05 | 1.85E+10 |

TABLE IV. RELATIVE DOSE VALUES (d) AT POINT $P_{1a}$ VS. F

| Water Layer [m] | F=1.5 | F=1.8 | F=2.0 |
|---|---|---|---|
| 1.0 | 1.12 | 1.18 | 1.21 |
| 2.0 | 1.13 | 1.20 | 1.24 |
| 3.0 | 1.13 | 1.20 | 1.24 |
| 4.0 | 1.13 | 1.20 | 1.24 |
| 5.0 | 1.13 | 1.20 | 1.24 |

TABLE V. RELATIVE DOSE VALUES (d) AT POINTS $P_{1b}$, $P_2$ AND $P_3$ VS. F

| Point | F=1.5 | F=1.8 | F=2.0 |
|---|---|---|---|
| $P_1$ | 1.11 | 1.18 | 1.21 |
| $P_2$ | 1.13 | 1.20 | 1.24 |
| $P_3$ | 1.13 | 1.20 | 1.24 |

## 5. CONCLUSIONS

From the nuclear criticality calculation, one can conclude that a 3 mm thick 1% boron steel hexagonal shell allows to achieve a densification factor of 1.98 assuring the required sub-criticality level ($k_{eff} \leq 0.95$).

The densification of the fuel storage arrangement leads to a dose rate increment near the storage pool of 1.11 to 1.24 times, as the densification factor varies in the range from 1.5 to 2.0. From the results, the approximate expression $d = 0.75+0.25F$ for estimating the relative dose rate was obtained.

The results of the present study allow to conclude that from the point of view of the criticality and radiation shielding condition, the densification of the fuel storage is a possible option to increase the interim storage capacity at Juraguá NPP.

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IAEA-SM-352/5P

# SPENT FUEL TEMPORARY STORAGE ENVIRONMENTAL IMPACT ASSESSMENT

XA9951817

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## 1. INTRODUCTION

The aim of the photon field analysis was to assess the radiation risk inside the storage hall and the influence of the storage facility on the surrounding environment, as well as to evaluate and estimate the time dependence of these factors, corresponding to the gradual storage filling. The storage facility was build on the Dukovany nuclear power plant territory and opened for a trial operation at the end of 1995. The Castor 440 containers are used for the storage of the spent fuel assemblies. The maximum capacity of the store is sixty containers and about one third of the capacity is used till now. A set of the periodical measurements was performed in order to get basic information on the time dependence of the photon field spatial distribution and spectral characteristics at the temporary storage facility and its vicinity.

## 2. DESCRIPTION OF THE MEASUREMENT

For the direct measurements, the high sensitive dose/dose rate portable monitor with special combined scintillation detector, compensated for the energy and directional dependence, was used. The μ-Nomad (EG&G Ortec) portable scintillation spectrometer with ϕ3"x3" NaI(Tl) detector was used for the spectrometry measurements. The deconvolution technique based on the knowledge of the detection system response matrix and the Scofield-Gold iterative method was used for the spectrometry data mathematical processing. This technique enables to calculate the spectral distributions of the required dosimetric quantities. Knowledge of those spectral distributions makes it possible to identify (in limits of detector resolution) the main radionuclide, contributing to the analysed photon field and evaluate their contributions. The necessary detection system response matrixes were calculated using a Monte Carlo method. For correct interpretation of the results of low level (natural) background measurements, it is necessary to consider that spectra in the energy interval up to 3 MeV, do not cover the main part of cosmic ray component response and calculated values correspond practically only to the terrestrial component.

## 3. RESULTS AND DISCUSSION

A set of measurements in the vicinity of the storage facility was done for nine reference points, selected around the storage building. To follow the development of the photon field characteristics whilst gradual filling the storage hall, the set of identical measurements was repeated periodically

(8 times up to now). The first measurements (27/11/95) were carried out for the empty storage facility before starting operation. An overview of measured air kerma rates is shown in Fig. 1. The systematically lower values on 11/12/95 and 11/03/96 are due to a snow layer of about 15-20 cm. Most of the results lie safely in the 15% error band (guarantied measuring method accuracy). An exception is the slight increase (about 25-35 % in comparison with the mean values) of the measured values in the points ***e*** end ***g***, which are closest to the filled part of the storage facility.

Spectral distributions of the air kerma rates (calculated from the scintillation spectrometry data) for all measured points represent typical background terrestrial component. No specific contribution of the stored fuel to any part of spectra was identified up to now (including mentioned points ***e*** and ***g***).

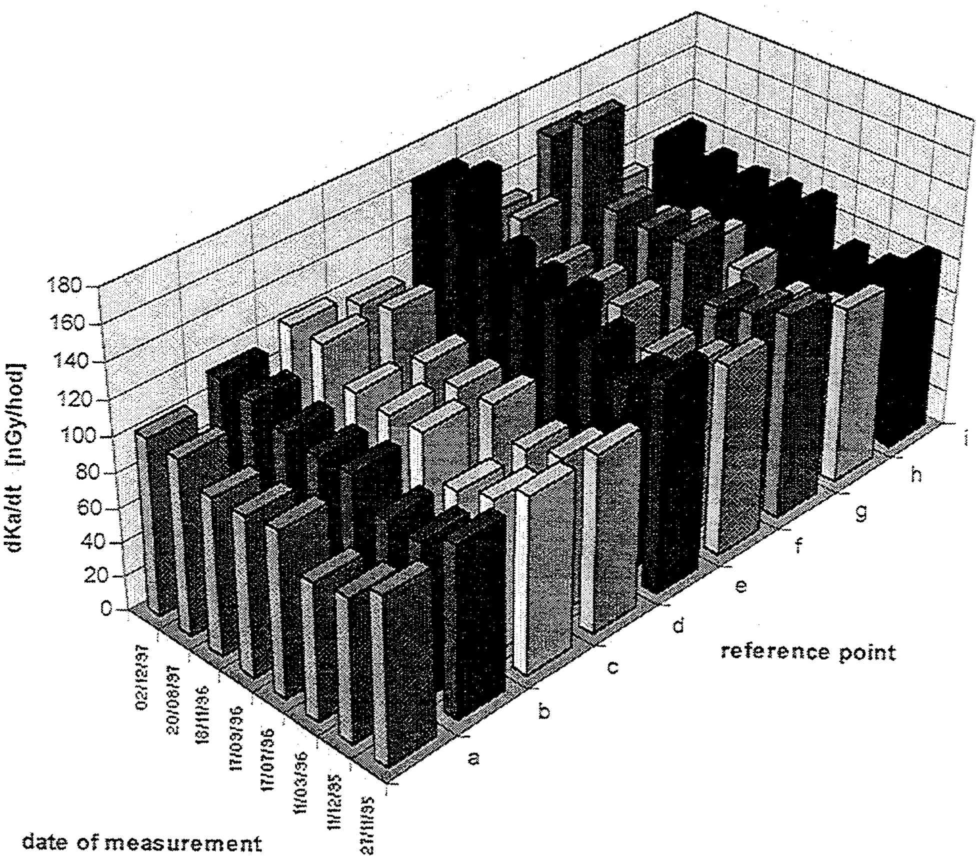


*FIG. 1. Air Kerma rates in the vicinity of the temporary spent fuel storage facility*

The measurement results inside the storage hall are presented in Fig. 2. To follow the development of the photon field characteristics with the gradual filling of the storage facility, the set of measurement was repeated periodically.

The spatial distribution of the air kerma rate in the storage hall was calculated from values measured in the individual points. An example of an isolines map (for the last measurements on 01/12/1997) is shown in Fig. 3. The shape of the isolines near the containers is only rough due to relative low number of measuring points (11 measuring points were used, see the marks in the Fig. 3), mutual screening among containers, etc.

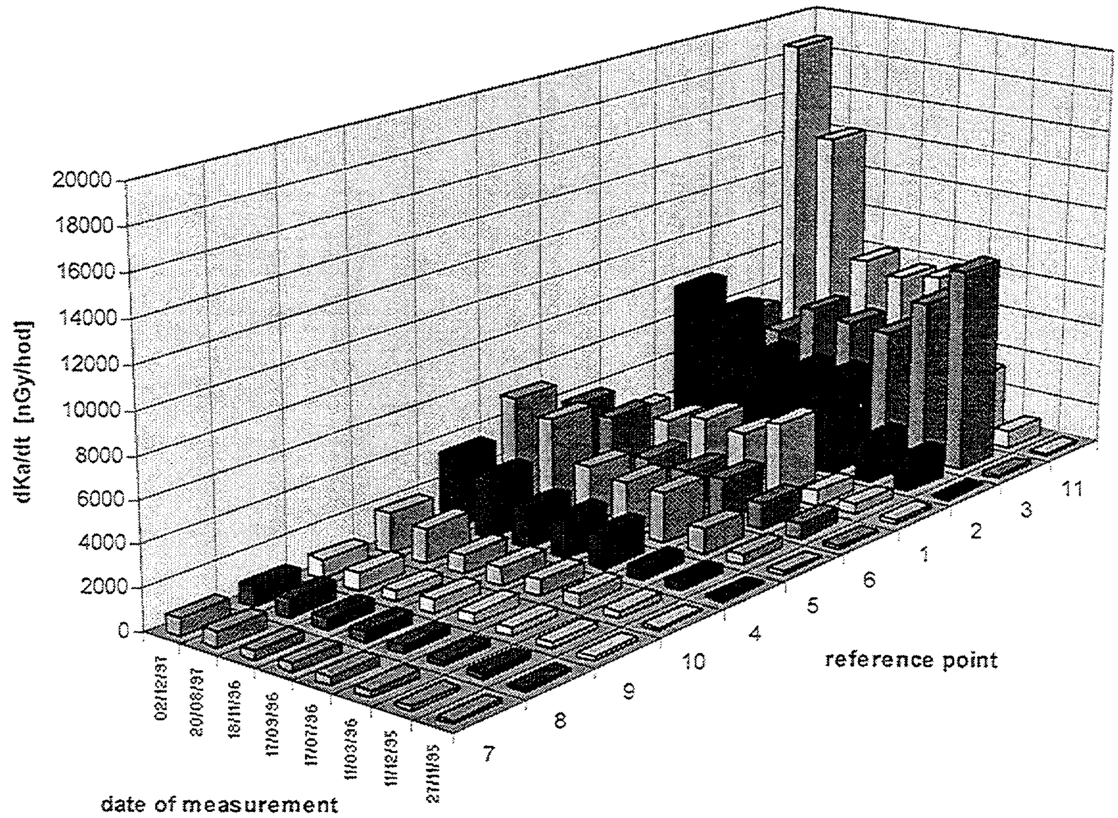


*FIG. 2. Air Kerma rates in the temporary spent fuel storage facility*

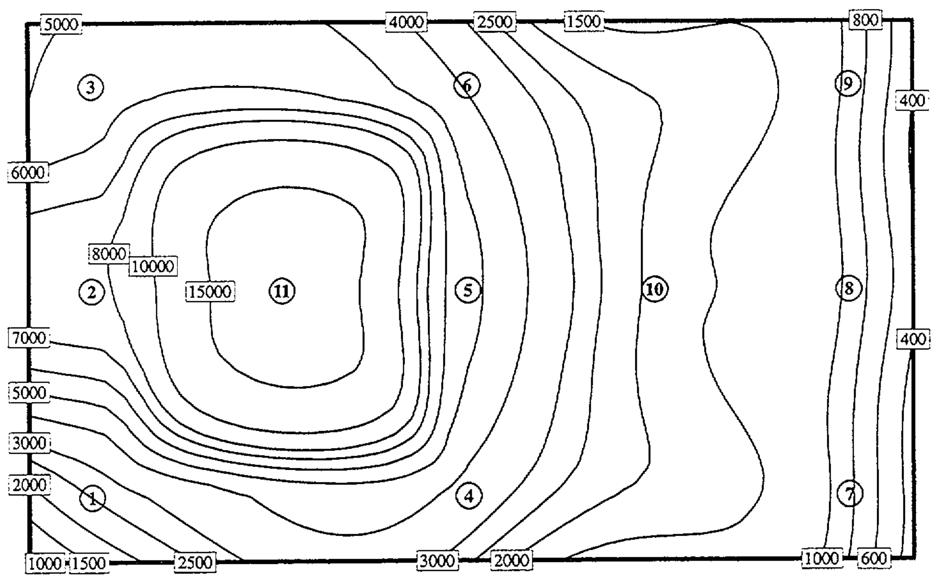


*FIG. 3. Isolines $dK_a/dt$ [nGy/h] in the storage hall on 01/12/1997*

The spectral analysis of the gamma field shows about 10% contribution of the high energy part (up to 3 MeV). In order to interpret the high energy response, an additional study of the neutron fields using the Bonner spectrometer was done. The relatively high neutron fluences were confirmed. It was concluded that the high energy response consist of two parts:

- response of prompt gamma radiation;
- response to neutrons due to sensitivity of NaI(Tl) detector to neutrons.

The spectrum measured in the reference point 5, represents a typical spectrum in the storage hall (see Fig. 4).

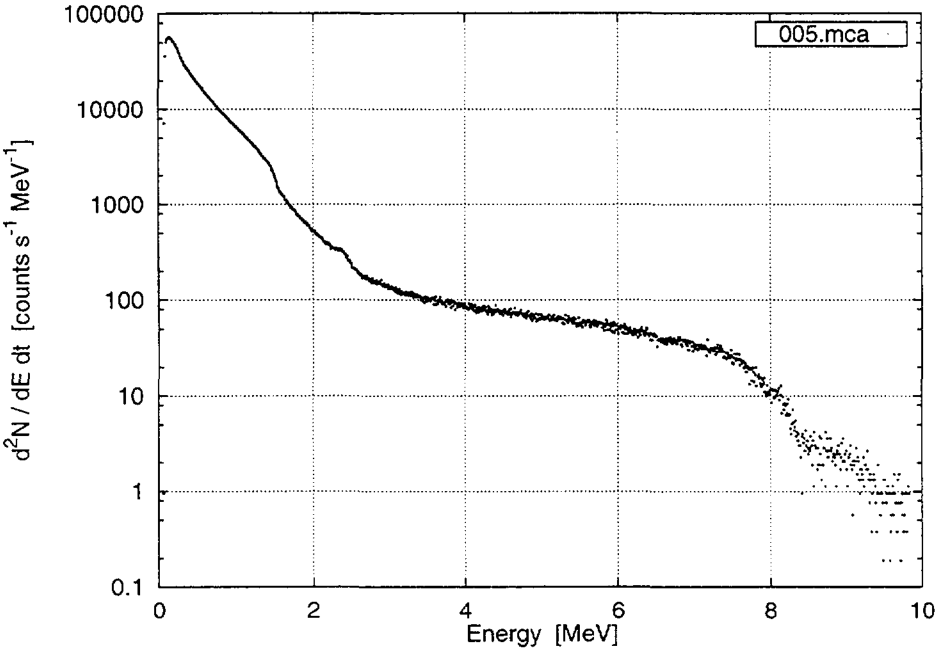


*FIG. 4. Typical spectrum in the storage hall (measuring point 5, 19/08/1997)*

## 4. SUMMARY AND CONCLUSION

The assessment indicates that the scintillation spectrometer and the deconvolution technique, based on the knowledge of the detection system response matrix and the Scofield-Gold iterative method, are suitable for the calculation of the spectral distributions of the required dosimetric quantities.

The performed photon field analysis in the vicinity of the temporary spent fuel storage facility and inside the storage hall respectively, gives a basic view and assessment of the radiation hazards and trends of development of this hazards. It was confirmed that there is no substantial influence of the storage facility on the photon fields and the surrounding environment up to now.

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IAEA-SM-352/6 P

# NUCLEAR SAFETY ASPECTS OF THE DRY SPENT RBMK FUEL CASK STORAGE SITE AT THE IGNALINA NUCLEAR POWER PLANT IN LITHUANIA

XA9951818

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For long-term interim storage of spent RMBK fuel of the Ignalina Nuclear Power Plant (INPP) in Lithuania, it was decided to use dry storage in casks. In this case, the total activity to be stored is split into individual units (casks). Each cask represents a closed and independent safety system, fulfilling all safety-relevant requirements for both normal operational and hypothetical accidental conditions.

Two cask types have been delivered and are under contract for the INPP storage site:

(1) The CASTOR RBMK cask made of ductile cast iron and closed with a screwed double barrier lid system.
(2) The CONSTOR RBMK sandwich cask made of an inner and outer steel shells and of reinforced heavy concrete. The lid system consists of a screwed lid and two welded lids.

The cask are designed in such a way that the neutron multiplication factor $k_{eff}$ is below the limiting value of 0.95 and the maximum dose rates fulfil both the criteria of IAEA and the Russian Standard OSP 72.

For the gamma and neutron shielding calculations, the two-dimensional transport program DORT in its version 2.8.14 [1] is used. DORT is directly based on the earlier ORNL DOT codes and it is well established and widely used in German licensing procedures.

The calculation model including the source consists of nearly 15,000 mesh points for the CONSTOR RBMK to match the real geometry as closely as possible. In the calculation model, the fuel half-assemblies are combined with the basket to form cylindrical sources.

The results for the side wall (see Fig. 1), bottom and lid side show that the dose rates are below the design limit of 2 mSv/h by a large margin. At 1 m and 2 m distance the calculated maximum dose rates are well below 0.1 mSv/h.

The subcriticality analyses were performed using the validated SCALE 4.3 program system [2]. For subcriticality analyses it contains different cross-section libraries from which the library 44GROUPNDF5 [3] has been selected for this problem.

For the subcriticality analyses, RBMK fuel-half-assemblies, so called bundles are modelled in detail.

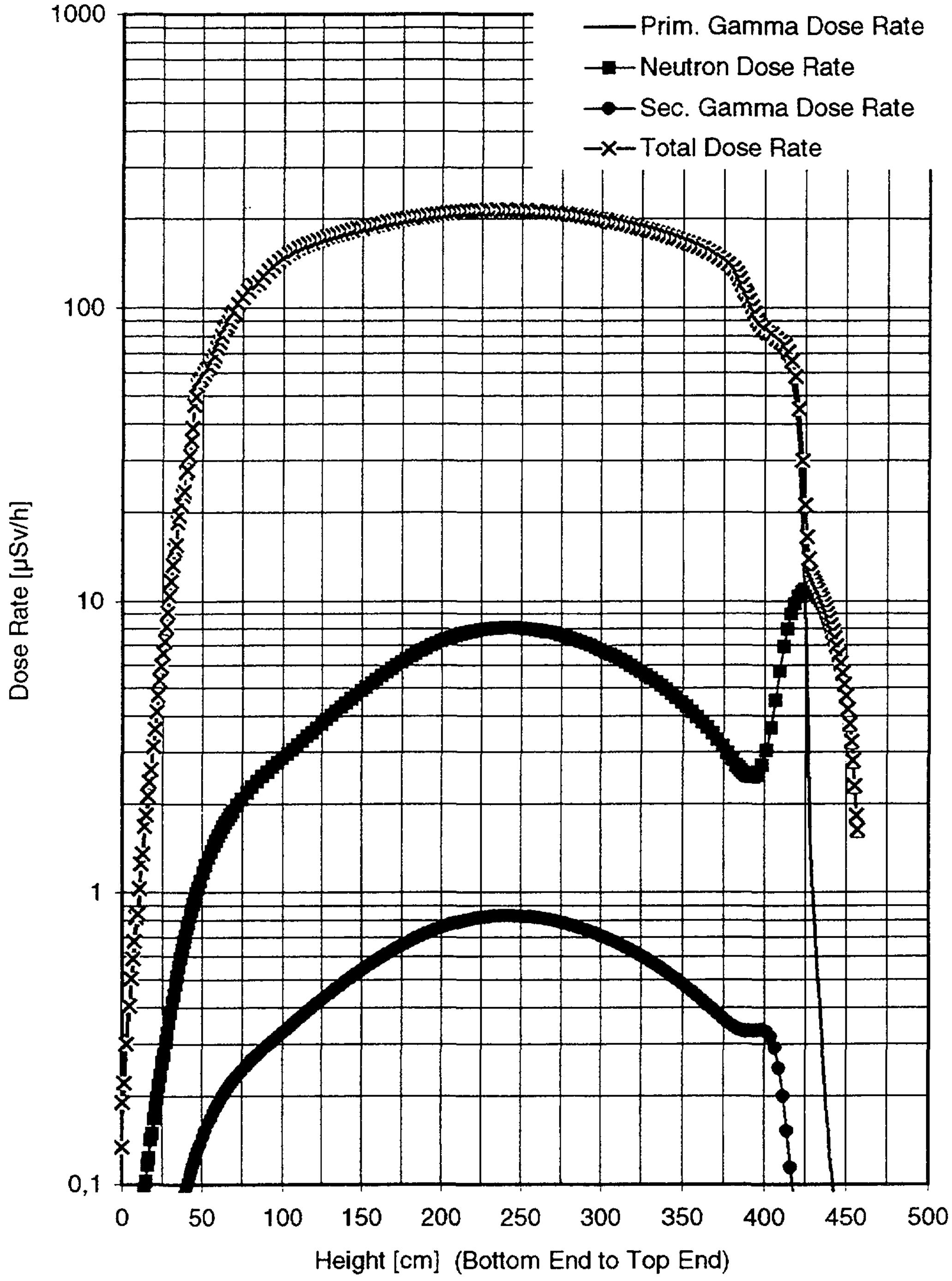


*FIG. 1. Shielding calculations for the CONSTOR RBMK cask*

The results of the criticality analyses for an unlimited number of hypothetically flooded casks are shown in Fig. 2. It can be stated that for all normal and for hypothetical accident conditions the values of $k_{eff}$, including 2 standard deviations, are well below the limiting value $\leq 0.95$.

The CASTOR RBMK cask was licensed for storage by the Lithuanian Competent Authority VATESI. The CONSTOR RBMK cask design was certified as type B(U)F package by GOSATOMNADZOR of Russia. The Lithuania licensing procedure for this type of storage cask will be finished in 1998.

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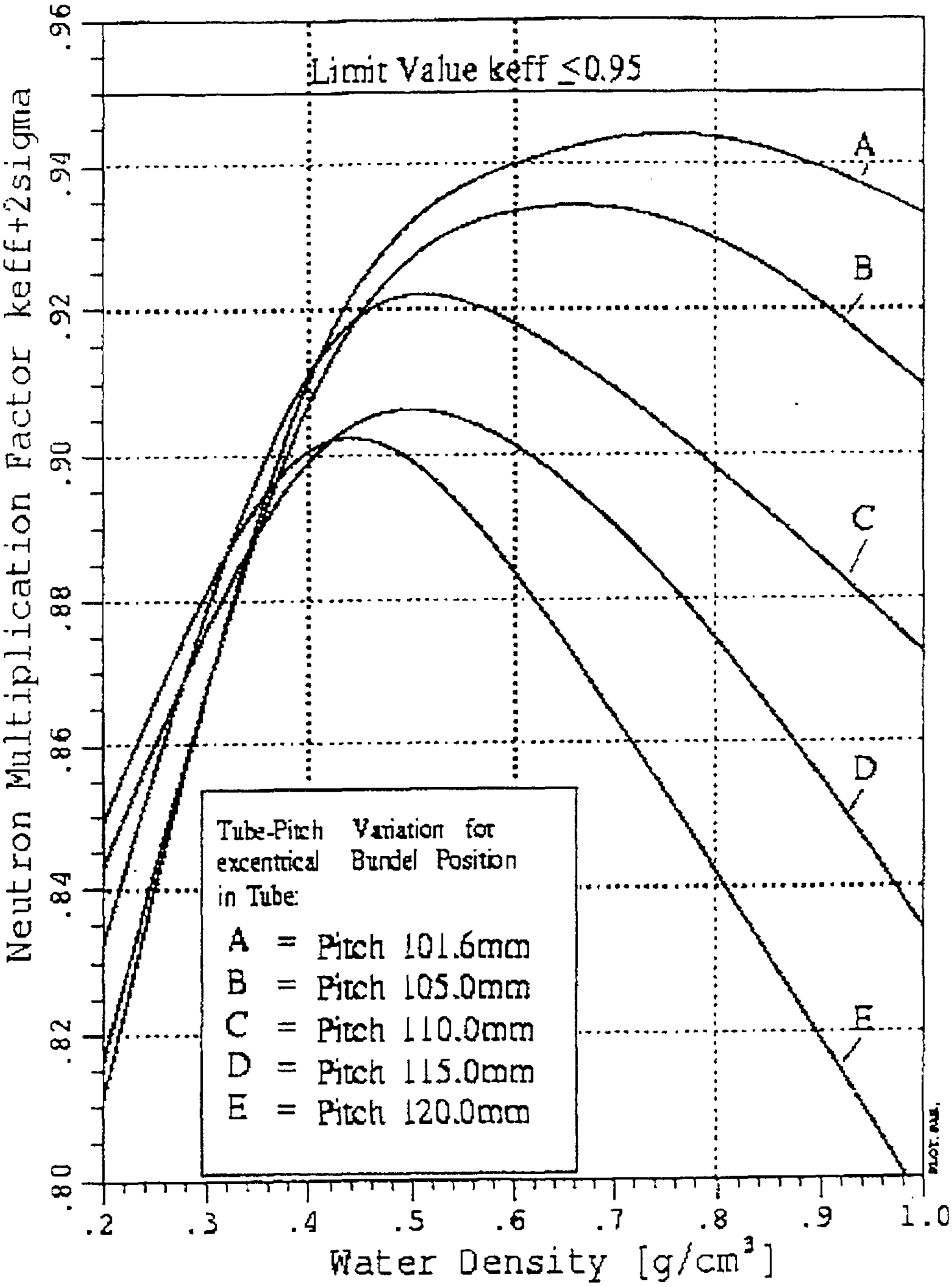


*FIG. 2. Results of criticality analyses*

IAEA-SM-352/7P

XA9951819

# HEAT REMOVAL TESTS ON DRY STORAGE FACILITIES FOR NUCLEAR SPENT FUELS[1]

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## 1. INTRODUCTION

In Japan, spent fuel generated in a nuclear power plant is controlled and stored in a storage facility until it is reprocessed. It is foreseen, that the amount of the spent fuel increases every year and a storage facility with large capacity constructed away-from-reactor is needed. If a large amount of spent fuel is stored in a dry storage facility away-from-reactor, the natural convection system of the storage facility is advantageous from the safety and economic point of view. To realize this type of storage facility, it is necessary to develop an evaluation method for natural convection characteristics and to make a rational design taking into account safety and economy.

## 2. HEAT REMOVAL TESTS OF STORAGE FACILITIES

To evaluate the heat removal characteristics of storage facilities such as cask, vault, and silo, tests were performed with 1/2 or 1/5 scale model of the facility [1]. The main results are:

(1) Cask storage system

Figure 1 shows test equipment for a cask type storage system. In the test, flow patterns of cooling air were observed (Fig. 2). It was observed that near the floor between the heater row, air coming from the inlet flowed to the center with relatively high velocity. Heat transfer of the heater surface dominates the vertical flow of the natural convection mainly but it is necessary to take account of the effect of the cross flow;

(2) Vault storage system (cross flow type)

It is desirable, that the cooling air flows across the heater to promote the heat transfer of the heater surface in this system. The test equipment is shown in Figure 3 and heat removal tests were performed. As a result, a distinction method of the flow pattern in the test module was discussed. Ri number is an indicator of the flow pattern. If $Ri < 3$, the cross flow dominates in the heater zone. The average heat transfer rate of the heater surface after row No.4 is almost the same and agrees with existing empirical equation (Fig. 4);

(3) Silo storage system

Figure 5 shows the test equipment. The relation between the heat transfer and flow rate can be arranged by Ri number. If the flow rate is small (Ri number is large), the heat transfer rate can be calculated by the equation of the natural convection on a horizontal cylinder. If the flow rate is large, the heat transfer rate can be calculated by the equation of the forced convection on a horizontal cylinder (Fig. 6).

## 3. CONCLUSION

Through the heat removal tests with the reduced scale models of the storage facilities (cask, vault, silo), the flow pattern in the test modules have been identified. The temperature and velocity distributions were obtained and the heat transfer characteristics are evaluated.

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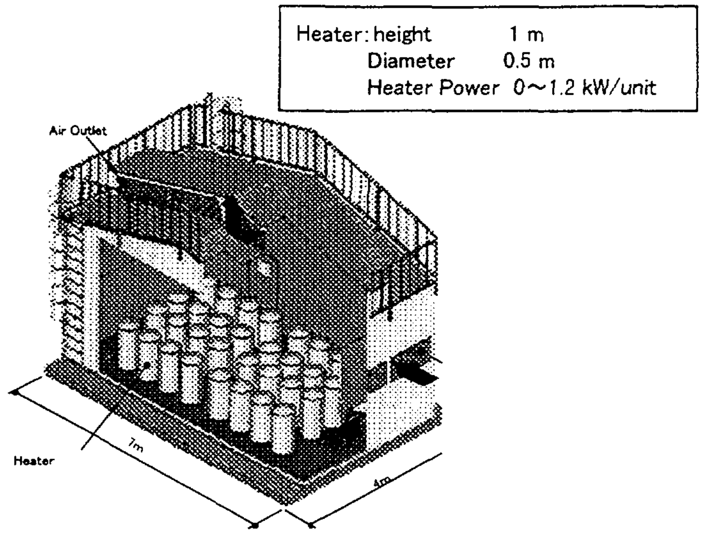


FIG. 1. Test equipment (Cask Storage System)

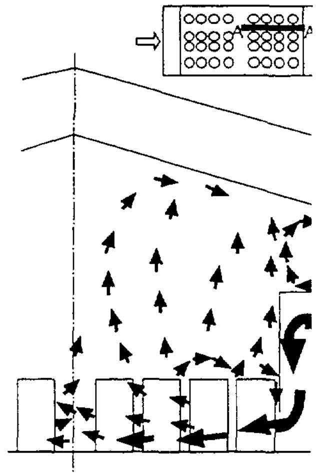

FIG. 2. Flow pattern (Cask Storage System)

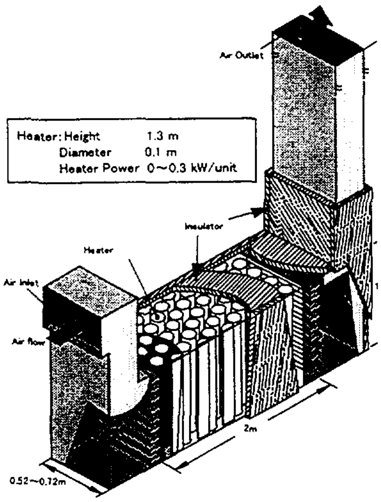


FIG. 3. Test equipment (Vault Storage System)

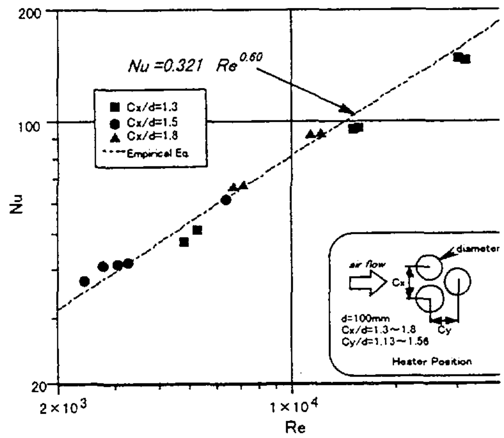


FIG. 4. Relation between Re and Nu (Vault Storage System)

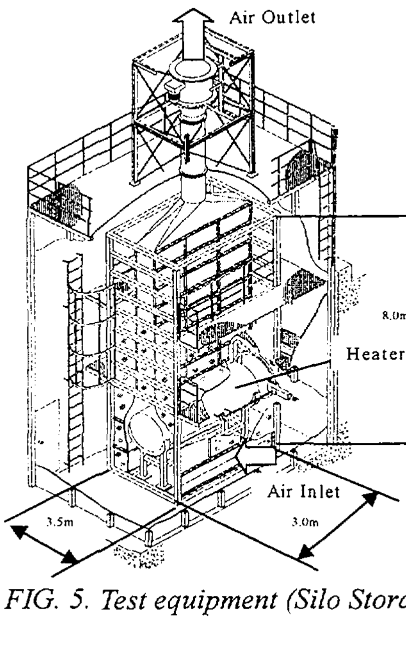


FIG. 5. Test equipment (Silo Storage System)

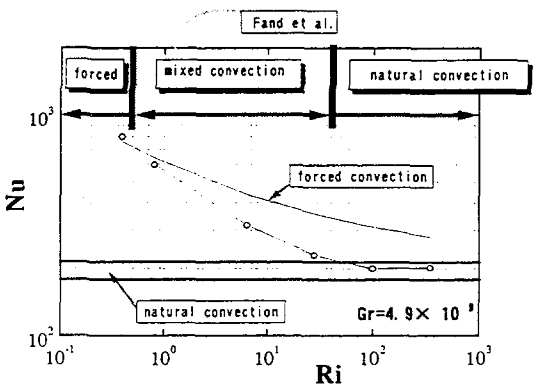


FIG. 6. Relation between Ri and Nu (Silo Storage System)

IAEA-SM-352/8P

# HEAT TRANSFER STUDY ON DRY VAULT STORAGE SYSTEM



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Japan

## 1. INTRODUCTION

Spent fuel assemblies of commercial light water reactor have been stored in storage pools in Japan. The national programme intends to reprocess the spent fuel in Japan. However, it seems that the amount of discharged fuel will exceed the reprocessing capacity, because the next commercial reprocessing plant project has been decided to be prolonged, and so on. It becomes necessary to store spent fuel in the storage pool or at another type of facility, until it can be reprocessed. Some utilities are planning to increase the storage capacity by construction of additional pools or storing the spent fuel in dry metal casks.

IHI constructed a high level vitrified waste storage facility, which is a dry vault type with a natural convection air cooling system. The dry vault type is one of the interim dry storage systems that is expected in the future, because the natural convection air cooling needs no active systems and generates very little radioactive waste.

## 2. DRY VAULT STORAGE SYSTEM

Spent fuel assemblies are loaded in a cylindrical canister with inert gas, such as helium, to prevent the cladding from oxidation and to improve the thermal conductivity. Then, canisters are put into the storage pit that is surrounded by ventilation pipes. The decay heat of spent fuel is transferred by conduction and radiation from the inside of the canister to the outside and removed by the air flow between the canister and the ventilation pipe. Cooling air flows through the annular gap between the canister and the ventilation pipe. Buoyancy force is generated by the heated air. Cooling air flows as to balance with the resistance of the flow path. This concept is shown in Figure 1. This type has been selected because of the secure of air flow path which makes the air flow stable and all of the canisters could be cooled homogeneously.

An example of the dry vault storage facility is shown in Figure 2. The spent fuel storage area is set up underground. Cooling air flows into the inlet shaft, rises in the storage area and is released to the atmosphere thorough the outlet shaft.

## 3. HEAT TRANSFER TEST

The heat transfer tests have been performed with the simulated dry vault storage system apparatus to verify the heat transfer aspects of the canister. The experimental model consisted of a 1 m high section of a canister containing a simulated BWR fuel assembly with channel box. These tests were carried out taking notice of the behaviour of the thermal convection in the canister. It is expected that the temperature of the fuel cladding will be reasonably evaluated by considering thermal convection inside and around a fuel assembly. The canister shell was a stainless steel cylinder, with a

890 mm diameter and 15 mm thick. The simulated fuel assembly was contained within the canister. The bottom plate of the canister was welded to the canister shell. The top cover of the canister sealed against the outside of the canister to provide a helium (0.1 MPa) or a vacuum atmosphere within the canister. The decay heat of each fuel assembly was simulated by electric heaters. The simulated fuel assembly consisted of stainless steel tubes that simulates fuel rods, containing a sheathed heater that is filled with powder of magnesium oxide. The simulated fuel assemblies were covered with a rectangular channel box with the same dimensions as an actual BWR fuel channel box.

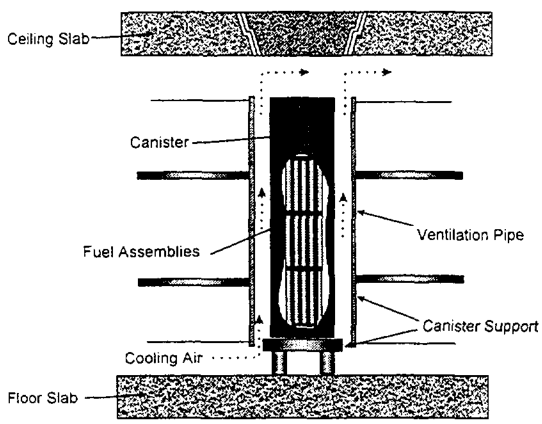
The test canister was installed into the ventilation pipe. Cooling air flew through the annular gap between the canister and the ventilation pipe. A schematic configuration of the experimental apparatus is shown in Figure 3. The cooling airflow was supplied by a ventilator and heated by a pre-heater. The cooling air temperature and velocity was controlled by using a PID method.

## 4. ANALYTICAL STUDY

The analytical heat removal calculations were carried out for every test case. A commercial computational finite element code was used because of its capability for combined convection and conduction analysis for predicting the canister internal heat transfer. The modelling work of the test canisters were 3-dimensional simulations of the internal heat transfer inside the test canister. The measured temperature of the canister wall was applied to the boundary condition.

## 5. CONCLUSION

Experimental and analytical studies were carried out using the test canister and the computational finite element code. The comparison of the temperature distribution between experiment and analysis showed good results.



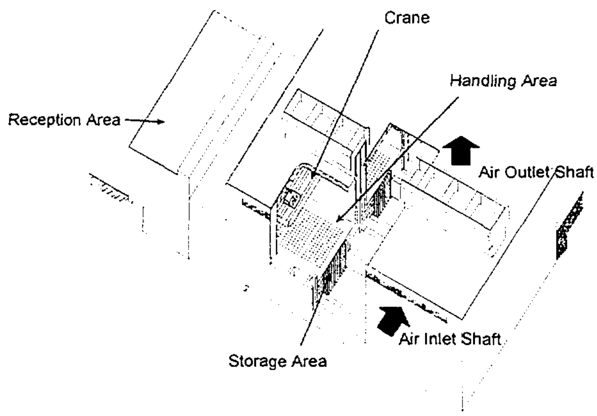




*FIG. 1. Cooling concept of dry vault storage system* *FIG. 2. An example of vault storage facility*

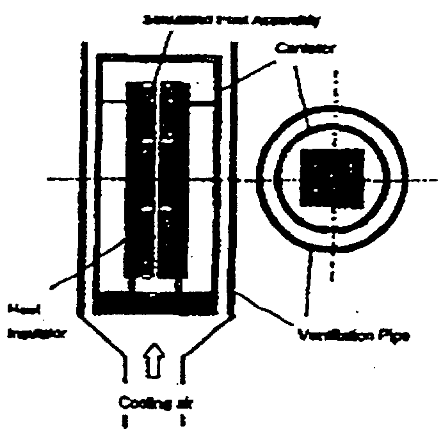



*FIG. 3. Test arrangement*

# DETERMINATION OF BURNUP, COOLING TIME AND INITIAL ENRICHMENT OF PWR SPENT FUEL BY USE OF GAMMA-RAY ACTIVITY RATIOS

XA9951821

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## 1. INTRODUCTION

The burnup, cooling time and initial enrichment of spent nuclear fuel are basic parameters to characterize the radionuclide inventory, decay heat output and fissile material of the spent fuel. It is very important to determine these parameters with a high level of confidence, because of its impact on safety and costs during spent fuel management operations, such as storage, transportation, reprocessing and disposal. For the non-destructive determination of these spent fuel parameters, radiometric methods have been developed in the USA, Europe and Japan. These methods are based on a combination of the neutron emission and gamma measurement techniques.

The Korea Atomic Energy Research Institute (KAERI) has been developing the algorithms for the sequential determination of cooling time, initial enrichment and burnup of the pressurized water reactor (PWR) spent fuel assembly by only use of the gamma ratios measurements, i.e., $^{134}Cs/^{137}Cs$, $^{154}Eu/^{137}Cs$ and $^{106}Ru\ ^{137}Cs/(^{134}Cs)^2$. This method has advantages over combination techniques of neutron and gamma measurement, because of its simplicity and insensitivity to the measurement geometry.

For verifying the algorithms developed, an experiment for determination of cooling time, initial enrichment and burnup of the two PWR spent fuel rods have been conducted by use of the high-resolution gamma detector (HPGe) system only. This paper describes the method used and interim results of the experiment.

## 2. METHOD

Fig. 1 shows the flow chart for the burnup measurement algorithm used in this work. As shown in Fig. 1, the algorithm can be divided into three parts, activity ratio calculation including the regression analysis of the activity ratios, the activity ratio measurements by gamma spectroscopy and the determination of the spent fuel parameters such as the cooling time, initial enrichment and the burnup.

### 2.1. ORIGEN-S calculations

Activity ratios of $^{134}Cs/^{137}Cs$, $^{154}Eu/^{137}Cs$, and $^{106}Ru^{137}Cs/(^{134}Cs)^2$ in PWR spent fuel were calculated using the ORIGEN-S code with various spent fuel parameter input data; (1) enrichment (2.5~5.0 wt%), (2) cooling time (0~20 years), (3) burnup (10~60 GW·d/tU), and (4) fuel type (14x14, 16x16, 17x17). With the regression analyses of the calculation results, the activity ratios were correlated as a function of burnup (*bu*), cooling time after irradiation (*t*), and initial enrichment (*en*).

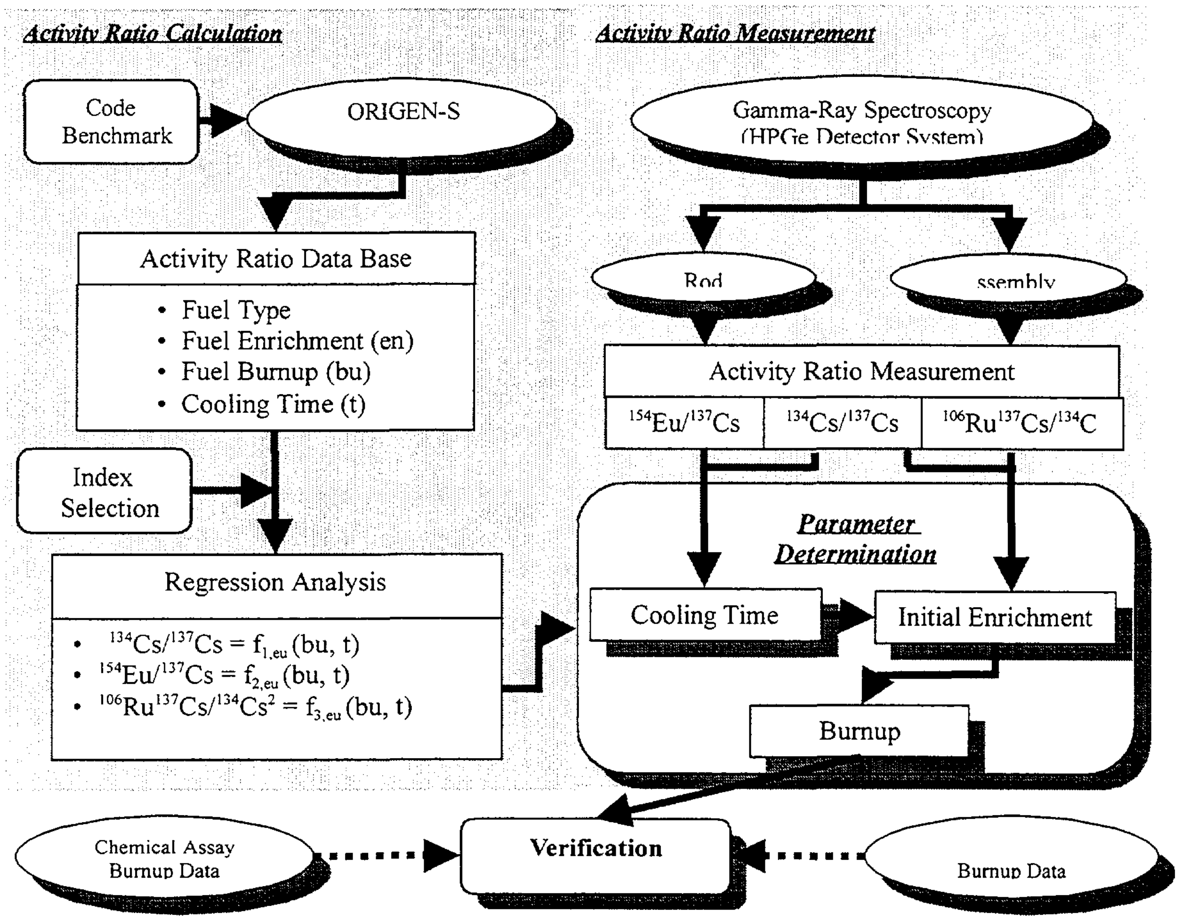


*FIG. 1. Flow chart for the burnup measurement algorithm*

$$^{134}Cs/^{137}Cs = \{P_1(en) + P_2(en)\cdot bu + P_3(en)\cdot bu^2\}e^{-(\lambda_{34}-\lambda_{37})t} \tag{1}$$

$$^{154}Eu/^{137}Cs = \{Q_1(en) + Q_2(en)\cdot bu + Q_3(en)\cdot bu^2\}e^{-(\lambda_{54}-\lambda_{37})t} \tag{2}$$

$$\ln[\,^{106}Ru\,^{137}Cs/(^{134}Cs)^2] = R_1(en) + R_2(en)\cdot\ln(bu) - (\lambda_{06} + \lambda_{37} - 2\lambda_{34})t \tag{3}$$

where *Pi(en)*, *Qi(en)* and *Ri(en)* are the regression coefficient at initial enrichment *en* and the activity ratio *i*.

### 2.2. Cooling time determination

At first, the burnup of the spent fuel was calculated with various cooling times and initial enrichments, putting the measured isotope ratios of $^{134}Cs/^{137}Cs$ and $^{154}Eu/^{137}Cs$ into Eqs. (1) and (2). The difference of the burnup calculated by Eqs. (1) and (2) should be minimal at the true values of the cooling time and the initial $^{235}U$ enrichment. For the computational analysis of the burnup differences, an estimator $F$ was introduced which is defined as:

$$F \equiv \frac{1}{N}\sum_{i=1}^{N}\frac{|\,B_i(Cs) - B_i(Eu)\,|}{\overline{B_i}} \tag{4}$$

where i = *1* to $N$ is the index for the gamma ray measurement positions and $B_i(Cs)$ and $B_i(Eu)$ are the measured burnup using the activity ratios of $^{134}Cs/^{137}Cs$ and $^{154}Eu/^{137}Cs$, respectively. Therefore, $F$ is an average fractional difference between the burnup calculated using $^{134}Cs/^{137}Cs$ and

$^{154}Eu/^{137}Cs$. The characteristics of $F$ are insensitive on initial enrichment variations and exponentially dependent on the cooling time. Using these characteristics, the cooling time of spent fuel can be very sensitively determined as the value which minimizes $F$ with the given initial enrichment ranges [1].

### 2.3. Initial enrichment determination

With known cooling time of the spent fuel, the initial enrichment can be determined in a similar way as in the cooling time determination step, using measured gamma-ray activity ratios of $^{134}Cs/^{137}Cs$ and $^{106}Ru^{137}Cs/(^{134}Cs)^2$.

For the determination of the initial enrichment, another estimator $F'$ is introduced, defined as:

$$F' \equiv \frac{1}{N}\sum_{i=1}^{N}\frac{|\,B_i(Cs) - B_i(Ru)\,|}{\overline{B_i}} \qquad (5)$$

where $B_i(Ru)$ is the measured burnup using the activity ratio of $^{106}Ru^{137}Cs/(^{134}Cs)^2$.

Using the cooling time (previously determined) and the measured gamma-ray activity ratios of $^{134}Cs/^{137}Cs$ and $^{106}Ru^{137}Cs/(^{134}Cs)^2$, one can plot the $F'$ value as a function of the initial enrichment with Eqs. (1), (3) and (5). The initial enrichment can be determined as the time value of the minimum $F'$ value.

### 2.4. Burnup determination

If the cooling time and the initial $^{235}U$ enrichment are determined, one can easily determine the burnup at each measurement position using Eqs. (1)-(3) [2].

## 3. EXPERIMENT

The gamma-ray spectrometric experiments were carried out on the two PWR spent fuel rods using a high purity Ge(HPGe) detector system installed in the KAERI hot cells. The first rod (Rod A) irradiated in Kori Unit-1 (14x14 fuel type) has a discharge burnup of 39 GW·d/tU (operator declared rod average burnup), a initial enrichment of 3.2 wt% and a cooling time of 8.45 years as of measurement time. While the other rod (Rod B) irradiated in Kori Unit-2 (16x16 fuel type) has a discharge burnup of 35 GW·d/tU, a initial enrichment of 3.5 wt% and a cooling time of 5.3 years as of measurement time (see Table I).

TABLE I. SPECIFICATION OF EXAMINED SPENT FUEL

| Spent Fuel | Discharged Date (Cooling time) | Initial Enrichment (wt%) | Operator Declared Burnup (GWd/tU) |
|---|---|---|---|
| Rod A (14x14 PWR) | January 1989 (8.5 years)* | 3.2 | 39 |
| Rod B (16x16 PWR) | May 1992 (5.3 years)* | 3.5 | 36 |

* As of measurement time.

4. RESULTS

### 4.1. Cooling time

Fig. 2 shows the cooling time as a function of the initial enrichment. This figure indicates a very weak correlation between initial enrichment and the cooling time. The average cooling time of Rod A and B over initial enrichments from 2.5 to 5.0 wt% were calculated to be 8.76 years and 5.22 years, respectively. These values are in good agreement with the operator declared cooling times within 2.5 %.

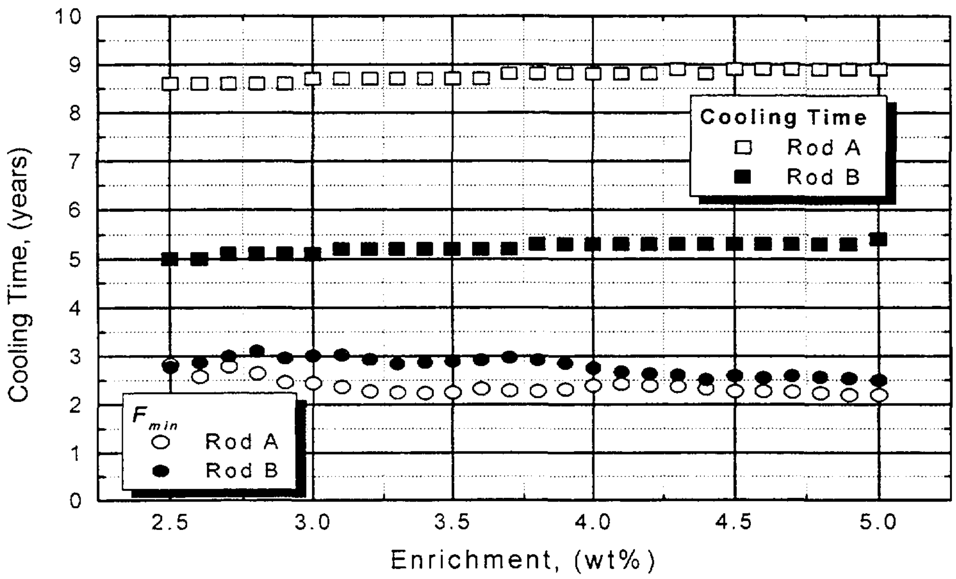


*FIG. 2. Cooling time as a function of initial enrichment*

### 4.2. Initial enrichment

Fig. 3 shows an estimator *F'* of Rod A and B as a function of initial enrichments with the previously determined cooling time of 5.22 years. It indicates, that *F'* of Rod B is minimum at 3.6 wt% initial enrichment, i.e., the initial enrichment of Rod B was determined to be 3.6 wt%, which is in good agreement with the operator declared initial enrichment of 3.5 wt% within 2.9 %. However, the determination of the initial enrichment of Rod A failed. The main reason is likely caused by the uncertainty from the $^{106}$Ru inventory from the ORIGEN-S calculation.

### 4.3. Burnup

Putting the cooling time and initial enrichment value previously determined into Eq. (2), the averaged burnup of Rod A and B were easily determined to be 41.3 and 34.6 GW·d/tU, respectively, which differ from the operator declared burnup of 39 and 35 GW·d/tU by 5.9 and 1.1% (see Fig. 4).

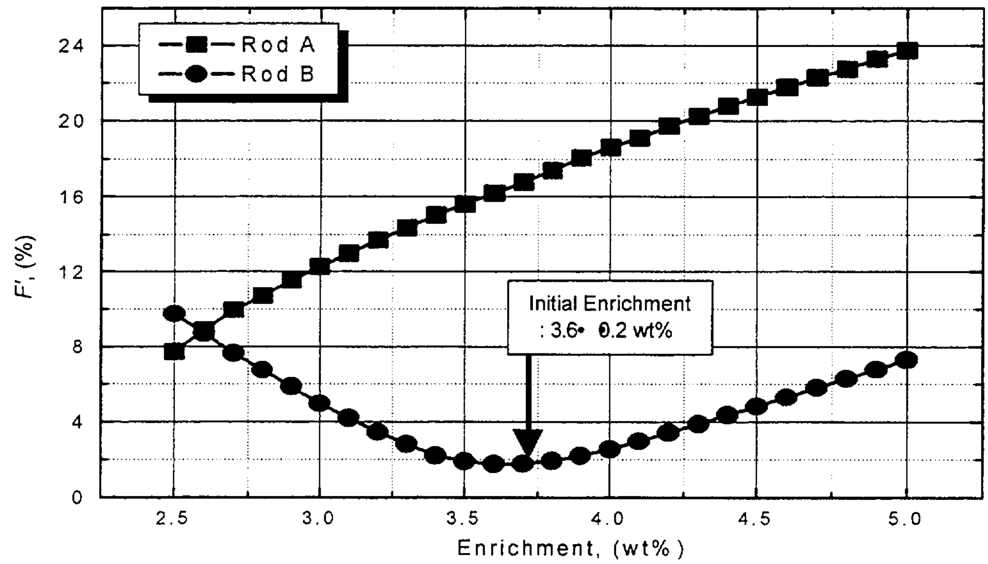


*FIG. 3. Initial enrichment determination using estimator F's of Rod A and B.*

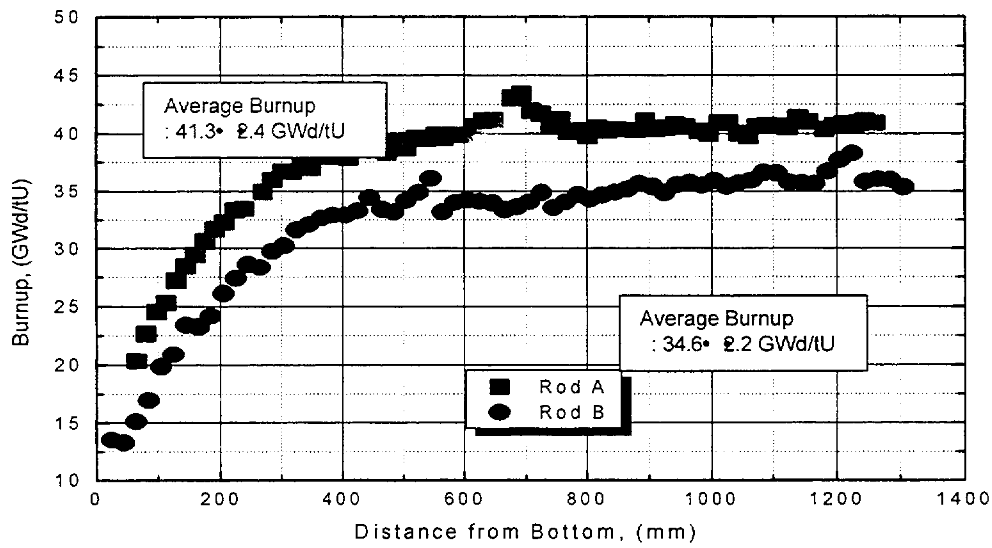


*FIG. 4. Burnup distribution of Rod A and B*

## 5. SUMMARY

Experimental results for PWR rod A and B are summarized in Table II. It indicates that the method (or algorithms) has a good reproducibility, i.e., within 2.5 % for the cooling time and 6 % for the burnup. However, the initial enrichment of Rod A was not determined, probably due to the uncertainty of the $^{106}$Ru inventory from ORIGEN-S calculation.

It is preliminary demonstrated that this technique is a very useful tool in burnup determination without information on the cooling time and initial enrichment of the spent fuel. This technique can be applied to spent fuel characterization, burnup credit and safeguards of the spent fuel management facility.

For the evaluation of the fuel assembly burnup, further examinations of the PWR spent fuel assembly, improvement of the algorithms including the $^{106}$Ru benchmarking and the axial/radial averaging routines are required.

TABLE II. RESULTS OF THE COOLING TIME, THE INITIAL ENRICHMENT AND THE BURNUP OF PWR SPENT FUEL RODS A AND B.

| Rod ID | Cooling Time (years) | Initial Enrichment (wt%) | Burnup (GW·d/tU) |
|---|---|---|---|
| Rod A | | | |
| - Measured | 8.76 (2.4%)* | - | 41.3 (5.9 %) |
| - Operator declared | 8.45 | 3.2 | 39 |
| Rod B | | 5 | |
| - Measured | 5.22 (-1.9 %) | 3.6 (2.9 %) | 34.6 (-3.9 %) |
| - Operator declared | 5.3 | 3.5 | 35 |

*: Numbers in parenthesis stands for the differences in % between the measured and the operator declared values.

## ACKNOWLEDGEMENT

This work has been carried out under the R&D Programme of Ministry of Science and Technology (MOST), Korea.

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# STUDY FOR THE SELECTION OF A SUPPLEMENTARY SPENT FUEL STORAGE FACILITY FOR KANUPP



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## 1. INTRODUCTION

The Karachi Nuclear Power Plant (KANUPP) is a 137 MWe CANDU type reactor. It has been in operation since 1972. Irradiated fuel discharged from the reactor is stored in a spent fuel storage bay at the reactor site. This bay can hold a maximum of 23,760 fuel bundles. So far more than 14,000 bundles have been discharged in the bay and 2,277 fuel bundles are resident in the core. To meet the safety requirements, space for one full core must always be available in the bay. Further, it is planned to extend the operating life of the plant to 2015. With the present fuel consumption rate around 1,000 bundles will be discharged per year. As a result, the space available in the spent fuel bay would be filled by mid of 2004. The duration available would be even shorter, as some space must always remain available to permit manipulation of the fuel trays and/or fuel bundles for various purposes. About 18,000 spent fuel bundles are expected to be discharged in future so that a total capacity for about 35,000 spent fuel bundles is needed. The capacity of the spent fuel storage bay is quite inadequate to meet the future needs and must be increased to hold an additional 12,000 bundles to be discharged in twelve years (2004 - 2015). Thus, now is the appropriate time to start the preliminary design work and make the supplementary storage available at the beginning of 2004, for an average annual loading of 1,000 bundles. To select the most suited option for a supplementary storage facility, various concepts and practices for storing spent fuel have been reviewed and steps of implementation are given in this paper. Comparison and evaluation of existing facilities indicate that the concept of ***concrete canisters*** near the KANUPP site is a promising choice. In order to prepare a preliminary design of these canisters, the characterisation of KANUPP spent fuel has also been carried out.

## 2. STEPS OF IMPLEMENTATION

The steps of implementation for construction of the spent fuel storage facility are the following:

### Step-1. Choice of conceptual design and site selection

Selection of the conceptual design is of prime importance. It will depend on the characterisation of the irradiated fuel bundles, review of the prevailing practices for storing the spent fuel and the evaluation criteria for the suitability of a particular conceptual choice. The characterisation study of irradiated KANUPP fuel bundles will provide the needed reference material and define the Source Term. The site selection is influenced by the choice of the conceptual design. A detailed site characterisation is essential for inclusion in the preliminary safety analysis report (PSAR).

### Step-2. Preliminary design and preparation of PSAR

The preliminary design of the facility follows the conceptual design and results in the form of a Preliminary Safety Analysis Report (PSAR) for submission to the regulatory body for approval. This report includes information regarding the nature, characteristics and purpose of the facility to allow the licensing body to review and evaluate the facility for safety of construction and operation. The design activities may be divided into the various facets, as the structural design and demonstration, shielding and heat transfer analysis and the design of fuel handling equipment. Work may be undertaken simultaneously on all of these facets.

**Step-3. Construction of the facility and preparation of FSAR**

For construction purposes all the technical specifications and sequence of operations is contained in the PSAR. After/during construction of the facility, the Final Safety Analysis Report (FSAR) would be submitted to the regulatorv body for approval which incorporates all the changes in the PSAR.

**Step-4. Testing/commissioning and loading of the storage facility**

Successful testing of the storage facility would allow its loading with spent fuel subject to the approval by the regulatory body.

## 3. CHARACTERISATION OF THE KANUPP IRRADIATED FUEL BUNDLES

The spent fuel characterisation is essential for design of the storage facility and would be used in evaluating the suitability of a particular type of facility. This includes calculation of fission product and actinide inventory, variation of decay heat and gamma spectra with cooling time, burnup and estimation of neutron sources. Such studies have already been carried out for KANUPP-fuel [1].

### 3.1. KANUPP fuel

The fuel for KANUPP is basically quite similar to the fuel of other CANDU reactors [1]. It consists of 19 cylindrical elements assembled into a bundle forming two concentric rings of 6 and 12 elements around a central element The bundle elements are equal in diameter and length and are spaced 0.119 cm from each other. Total bundle length and diameter are 49.609 and 81.40 cm, respectively. The end plates and spacers maintain separation between the fuel pins. The fuel element consists of uranium dioxide in the form of pellets sheathed in Zircaloy-4. The total U-mass per bundle is 13,395 kg.

### 3.2. Decay characteristics of the irradiated fuel bundles

The decay characteristics of KANUPP irradiated fuel bundle depend strongly on its burnup and irradiation history. The irradiation histories processed by the computer code ORIGEN (with some modification in the associated data library to incorporate the effect of neutron spectrum on neutron cross sections) provided results consisting of fission product and actinide inventory, the decay heat and gamma spectra as a function of cooling time, which are subsequently used for characterisation of the fuel bundles.

As a case study, decay characteristics of a bundle irradiated in the inner zone (maximum burnup of 8,826 MW·d/tU) and a bundle irradiated in the outer zone (minimum burnup of 4,645 MW·d/tU) are presented. The activities of long lived fission products and actinide concentrations at the time of discharge of the fuel bundles are given in Table I and II, respectively, while the photon release rate from fission products after 1,000 and 2,000 days of fuel bundle discharge is given in Table III. The variation of the $^{137}$Cs activity shows the same linear relationship with burnup as was observed experimentally [2]. The relationship can be expressed as:

$$^{137}\mathrm{Cs(Curies)}=0.043\cdot \mathrm{BU}\ (\mathrm{MW\cdot d/tU}) + 15.964$$

TABLE I. ACTIVITY OF LONG LIVED FISSION PRODUCTS (IN CURIES) AT THE TIME OF DISCHARGE OF FUEL BUNDLES

| **Fission Product** | **Half life** | **Bundle Irradiated in** | |
|---|---|---|---|
| | | Outer Zone | Inner Zone |
| $^{85}Kr$ | 10.59 y | 25.1 | 39.0 |
| $^{90}Sr$ | 28.50 y | 164.0 | 262.0 |
| $^{95}Zr$ | 65.2 d | 9,320.0 | 2,020.0 |
| $^{105}Ru$ | 39.6 d | 7,260.0 | 857.0 |
| $^{106}Ru$ | 369.0 d | 1,520.0 | 2,400.0 |
| $^{110m}Ag$ | 255.0 d | 2.02 | 5.38 |
| $^{125}Sb$ | 2.73 y | 18.0 | 35.1 |
| $^{129m}Te$ | 33.3 d | 335.0 | 29.1 |
| $^{134}Cs$ | 2.08 y | 113.0 | 309.0 |
| $^{136}Cs$ | 13.65 d | 105.0 | 17.9 |
| $^{137}Cs$ | 30.1 y | 216.0 | 396.0 |
| $^{140}Ba$ | 12.83 d | 8,760.0 | 1,070.0 |

TABLE II. ACTINIDE CONCENTRATION (IN GRAMS) IN THE KANUPP IRRADIATED FUEL AT THE TIME OF DISCHARGE

| **S. No.** | **Nuclide** | **Bundle Irradiated in** | |
|---|---|---|---|
| | | Outer Zone | Inner Zone |
| 1 | $^{235}U$ | 43.6 | 23.0 |
| 2 | $^{236}U$ | 7.66 | 10.7 |
| 3 | $^{238}U$ | 1,310.0 | 13,200. |
| 4 | $^{237}Np$ | 0.248 | 0.535 |
| 5 | $^{238}Pu$ | 0.257 | 0.108 |
| 6 | $^{239}Pu$ | 29.3 | 35.0 |
| 7 | $^{240}Pu$ | 7.41 | 13.6 |
| 8 | $^{241}Pu$ | 1.93 | 4.97 |
| 9 | $^{242}Pu$ | 0.355 | 2.01 |
| 10 | $^{240}Am$ | 0.0197 | 0.179 |
| 11 | $^{241}Am$ | 0.0147 | 0.169 |
| 12 | $^{242}Cm$ | 0.00285 | 0.0117 |
| Total | U-metal initially in the fuel | 13297.0 | 13461.0 |

TABLE III. PHOTON RELEASE RATE AFTER 1,000 AND 2,000 DAYS FROM SPENT FUEL BUNDLE IRRADIATED AT OUTER AND INNER ZONE

| **Energy Group** | **Average energy** (MeV) | **Photon Release Rate($\gamma$/s) from fuel bundle irradiated at** Outer Zone | | Inner Zone | |
|---|---|---|---|---|---|
| | | 1,000 days | 2,000 days | 1,000 days | 2,000 days |
| 1 | 0.30 | 5.5E12 | 7.0E11 | 6.0E12 | 9.0E11 |
| 2 | 0.63 | 1.9E13 | 9.0E12 | 2.0E13 | 1.5E13 |
| 3 | 1.10 | 6.0E11 | 1.0E11 | 9.5E11 | 2.5E11 |
| 4 | 1.55 | 1.1E11 | 1.8E10 | 2.0E11 | 5.0E10 |
| 5 | 1.99 | 8.0E10 | 6.5E9 | 9.0E10 | 2.0E10 |
| 6 | 2.38 | 9.0E9 | 9.0E8 | 1.2E10 | 3.5E9 |
| 7 | 3.05 | 9.5E7 | 1.1E7 | 1.6E8 | 9.0E7 |

### 3.3. The toxic life and value of the spent fuel

The toxic life of CANDU fuel, as defined in [3], is said to be at least a quarter of a million years ($2.5 \times 10^5$ years). Hence, it requires very long safe storage and supervisory control. Spent KANUPP fuel contains valuable materials which might have to be recycled to enhance the nuclear fuel resources. Therefore, at this stage the spent fuel can not be considered a final waste.

## 4. CURRENT PRACTICES FOR SPENT FUEL STORAGE

There are two major concepts of storing the spent fuel from nuclear power plants namely 'wet storage' in which the fuel is water cooled, and the 'dry storage' in which the stored fuel is cooled by air or a gas.

It is most convenient, and a universal practice, to store freshly discharged spent fuel in an on-site, water pool. This is a temporary storage which may last for several years. These pools may be used for extended periods, if sufficient capacity is available. In other cases, additional, away-from-reactor or near the site, storage pools may be constructed.

Dry storage of aged spent fuel is increasingly used for the longer term (≥50 years). There are several types of dry storage facilities Based on the cooling techniques the dry storage of spent fuel can be divided into following two broad categories:

(1) Air cooling storage/inert atmosphere storage:
- Concrete canisters;
- Vault storage;
- Spent fuel storage and storage/transportation casks;
- Horizontal concrete modules;

(2) Heat sink cooling storage:
- Dry wells;
- Burial in the salt mines;
- Disposal in deep trenches.

Selection of one or the other depends upon relative merits and demerits and the particular situation.

## 5. EVALUATION OF THE CONCEPTS

The evaluation of the storage concepts, having potential application for KANUPP, is based on following ground rules:

1) Supplementary storage facility should serve as an interim facility for ≥ 50 years;
2) The fuel must be fully and easily retrievable at any time during the interim storage;
3) The oldest spent fuel would be transferred out from the bay to the supplementary storage facility;
4) The facility must meet the latest safety standards and should be economical.

In addition, the following characteristics are also desirable in the proposed storage facility:

- The facility should be relatively simple;
- No active cooling should be required;
- Adequate safety should be available with minimum operation and maintenance requirements;
- No secondary waste should be produced in the form of filters etc.;
- It should be suitable for easy transformation to an ultimate storage.

After comparison of various storage types, it seems that construction of a dry storage facility based on concrete canisters at KANUPP site is a suitable option to enhance the storage capacity due to the following facts:

- Selection of a site in the KANUPP premises has several advantages. Firstly, the available KANUPP site investigation report will ease the job of PSAR preparation. Secondly, this would reduce the transportation cost and involve minimum safe-guard problems;
- Some of the KANUPP fuel bundles have already been under water for over 25 years. For an interim storage of $\geq 50$ years, the water pool may not be suitable because behaviour of the spent fuel for such a long duration under water is not known yet. Therefore, dry storage option is more suitable;
- Out of the dry storage methods the concrete canisters provide a better solution for the KANUPP spent fuel. The size of KANUPP-fuel bundle is such that it can easily be canned into the steel baskets and placed into concrete canisters;
- Canisters are of passive nature and need no active cooling;
- Fuel is fully and easily retrievable from the canisters;
- There would be no secondary waste generation from the canisters;
- The canisters are relatively easy in construction due to their simplicity;
- The module-ability allows the construction of canisters per requirement. This eliminates the possibility of high initial investment;
- The canisters are cost economical, based on a proven technology and meet the latest safety standards;
- The canister based facility can be expanded in future by adding more canisters to accommodate all the spent fuel from the KANUPP bay. Thus it can become a possible alternate interim storage for the entire KANUPP fuel;
- With an appropriate optimisation the canisters can be designed for direct transport to an ultimate storage at a later stage.

## 6. CONCLUSIONS

Construction of a storage facility based on concrete canisters at the KANUPP site is a suitable option to enhance the spent fuel storage capacity.

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# INTERNATIONAL LONG-TERM INTERIM STORAGE FOR SPENT FUEL; AN INDEPENDENT STORAGE SERVICE INVESTOR MODEL

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Private investors in the USA successfully demonstrate how the unclear situation of the Jucca Mountain repository project can be used to make profit by offering interim storage services for spent fuel to the American utilities.

Thinking more globally the obvious world-wide demands for **large storage capacities** for spent fuel within the next decades and the newly arising demands for **long-term** interim storage of spent fuel urges to respond by international interim storage facilities of high capacity [1].

To find investors acting as independent operators of such international interim storage facility needs mainly a storage technology which enables safe storage at low costs for periods exceeding hundred years.

Low cost storage can be achieved only by arranging the interim storage facility underground in a suitable host rock formation and by selecting the geographical area of such an interim storage facility by an international competition under those countries, who are willing to offer their land.

However, competition is not the only consideration. Any country which makes an interim storage site available internationally must impute the operation of the facility to the international supervisory body, must establish that the selected site will be accepted by the local population and that the financial means made available for the development of the site will benefit local industry and the population. Not last this state must underwrite the necessary guarantees for contracts made between the power plant operation and the storage operating enterprise.

The investor and operator of an international interim storage facility selected and realised by a competition on the free market procedure as well as the country where the storage is built, are both bound by two different kinds of contracts. The main contract is between the offering storage country/region and the independent operator (Fig. 1) and is embedded by:

- The atomic law of their country, including the Non-Proliferation Act;
- The IAEA Joint Convention on Safety of Spent Fuel Management (including Earliest Public Acceptance concept);
- The Basel Convention on the Control of Transboundary Movements of Hazardous Wastes.

The independent operator has in addition a series of contracts with various utilities, which are interested to have their spent fuel stored for a longer period.

The independent storage investor model was calculated on a conceptual design of such an underground storage facility. Economic calculations demonstrate that an upper limit of storage costs is \$ 44/kg U [2]. It is expected to reduce this figure significantly by the competition of low level industrial wages in, say, east European countries.

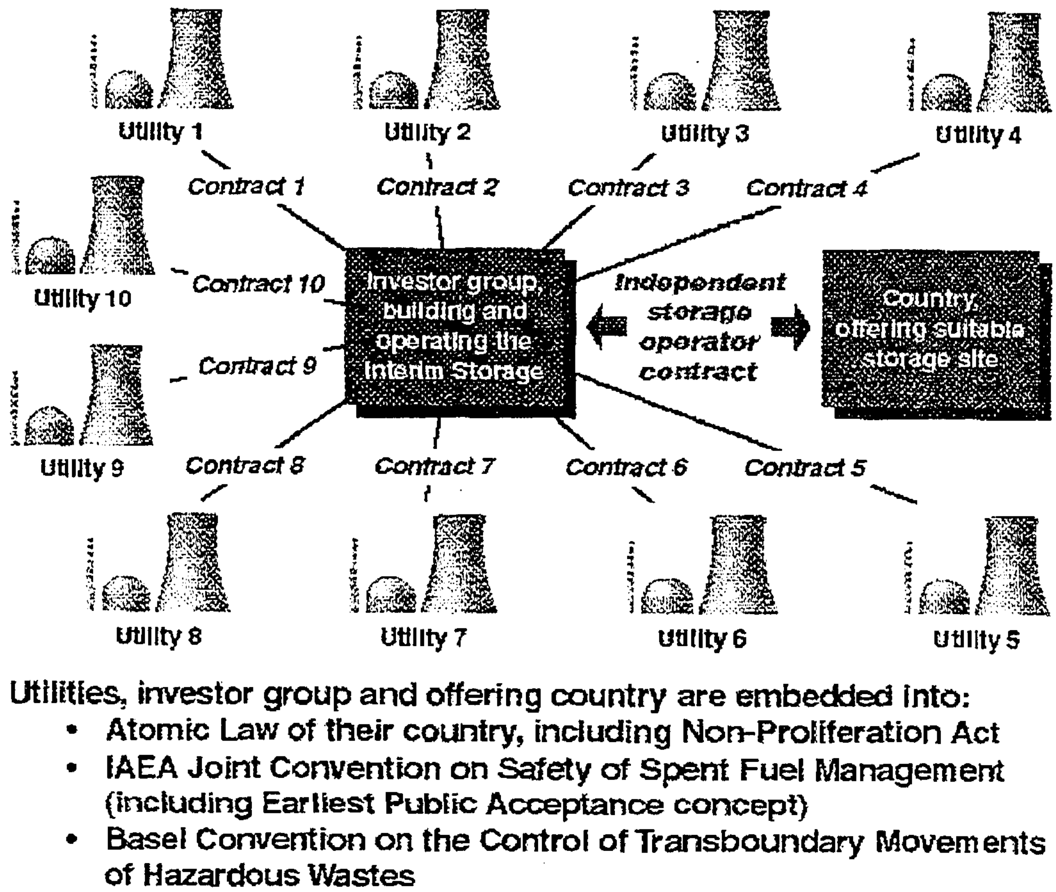


*FIG. 1. Model for independent interim storage facility*

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IAEA-SM-352/12P

# SCALE USER'S MANUAL FOR WWER-RELATED APPLICATIONS



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The issues of spent WWER fuel transport and storage, handling, consolidation at a storage facility and consolidated fuel handling and storage disposal at the repository, can be dealt with using the SCALE Code System (ORNL,USA).

A new guidance document for the SCALE code system aimed at WWER-related applications is presented as a poster. The document includes numerous examples using WWER fuel and describing problems specific to the WWER application field. The problems involve parameters of WWER-440 fuel assemblies and their arrangement as well as geometry and materials of a modeled container similar to the CASTOR 440/84 dry storage cask or the C30 container for the spent WWER-440 fuel. Most of the sample cases were prepared in collaboration with the SCALE development staff from the Oak Ridge National Laboratory. The guidance document was worked out thanks to support from the IAEA in Vienna (NEFW Nuclear Fuel Cycle and Waste Technology Division ) and understanding to the needs of the physicists working in the WWER application environment.

The original SCALE Manual of 4,500 pages was substantially shortened to give rise to this brief guidance document which facilitates problem orientation, mainly for the non English speaking users of Central and Eastern European countries operating WWERs.

The document could be a training tool to supplement the actual SCALE Manual and enables users with an interest in WWER applications to more readily perform accurate safety analyses. Thus, together with the SCALE Training Course held for the first time in Eastern Europe last year (sponsored by IAEA, OECD/NEA, and US NRC), the SCALE User's Manual for WWER-related Applications is a significant and efficient help to the specialists from the Central and Eastern European countries operating WWERs, who would like to carry out calculations of WWER spent fuel packages using the SCALE code system.

# THERMAL-HYDRAULIC ANALYSIS OF VVER-440 SPENT-FUEL STORAGE SYSTEMS

XA9951825

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In the framework of a contract between the IAEA and the KFKI Atomic Energy Research Institute, numerical models have been developed for the simulation of thermal-hydraulic behaviour of CASTOR type spent fuel store constructed in Dukovany NPP and MVDS type spent fuel store operated in Paks NPP with VVER-440 fuel. The model is based on the use of US code COBRA-SFS, which is well validated for spent fuel storage systems with Western type PWR and BWR fuel.

The COBRA-SFS code [1] performs thermal-hydraulic analysis of spent-fuel storage and transportation systems. It predicts flow and temperature distributions in spent fuel storage systems and fuel assemblies, under forced and natural convection heat transfer conditions. The code has detailed models of the basic processes taking place in dry storage facilities: it calculates the convective, conductive and radiative heat transfer regimes and determines the view factors for radiative heat transfer as well.

Two separate models were developed for CASTOR and MVDS store systems. However the fuel assembly simulation is common for the two cases, as both stores are used for the storage of VVER-440 fuel (Fig. 1). The hexagonal fuel assembly of VVER-440 reactor is simulated by fluid sub channels and fuel rods. The flow area of the assembly was divided into 37 channels. The radiative heat transfer exchange factors are defined within assemblies. The model describes the shroud of the fuel assembly as well.

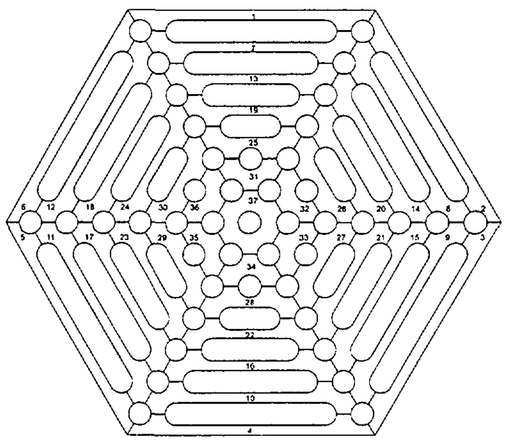

*FIG. 1. Fuel assembly simulation*

The CASTOR 440/84 spent-fuel storage cask consists of a rigged cast-iron body for structural integrity. The fuel basket is composed of hexagonal tubes made of borated steel 'atabor' in honeycombed arrangement. The casks are closed with two lids. The fuel assemblies are stored in helium gas environment. From modelling point of view the vertical arrangement the cask can be represented by one-sixth section of symmetry. The model developed for Dukovany CASTOR spent fuel storage system has been tested against the a real container measurements carried out at Dukovany NPP. On the basis of these assessment results, calculations were carried out for the design case and for the planned lifetime of the spent fuel storage system.

The CASTOR system was designed to store assemblies with maximum 250 W/assembly power. Total design power of the CASTOR is 21 kW with 84 fuel assemblies. In the present test this design power was set to each fuel assemblies of the model. The external temperature of the air were set to 40°C. The calculation provided detailed information on the temperature distribution of the system (Fig. 2).

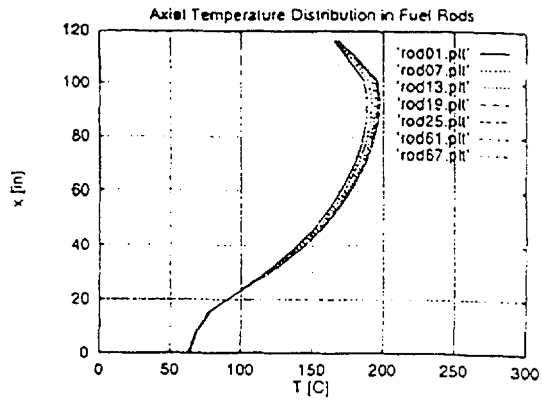


*FIG. 2. Temperature distribution*

In the MVDS (Modular Vault Dry Storage) the fuel assemblies are stored vertically in steel tubes. Each tube stores a single assembly. A passive self regulating cooling system induces buoyancy driven ambient air flow across the exterior of tubes and provides heat-removal from the store. In the model only a part of the whole vault module is described: six storage tubes were selected. The tubes are surrounded by concrete wall. The air flow is set to flow in on the left hand side at low elevations and to flow out on the right hand side at the top of the system.

In order to show the behaviour of the system under different boundary conditions sensitivity calculations were carried out, which included summer and winter conditions and also the effect of partial blockage. The results of calculations showed that the peak clad temperature follows the external temperature variation. In partial blockage cases the decay heat is driven away with increased outlet temperature.

The most important output of COBRA-SFS calculations can be the prediction of the maximum cladding temperature. To demonstrate the capability of the present model to solve such problems a series of calculations was performed for fuel assemblies with 3.6% enrichment for 50 years. The fuel had 40 GW·d/tU burnup and after 3 year wet storage was transported to the vault (Fig. 3).

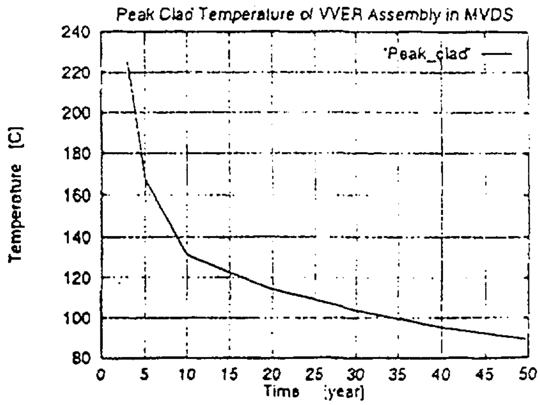


*FIG. 3. Cladding temperature*

The main results of this project are the CASTOR and the MVDS VVER-440 working models, which are available for the potential users through IAEA. The models have been tested and the results are reasonable. The models were constructed in such a way, that the new users could solve the typical spent fuel store thermal problems with small changes in the boundary and operational conditions.

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# ENCAPSULATION OF NUCLEAR FUEL RESIDUES FROM PIE AND FUEL TESTING AT STUDSVIK FOR INTERIM STORAGE AT CLAB IN SWEDEN

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A small quantity of the fuel irradiated in power reactors is often transferred to nuclear research facilities for further investigations. At STUDSVIK in Sweden, fuel testing programmes in the Material Testing Reactor (R2) and post irradiation examinations in the Hot Cell Laboratory have been going on since 1960. This work has generated a large amount of nuclear fuel segments and residues.

It is important that this type of waste can also be included in a Spent Fuel Storage programme. In 1986, a decision was taken to treat and encapsulate the accumulated older fuel waste as well as the waste currently generated at STUDSVIK to fit the interim storage for spent fuel (CLAB) and later disposal in the planned Deep Repository in Sweden. CLAB, which started operation in 1985, and the Deep Repository, which will be commissioned around the year 2020, are essential parts of the Swedish Nuclear Waste Management System.

The aim of the presentation is to describe the encapsulation technique adopted for the spent fuel waste, the treatment in STUDSVIK and the transport to and handling at CLAB for interim storage. The choice of canister " for enclosure of the fuel waste, and the design of a suitable transport and storage unit for a number of canisters, have **been made with regard to the** following demands:

- A canister type with due regard to the anticipated 40 year storage under water in CLAB and the subsequent treatment before final disposal in the Deep Repository;
- A transport module as well as a storage unit that makes it possible to be handled and stored like a fuel assembly at CLAB;
- The transport box with fuel waste should fit a transport cask owned by STUDSVIK for the shipment to CLAB. The same cask is used to transport full length fuel rods and other core components from the Swedish nuclear power stations to STUDSVIK for examination purposes.

The canister for the fuel waste has an outer diameter of 89 mm, a wall thickness of 4 mm and a length of 1,110 mm. The material is stainless steel, austenitic type, SS2353 (AISI 316 L). The bottom lid is welded to the tube with filler material in a single -V butt joint. It is done in the premanufacturing performed in a workshop. After loading the canister with fuel waste in a hot cell, the top lid is TIG welded to the tube with a single -V butt joint without filler material. The top lid has a centre hole for evacuation and refilling with helium gas. After He-filling of the canister the centre hole is sealed by TIG-welding, and the canister is leak tested. The external key grip in the top end fits into the internal key hole in the bottom end. This makes it possible to dismount the canisters from the transport boxes and handle them as separate unit at the time for the treatment before final disposal.

Figure 1 shows the set-up for welding the top lid of the canisters in the hot cell. The chamber for evacuation, He-filling, centre hole welding and leak testing is shown in Figure 2. Twelve fuel canisters are loaded into a transport box as shown in Figure 3. It was found suitable to design the box to fit into the storage racks at CLAB which normally are used for PWR fuel assemblies. The cross section of the box is 212 mm square and the total length is 4,069 mm. The material is stainless steel SS2343 and SS2353 (AISI 316 and AISI 316L). Four parallel tubes are joined together by a sheet construction and a bottom piece. In each tube, three canisters can be loaded. The top end consists of a handle for the enclosure of the box and for handling with an available tool on the loading machine at CLAB.

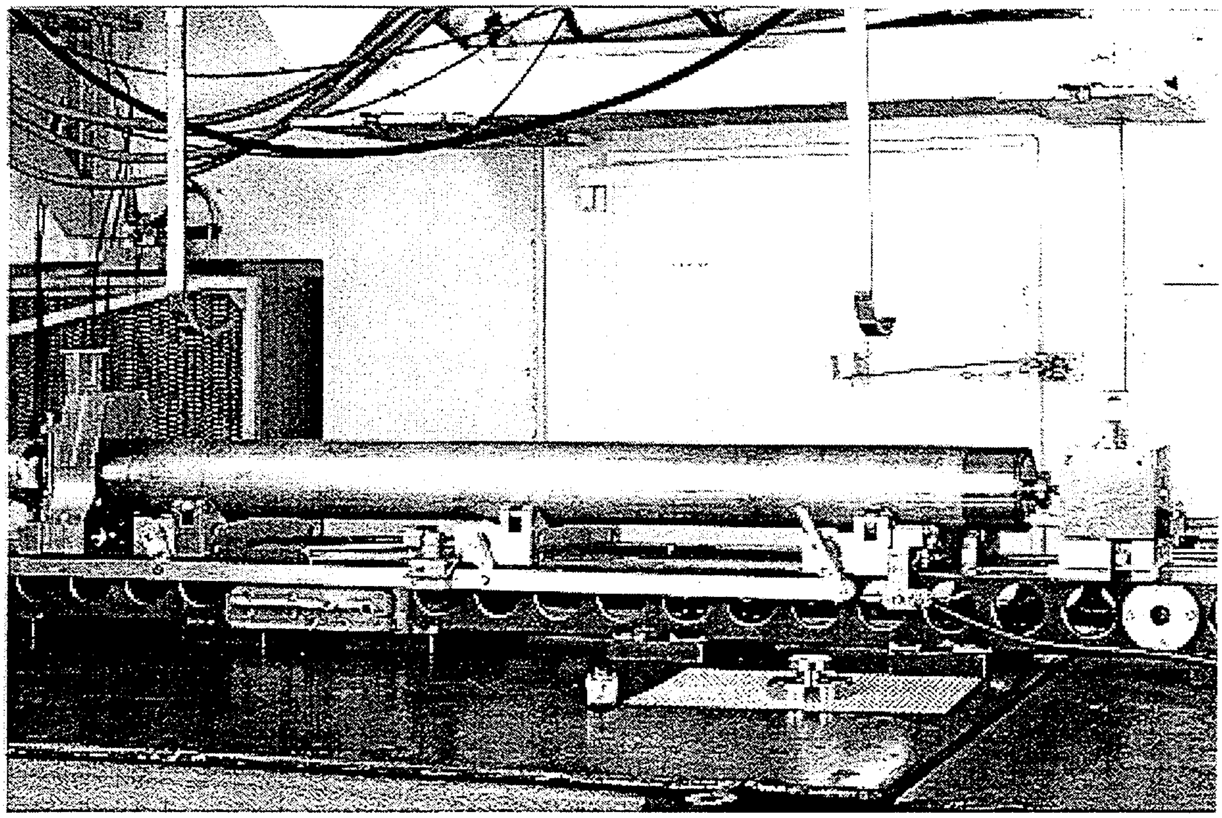

*FIG. 1. The test for welding the top lid of the canisters*

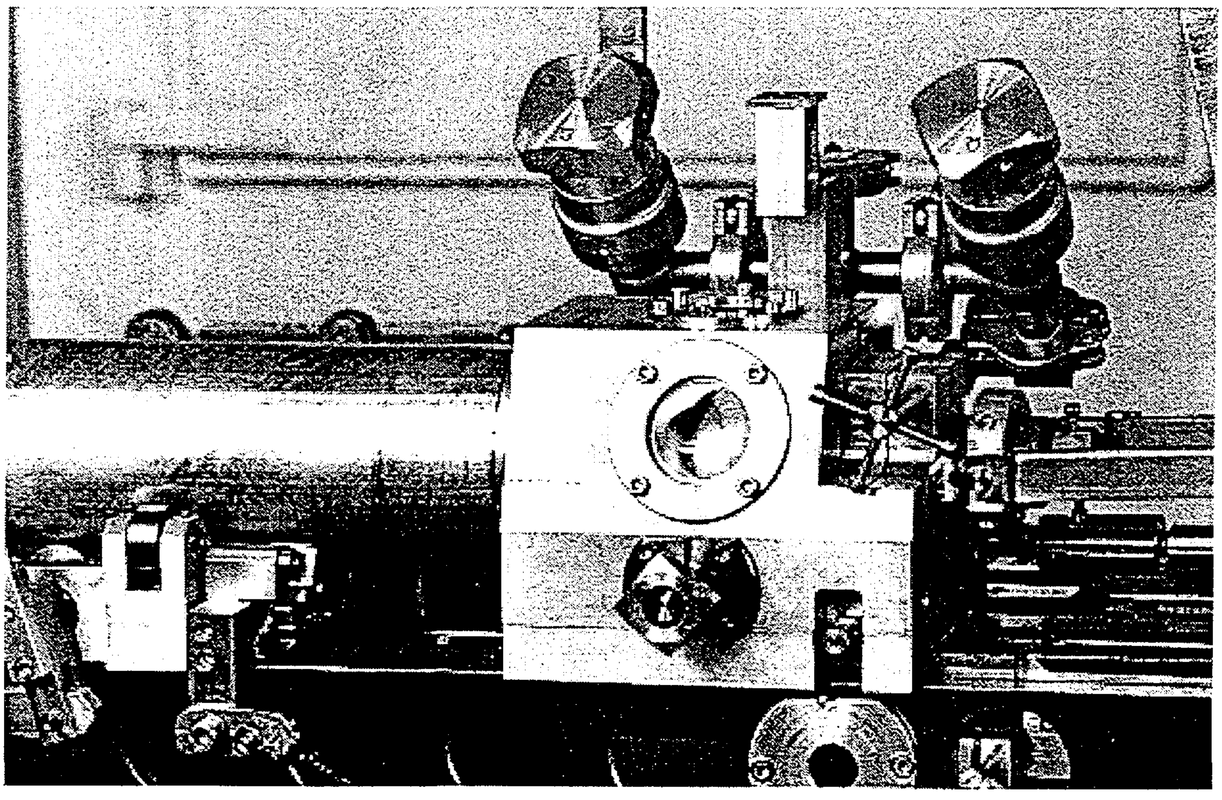

*FIG. 2. Chamber for evacuation, He-refilling, welding and leak testing*

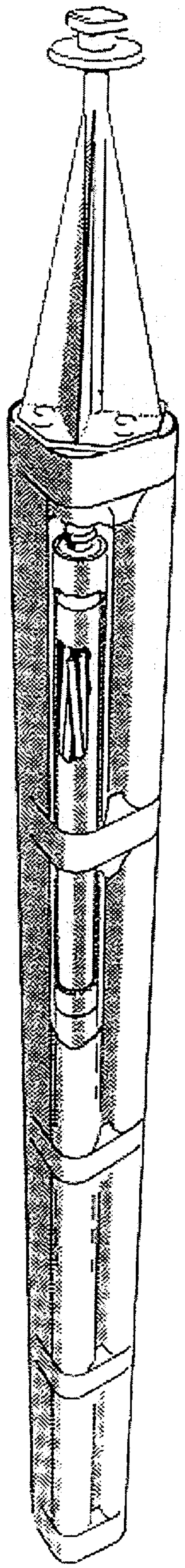

*FIG. 3. Transport box for nuclear fuel waste encapsulated in twelve canisters*

The manufacture of all canisters and transport boxes is subject to QA/QC procedures. The remaining quality control of the encapsulation process is performed in the hot cells. The licensing approval for the handling and storage of this nuclear fuel waste was granted to STUDSVIK and SKB (the Swedish Nuclear Fuel and Waste Management Company) in 1988. The first ~fer operation was performed in May 1989. Up to 1997, sixteen transport boxes containing 2,233 kg fuel waste have been transferred to CLAB for interim storage.

Most of the accumulated older waste stored at STUDSVIK has now been encapsulated. In the future the fuel waste subsequently generated in the Hot Cell Laboratory will be treated without short-term storage at STUDSVIK and transferred to CLAB.

IAEA SM-352/15P

XA9951827

# INVESTIGATION OF THE BEHAVIOUR OF WWER SPENT FUEL ROD IN DRY STORAGE CONDITIONS

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## 1. INTRODUCTION

The present report describes the of investigation the dry storage performance of spent WWER-440 fuel in the temperature range 300°C - 400° C. This study is aimed at accumulating reliable experimental data on the change of Zr+1%Nb alloy fuel cladding condition during the storage period with also regard of its important role thermal creep process of the cladding material.

## 2. PRELIMINARY STAGE

At the preliminary stage, a detailed test programme was developed, which considered the experimental conditions, the programme of pre- and post-test fuel rod characterization, the number and duration of test phases, the selection of a fuel assembly and rods for testing, measurement of their characteristics, a prediction of research results.

The necessary experimental conditions for the WWER-440 spent fuel storage testing were based on the thermophysical analysis fulfilled for the cask "CASTOR WWER-1000". According to the calculations, the use of 12 WWER-1000 spent fuel assemblies (FAs) with the total residual heat - 39 kW as a heating source might provide the required test rod temperature range 330 -400 ° C during the whole test period.

### 2.1. Selection of FAs and rods for testing; results of fuel characteristics measurements

The WWER-440 assembly selected for testing was standard and had operated in the reactor for 4 fuel cycles to a burnup of 3,668 MW·d/kgU (Table I).

TABLE I. OPERATIONAL HISTORY FOR FA, SELECTED FOR TESTING

| Parameter | Cycle No | | | |
|---|---|---|---|---|
| | 18 | 19 | 20 | 21 |
| Irradiation beginning | 09.12.89 | 27.06.91 | 14.07.92 | 09.11.93 |
| Fuel cycle finish | 04.01.91 | 30.04.92 | 05.06.93 | 01.11.94 |
| Cycle duration, EFPD. | 348,3 | 306,0 | 320,1 | 347,2 |
| Cell parameters | 14-25 | 15-32 | 15-32 | 01-46 |
| FA power, MW·d | 3,15 | 4,69 | 4,26 | 1,30 |

In compliance with the national standard programme for testing spent WWER-440 fuel, the following non-destructive investigations were carried out:

- visual inspection and photographing of the FA;
- check of fuel channel faces;
- measurement of forces applied for fuel channel tube removal .

On fuel channel tube removal the following actions were performed:

- visual inspection and photographing of fuel bundle;
- measurement of forces applied for fuel rods removal.

On completing the inspection and taking pictures, the fuel bundle was disassembled by pulling fuel rods through the spacers. The visual inspection showed satisfactory condition of the fuel bundle, no change of the bundle shape, no bends or deformations of the peripheral rods. The loss of the fuel cladding integrity was not indicated. Corrosion of the cladding, welds and fitting ends was not found either. After the preliminary inspection, several fuel rods were selected for preparation of dry storage testing. These rods were thoroughly inspected and the fragments of their fuel cladding photographed (Fig.1).

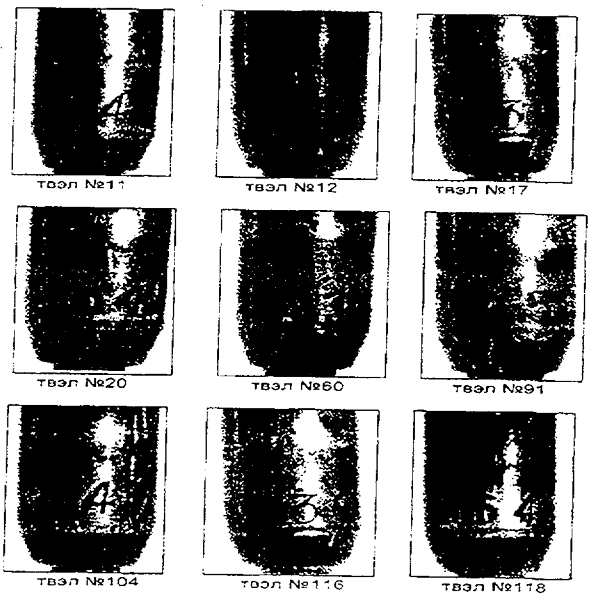



*FIG. 1. Fuel rod lower end pieces*

Fuel leak testing was carried out by measuring the release of noble gas activity (Kr-85, 87, 88, Xe-133, 135) in the air samples taken from the leak-tight cells of the canister. In addition, the following studies were performed:

- γ-scanning;
- outer diameter measuring over the full rod length;
- eddy-current defectoscopy of fuel cladding;
- measurements of the rod length;
- measurement of the gas quantity and free volume under the fuel cladding of the fuel rods neighbouring (in the fuel bundle) to those selected for testing.

Fig. 2 presents typical results of γ-scanning for one of 9 test rods. The diagram distinctly reflects the joint of fuel pellets and local decrease of FA spacers. As a whole, the condition of the $UO_2$ fuel was considered as satisfactory.

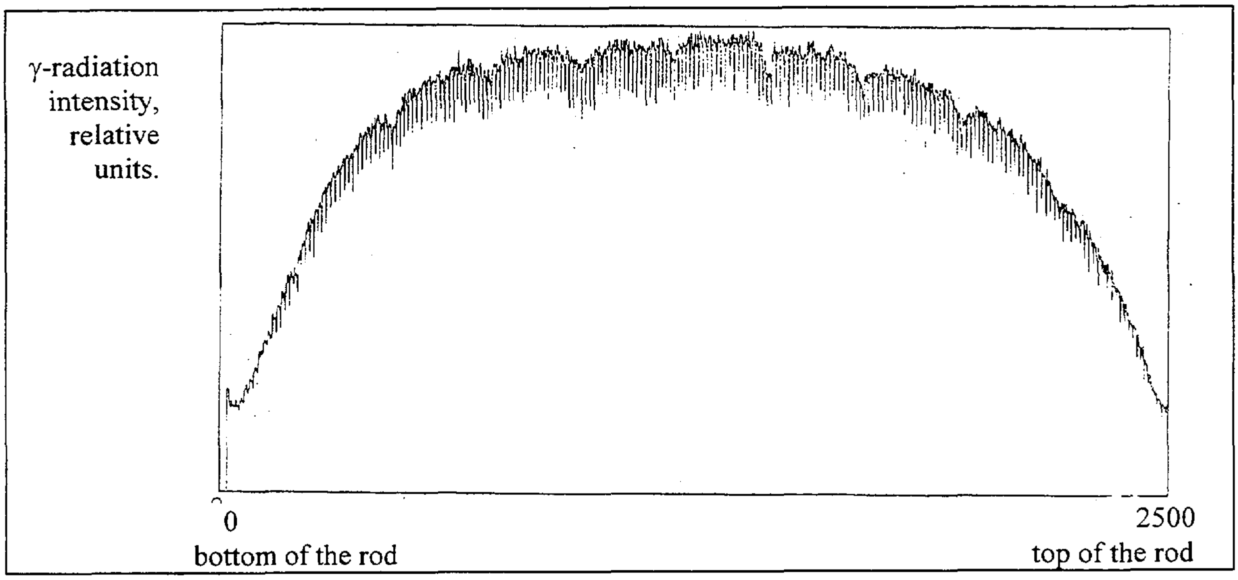



*FIG. 2. Gamma-scanning diagram of fuel rod 116*

The fuel rod length was optically measured by remote means through the peephole of the hot cell by comparing its length with that of a standard specimen. The confidence level of the length determination error is ±0.7mm. The rod diameter was assessed by two ways:

- eddy-current technique providing the full picture (profile) of the length wise diameter change;
- contact technique with the use of special profilometer which allowed to measure the diameter at a given section and define the ellipticity of the fuel cladding (at turning the fuel rod around its axis).

Combining these two techniques produced the required scope of reliable information. The diameter distributions over the test rod length are presented in Section 4.3. In all cases the average diameters show a - 0.1 mm smooth decrease from the ends toward the middle of the rod, which is characteristic for spent WWER fuel. The confidence limits of the diameter measurement error are ± 0.012 mm.

**2.2. Forecast of fuel rod dry storage performance**

The above mentioned forecast was based on the data, obtained in the pre-test characterization and in-reactor operation of the fuel rods selected for the long-term storage test. The results of the mathematical modelling of fuel rod behaviour by using the "PULSAR" Code, which was certified by the Federal Supervision Body of Russia on nuclear and radiation safety in 1996, showed that in the one year test conditions the change of the fuel rod diameter will not exceed 1 μm, which corresponds to the creep deformation of $10^{-2}$ at the elongation less than 1 mm.

3. FIRST PHASE OF DRY STORAGE TEST

On completing the pre-test characterization the fuel rods were equipped with thermocouples to provide temperature monitoring during the test (Fig. 3).

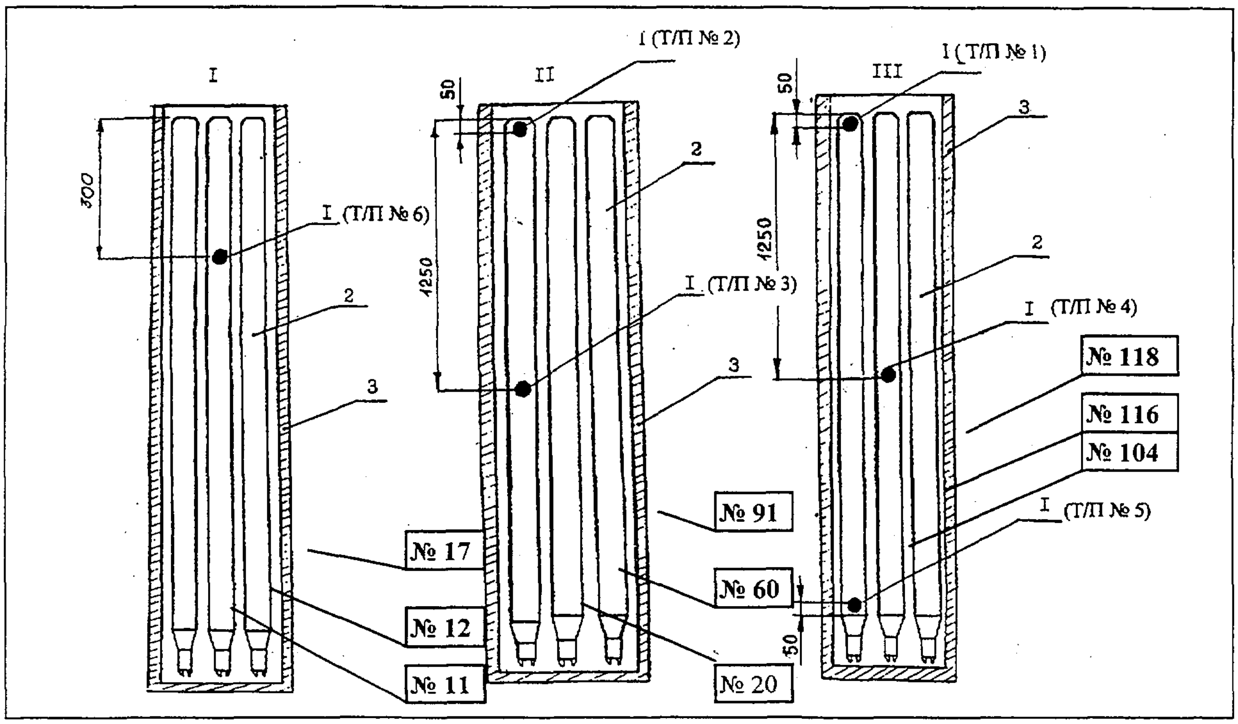
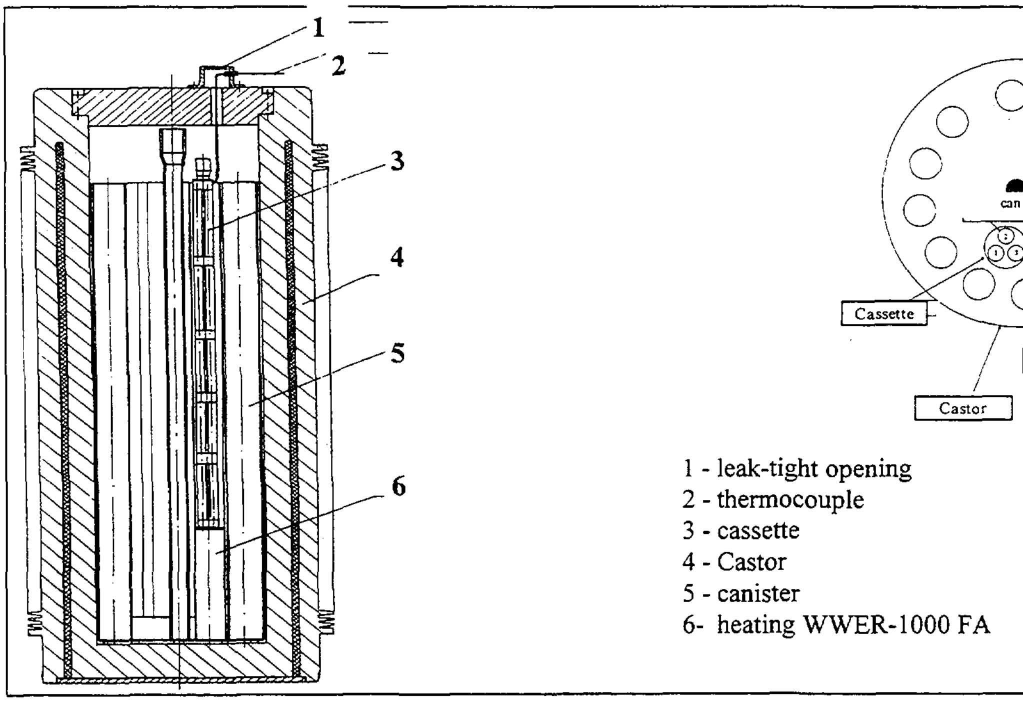



*FIG. 3. Thermocouple arrangement in cans with tested WWER-440 fuel rods*
*Т/П - thermocouple; I, II, III - numbers of cans with thermocouples*

The cask "CASTOR-WWER-1000" was loaded with 12 spent fuel assemblies of WWER-1000 type and a cassette containing 3 cans with 9 rods, 3 rods per can (Fig. 4).




1 - leak-tight opening
2 - thermocouple
3 - cassette
4 - Castor
5 - canister
6- heating WWER-1000 FA

*FIG. 4. Arrangement of heating WWER-1000 assemblies and can with tested WWER-440 fuel rods*

Next follows cask closure with the lid, connecting and adjustment of the secondary instruments for cask temperature and pressure monitoring, cask draining, evacuation, cask cavity drying, ventilation with Ar, 2nd evacuation and pressurization with Ar.

The temperature stabilization on the rod surface took place over 2 days with Ar excessive pressure in the cask cavity of about 1.4 kgf/cm$^2$, with temperatures of 350°C ,363°C and 380°C at the bottom, in the middle and at the top section of the end, respectively. The subsequent temperature monitoring confirmed the conclusions about the rod temperature stabilization (Fig. 5).

Periodic radiation monitoring of the gas medium in the cask pointed to the maintenance of the integrity of all heating and tested fuel rods during the 1st phase. The 1st phase lasted for 76 days and the cladding temperature at various section of the test fuel was within a range of 350 - 400°C.

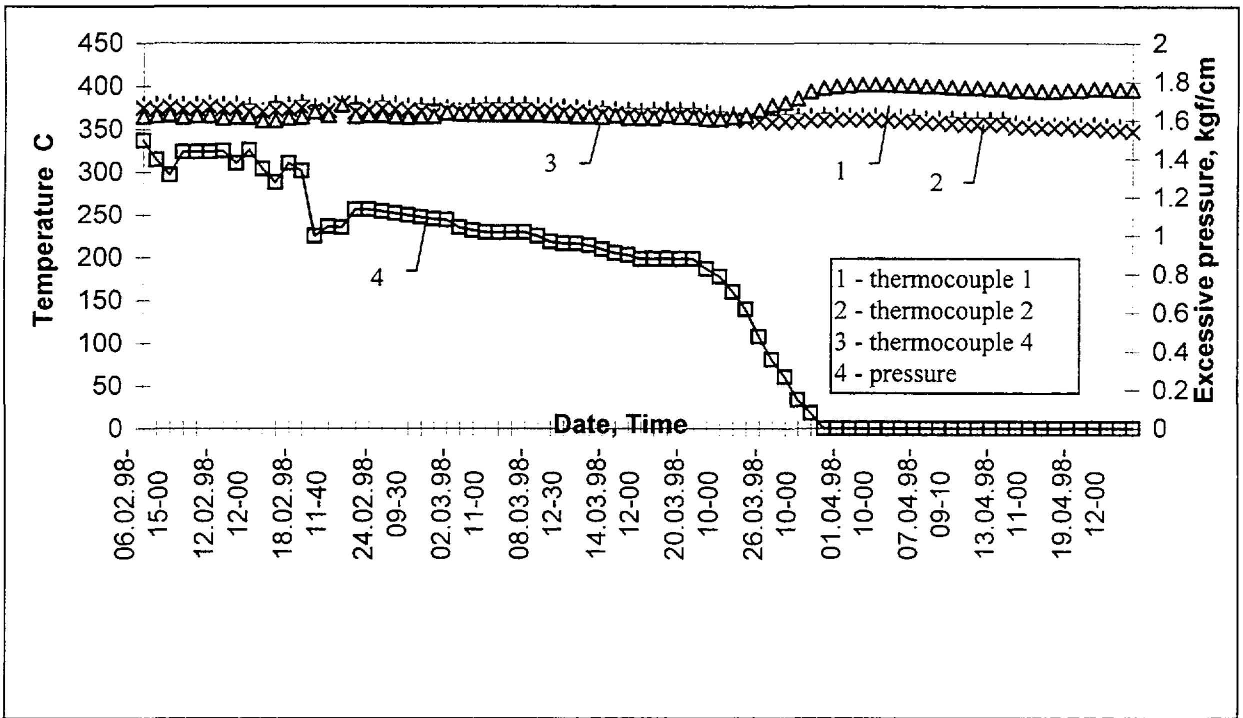



*FIG. 5. Test fuel cladding temperature and cask pressure during testing*

## 4. POST-TEST CHARACTERISATION OF WWER-440 FUEL RODS

On completing the 1st dry storage phase the canister with 9 fuel rods was transported to the hot cell for investigation.

### 4.1. Visual inspection of fuel rods

The inspection results were as follows:

- As is the case before testing, the surface oxide film has dark-gray color, which is typical for WWER-440 spent fuel;
- Beginning with distance ~ 1 m from the lower end fitting, the cladding of all fuel rods contained round white spots which had 2 mm diameter and varied in tint (from a dark core bounded with a district circle to a uniform white spot);
- The number of spots also varied from rod to rod. The rods 60 and 116 contained much more spots than other rods and the individual sections of rod 116 showed even spot merging and shelling of the white layer in the spot core.

Based on the visual inspection results for all 9 fuel rods, it was decided to carry out detailed non-destructive investigations of rods 11 and 12 which contained the least number of white spots and rods 60 and 116 with the greatest number of spots (among them merged and shelled spots).

### 4.2. Rod length measurement

The post-test values of ΔL (the parameter used as a reference for the fuel length control and representing the difference of the rod length and that of the jig in which the rod is stiffly gripped) are presented in Fig. 6.

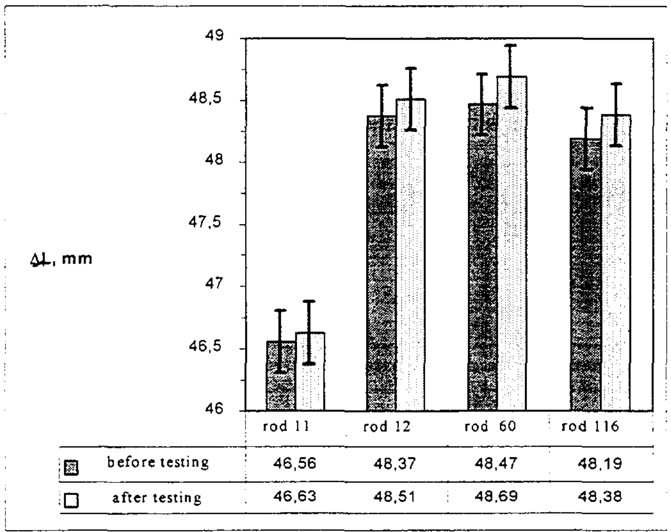



*FIG. 6. Results of rod's length measurement before and after testing(ΔL)*

The measuring error for this parameter is ± 0.25 mm. It can be seen, that in view of this measuring error, the rod length during the storage remained practically unchanged.

### 4.3. Rod diameter measurement

*4.3.1. Eddy-current method*

Comparing the pre-and post-test rod length wise diameter distributions revealed no noticeable changes of the rod profiles during the storage period.

*4.3.2. Contact method*

The pre-and post-test measurements of the rod diameter were performed with the use of the same technique at the same azimuthal length-wise positions ($0^0$, $60^0$, $120^0$ - orientation by the mark on the fuel cap at the distance of 1,000, 1,050 and 1,100 from the top fuel rod end). Three measurements were carried out for each of the length-wise and azimuth positions. In doing so the rod position was not changed. Then, the rod was removed from the grip and reinstalled back into the grip. After the rod repositioning three measurements were performed at each turning angle. The average diameter values were calculated for each azimuth orientation in each lengthwise position and in the section from 1,000 to 1,100 mm. The average diameter calculation errors for each measurement procedure were assessed as 10μm, 16μm and 10 μm, respectively. Comparing the pre-and post-test results of the average rod diameters calculations points to their practically entire agreement (Table II and Fig. 7).

TABLE II. AVERAGE FUEL ROD DIAMETER WITHIN 1,000 TO 1,100 MM ROD SECTION

| Rod number | Before testing | After testing |
|---|---|---|
| 11 | 8,995 | 8,995 |
| 12 | 8,998 | 9,000 |
| 60 | 8,989 | 8,989 |
| 116 | 9,010 | 9,011 |

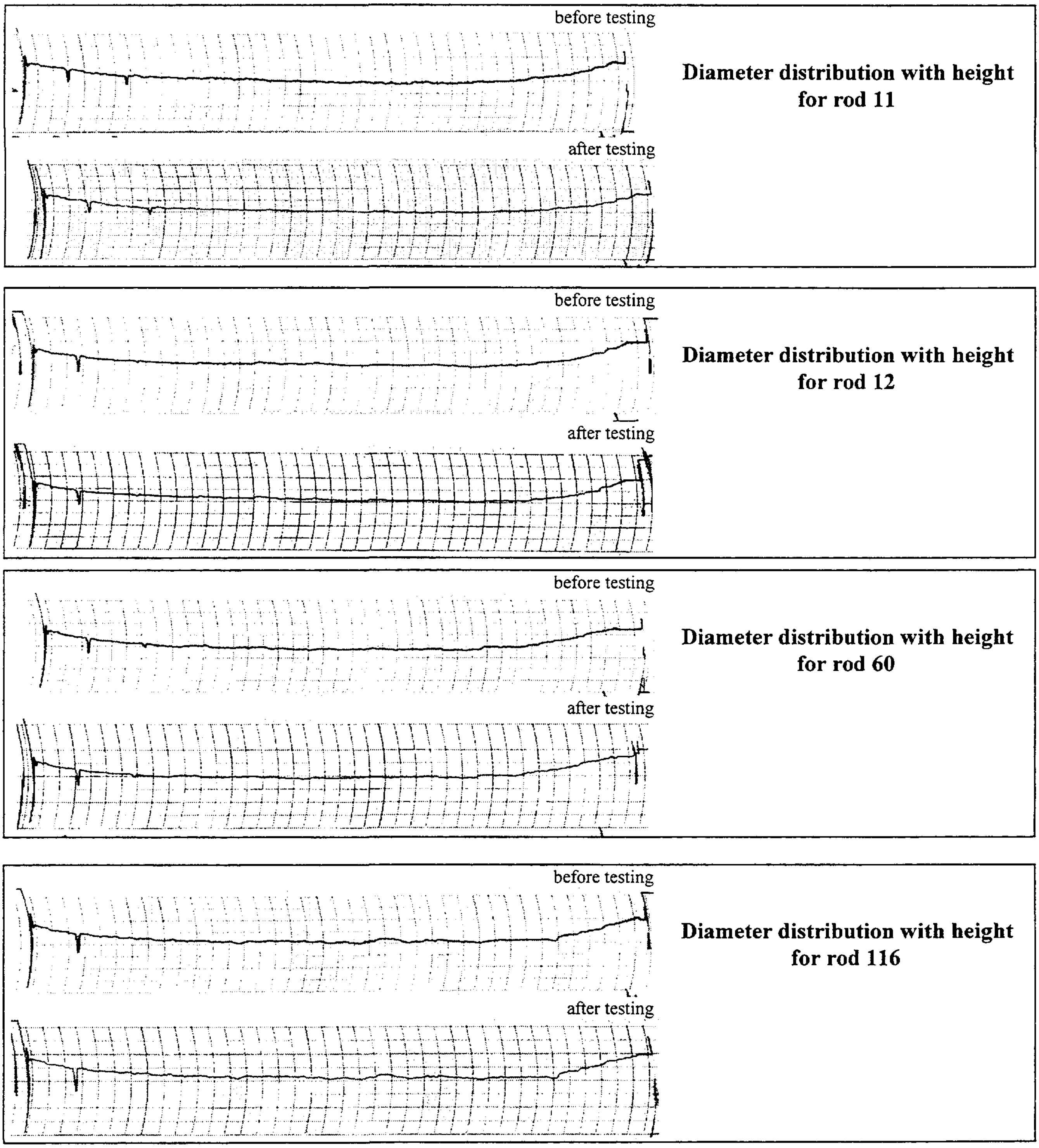



*FIG. 7. Diameter distribution with height for various rods*

### 4.4. Measurement of free volume and gas pressure under fuel cladding

The results of measuring the above parameters are listed in Table III for the rods 11, 12, 60 and control group of neighbouring rods from the same assembly. The values of the parameter agree well between these groups of rods; this fact points to the absence of any detectable fuel swelling and to only slight gas evolution from fuel during the test period.

TABLE III. FREE VOLUME, GAS VOLUME AND PRESSURE IN FUEL RODS

| Group of rods | Rod number | Free volume Vrod $sm^3$ | Gas volume Vgas $sm^3$ | Pressure MPa |
|---|---|---|---|---|
| Testing | 11 | 13,8±0,6 | 28,6±0,6 | 0,20 |
| | 12 | 12,9±0,4 | 29,0±0,7 | 0,22 |
| | 60 | 13,4±0,4 | 26,4±0,6 | 0,19 |
| Neighbouring | 14 | 11,9±0,3 | 26,6±0,6 | 0,22 |
| | 68 | 12,5±0,5 | 25,7±0,6 | 0,21 |
| | 69 | 12,7±0,4 | 35,9±0,8 | 0,28 |
| | 70 | 12,9±0,5 | 30,5±0,7 | 0,24 |
| | 80 | 13,1±0,3 | 30,7±0,7 | 0,23 |

## 5. CONCLUSION

The post-test investigation results for 4 (of 9) fuel rods revealed no noticeable change as compared to the pre-test measurements. This fact is in agreement with the forecast of Zr+1%Nb fuel rod performance in dry medium at 330-400 $^{\circ}$C.

Visual investigation showed the presence of oval white spots on the surface of all the tested rods. This is probably due to the oxidation of the cladding material. Further studies are needed, however, for the detailed characterization of the cladding condition.

In the future tests it is necessary to take into account that the behaviour of WWER-1000 spent fuel assemblies used as heating ones is of great interest. The current programme does not include investigation of the WWER-1000 spent fuel assemblies itself, which were used as heating elements. However, such investigations might give additional information proving the calculational results received by VNIINM in estimation of the maximum permissible temperature of WWER-1000 spent fuel assemblies under dry storage and could be used in safety analysis of SNF dry storage design basis and beyond the design basis accidents. Such storage facilities for WWER-1000 SNF are planned to be built at the NPP “Kozloduy”, Bulgaria, Zaporozhskaya NPP, Ukraine, and other nuclear power plants.

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## AUTHOR INDEX[1]



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## INDEX OF PAPERS AND POSTERS BY NUMBER

### Papers



### Posters

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| | | |
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