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# ***Spent fuel management: Current status and prospects 1997***

*Proceedings of a Regular Advisory Group meeting  
held in Vienna, 9–12 September 1997*



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## FOREWORD

Spent fuel management has always been one of the important stages in the nuclear fuel cycle and it is still one of the most vital problems common to all countries with nuclear reactors. It begins with the discharge of spent fuel from a power or a research reactor and ends with its ultimate disposition. Two options exist — an open, once-through cycle with direct disposal of the spent fuel and a closed cycle with reprocessing of the spent fuel, recycling of reprocessed plutonium and uranium in new mixed oxide fuels and disposal of the radioactive waste. The selection of a spent fuel strategy is a complex procedure in which many factors have to be weighed, including political, economic and safeguards issues as well as protection of the environment.

Delays in the implementation of the fuel reprocessing option in some countries, the complete abandonment of this option in other countries and delays in the availability of final spent fuel disposal in almost all countries has led to increasingly long periods of interim spent fuel storage. This “wait and see” approach gives more time and freedom to evaluate the available options and to select the most suitable technology. The problem of spent fuel management has therefore increased in importance for many countries.

Continuous attention is being given by the IAEA to the collection, analysis and exchange of information on spent fuel management. Its role in this area is to provide a forum for exchanging information and to co-ordinate and to encourage closer co-operation among Member States in certain research and development activities that are of common interest. Spent fuel management is recognized as a high priority IAEA activity.

The Regular Advisory Group on Spent Fuel Management was established in 1982. It consists of nominated experts from the Member States with considerable experience and/or requirements in spent fuel management. The country membership is selected in such a manner as to reflect the various spent fuel policies ranging from the closed fuel cycle to once-through concepts and includes representatives from both the developed and the developing nuclear power users. The objective of the Regular Advisory Group is to serve as a means of exchanging information on the current status and progress of national programmes on spent fuel management and to provide advice to the IAEA.

The results of the last Regular Advisory Group meeting (9–12 September 1997) are reflected in this report. It gives an overview of the status of spent fuel management programmes in a number of countries, a description of the current status and prospects of activities in this field and recommendations of the participants.

The IAEA wishes to thank all participants of the Regular Advisory Group meeting for their fruitful contributions. Special thanks are due to the Chairman of the meeting, J.R. Williams, and to M.J. Crijns who collaborated in preparing and editing this report. P.H. Dyck of the Division of Nuclear Power and Fuel Cycle is the officer responsible for the overall preparation of this report.

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## **SUMMARY OF THE ADVISORY GROUP MEETING: REVIEW AND RECOMMENDATIONS**

### **1. CURRENT STATUS AND PROSPECTS**

The Advisory Group observed that the activities related to the management of spent fuel continue to be of high priority in assuring the safe use of nuclear energy. The objectives to properly manage spent fuel are identical in all countries (e.g. safety, reliability for long term, public acceptance, safeguards criteria, environmental impacts, etc.). The technical solutions however are different. It was observed that the spent fuel management (SFM) strategy adopted by each country depends upon many factors which are country specific.

It was observed that the following different strategies for SFM are being pursued:

1. Reprocessing and recycling;
2. Direct disposal;
3. Deferral of decision while actively evaluating the strategies.

It was noted that some countries are implementing combinations of those strategies.

#### **1.1. Review of national programmes**

A common feature of most national situations, independent of the fuel cycle back-end options, is the ongoing need for additional storage capacity.

- Re-racking of existing reactor spent fuel storage pools is usually the first step in expanding storage capacity. Neutron absorbers, and, in a few cases, burnup credit have been implemented (e.g. Republic of Korea and Spain) to make more effective use of storage space in spent fuel pools.
- Some countries (e.g. Bulgaria, Finland, Russia and Slovakia) have expanded storage capacity by adding additional wet storage at reactor sites while many other countries (e.g. Canada, Czech Republic, Hungary, Japan, Republic of Korea, Ukraine and USA) are utilizing dry storage to expand capacity.
- Both wet and dry storage have been used for offsite storage. In the case of Sweden wet storage is in operation and, in the case of Germany, dry storage is used.
- Commercial reprocessing facilities and MOX fuel fabrication plants continue to operate satisfactorily. Treating the wastes arising from reprocessing and conditioning the residues with the goal to minimize the waste volume are making significant progress (e.g. France, India, Japan, UK).
- Other events that may impact spent fuel arisings and consequently storage needs include: increased burnup, premature reactor shut-down, extension of reactor lifetime, and re-use of spent fuel.
- The interrelation between the front end and the back-end of the fuel cycle has become obvious (e.g. higher burnup fuels with higher enrichment and MOX fuels may have an impact on the back-end).

Examples of developments that have taken place since the 1995 Advisory Group Meeting, are the following:

- In Canada two AFR (on site) dry interim storage facilities have been put in operation during 1995.
- After one year of trial operation, the dry storage facility at Dukovany (using dual purpose CASTOR casks) was licensed in January 1997 for a 10 years operational period.
- In France, the new high capacity MOX plant MELOX has reached its maximum licensed through-put as planned.
- In France it has been decided to load MOX fuel in 28 reactors, 12 of which are presently loaded.
- Hungary has constructed a modular vault dry storage (MVDS) facility which will be put in operation in 1997.
- The new Indian reprocessing plant at Kalpakkam has completed trial runs and the licensing process is expected to be completed before end of 1997.
- In Japan, it is anticipated that AFR spent fuel storage facility will be in operation by 2010. A programme is planned for utilizing MOX fuel in LWR beginning in 1999.
- The Republic of Korea is reviewing future options after the cancellation, due to geological reasons, of the earlier selected interim storage site.
- The Republic of Korea is proceeding with the construction of an experimental facility for the reuse (refabrication) of spent LWR fuel in a PHWR in a direct way (DUPIC).
- An interim wet storage facility has been put in operation at Smolensk NPP site in Russia.
- In South Africa, an AFR (on site) pipe storage facility for spent research reactor fuel has been put in operation.
- Sweden continues with the design of a plant for encapsulation of spent fuel prior to final disposal. Completion of a license application (PSAR) is planned for around 2001.
- Sweden has applied for a license for the expansion of the central storage facility (CLAB) to be put in operation by 2004.
- Following a successful commissioning phase, an operating license for THORP has been granted by the Regulatory Authority after a period of public consultation.
- The planning application by NIREX for the construction of a rock characterization facility at Sellafield was denied. Future options are under review.
- In Ukraine, a dry cask storage facility for use at the Zaporozhe site is under review by the Regulatory Authority.
- In the USA, three new dry storage facilities were placed in operation at nuclear power plant sites. Several transportable storage systems are under review by the US NRC for use at shut-down reactor sites. Proposed legislation to authorize an interim storage facility has received consideration by the US Congress.

It was noted that in 1997, the spent fuel arisings from all types of reactors in nuclear power plants amounted to about 10,500 t HM. The total amount of worldwide spent fuel accumulated at the end of 1997 was more than 200,000 t HM and projections indicate that the cumulative amount of spent fuel generated by the year 2010 may reach 340,000 t HM. About 130,000 t HM of spent fuel is presently being stored in at-reactor (AR) or away-from-reactor (AFR) storage facilities awaiting either reprocessing or final disposal. The quantity of accumulated spent fuel is over twenty times the present total annual reprocessing capacity. Assuming that part of the spent fuel to be generated in future will be reprocessed, the amount to be stored by the year 2010 is projected to be about 230,000 t HM. The first geological repositories for the final disposal of spent fuel are not expected to be in operation before the year 2010. Thus, the use of interim storage will be the primary spent fuel management option for the next twenty years in many Member States.

## 2. ADVISORY GROUP RECOMMENDATIONS

The Advisory Group concluded from the discussions of the country presentations that the back-end of the fuel cycle is a mature technology. It is recognized that new technologies for the management of spent fuel are under development and will be implemented in the future. The management of spent fuel is important for both operating and shutdown plants and will continue to have a high priority in assuring the safe and sustainable use of nuclear power.

The Advisory Group provides the following recommendations for consideration:

- The Advisory Group noted that there is increasing use of existing spent fuel storage facilities and/or an increase of the capacity of these facilities to support reactor operations. The Advisory Group notes the importance of understanding the environmental conditions to which the fuel is exposed in storage. Documentation of these conditions should help facilitate the granting of any licenses, etc.
- The Advisory Group observed that some countries have chosen long-term dry storage with a requirement for transportation and possible eventual disposal. The integration of spent fuel management activities, with particular emphasis on respective interface between storage, transportation and eventual disposal activities should be promoted.
- The Advisory Group noted the potential advantage of burnup credit at various stages of the back-end of the fuel cycle, such as storage, transportation and disposal, and recommends that studies be promoted to identify and describe areas of applications, collate relevant experience and where appropriate, encourage activities to improve knowledge and understanding.
- The Advisory Group recommends that consideration be promoted of implications of developments and decisions related to extended burnup and the use of MOX fuel in the back-end of the fuel cycle.
- The Advisory Group notes the importance of the research on spent fuel performance, particularly on the long-term integrity of spent fuel assemblies. The Advisory Group recommends to address the following considerations: 1) effects resulting from increasing burnup; 2) behavior of operationally defected fuel; 3) the ability to safely remove spent fuel after extended storage; and 4) licensing extensions for storage facilities.
- The Advisory Group recognizes that reprocessing is a mature technology and has a significant role in spent fuel management. Thus, evolving reprocessing technology should be kept under review.

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**Abstract**

The current status of the Canadian Spent Fuel Management is described. This includes wet and dry interim storage, transportation issues and future plans regarding final disposal based on deep underground emplacement in stable granite rock. Extension of wet interim storage facilities is not planned, as dry storage technologies have found wide acceptance.

**1.0 INTRODUCTION**

The Canadian nuclear programme is sustained by commercial type reactors which have been in operation since 1971. Ontario Hydro's CANDU reactors, representing 13,600 MWe of installed capacity are capable of producing about 92,000 spent fuel bundles (1800 tU) every year. Hydro Québec and New Brunswick Power each have a 685 MWe CANDU reactor, generating about 100 tU of spent fuel annually.

The typical CANDU fuel bundle contains natural uranium dioxide elements and weighs about 24 kg. Due to its relatively small size, its low burnup characteristics and its freedom from criticality hazards in light water, CANDU fuel is easily managed at the back end of its cycle.

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. The Canadian spent fuel management process is illustrated in Figure 1.

The spent fuel bundles are kept in at-reactor (RS) interim storage facilities. The implementation of various interim (wet and dry) storage technologies resulted in simple, dense and low cost systems. Summary of the spent fuel arisings is given in Tables 1-A and 1-B.

The most recent events which have taken place in Canada are as follows:

- In October 1995 Hydro Québec put into operation an Interim Storage Facility at Gentilly 2 NGS using AECL's concrete based vault type MACSTOR technology. This technology can also accommodate higher burnup spent fuel from light water reactors in a safe and economical manner;
- In November 1995 Ontario Hydro's used Fuel Dry Storage Facility (UFDSF) at Pickering NGS started up. The UFDSF is using an in-house developed Dry Storage Container (DSC) made with heavy concrete, which is also licensed for off-site transportation;
- Public hearings on AECL's permanent spent fuel disposal concept were organized by the federal government in Saskatchewan, Manitoba, Ontario, Québec and New Brunswick during 1996 and 1997. A report on the hearings is expected by the government before the end of 1997;
- Research and development activities for the immobilization and disposal of spent fuel at AECL's Whiteshell Laboratories in Manitoba are slowly progressing pending the outcome of the hearings and privatization initiatives concerning the facilities.

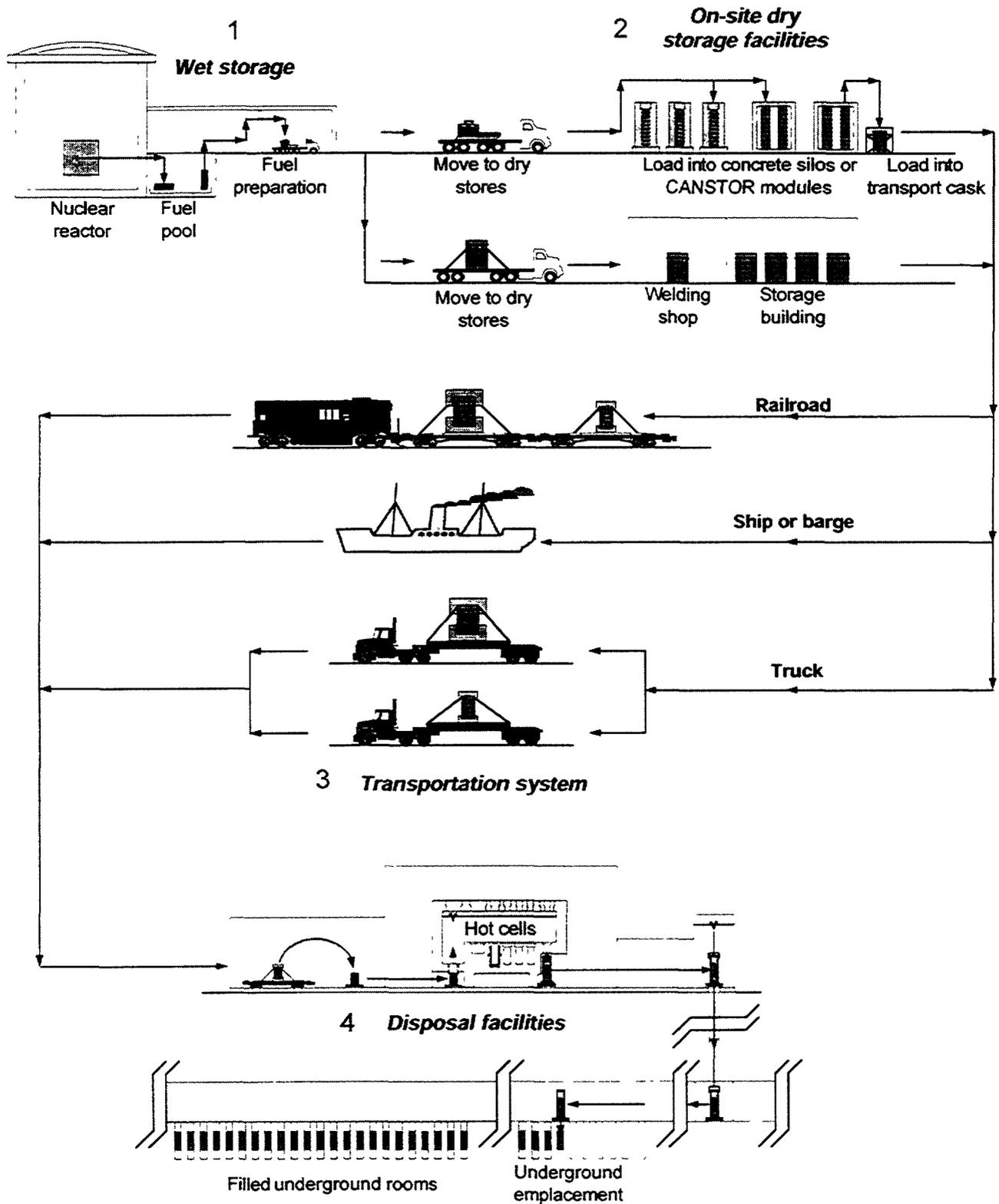


FIG. 1. Typical Canadian spent fuel management process.

TABLE 1-A CANADIAN "AT REACTOR" WET SPENT FUEL STORAGE FACILITIES  
(Status of wet storage facilities as of August 1997)

<i>Name of Facility</i>	<i>Location and Owner</i>	<i>Fuel Type (natural U)</i>	<i>Design Capacity (tHM)</i>	<i>Current Inventory (MTHM)</i>	<i>Planned Operating Life</i>	<i>Operating Period</i>
Gentilly 2 primary pool	Point Lepreau, New-Brunswick / New-Brunswick Power	CANDU 37 elements	Bundles: 52,896 (1,005 MTHM)	Bundles: ~ 50,000 (~ 950 MTHM)	30 years	In operation since 1983
Point Lepreau primary pool	Gentilly, Québec / Hydro-Québec	CANDU 37 elements	Bundles: 48,960 (930 MTHM)	Bundles: ~ 40,000 (~ 760 MTHM)	30 years	In operation since 1984
Pickering A primary pool	Ontario-Hydro Pickering	CANDU 28 elements	Bundles: 78,978 (1,580 MTHM)	Bundles: 74,033 (1,475 MTHM)	44 years	In operation since 1971
Pickering A secondary pool	Ontario-Hydro Pickering	CANDU 28 elements	Bundles: 204,288 (4,085 MTHM)	Bundles: 193,500 (3,850 MTHM)	37 years	In operation since 1978
Pickering B primary pool	Ontario-Hydro Pickering	CANDU 28 elements	Bundles: 146,304 (2,926 MTHM)	Bundles: 140,443 (2,795 MTHM)	43 years	In operation since 1982
Bruce A primary pool	Ontario-Hydro Bruce	CANDU 37 elements	Bundles: 31,680 (602 MTHM)	Bundles: 23,544 (450 MTHM)	50 years	In operation since 1977
Bruce A secondary pool	Ontario-Hydro Bruce	CANDU 37 elements	Bundles: 340,992 (6,479 MTHM)	Bundles: 310,344 (5,930 MTHM)	50 years	In operation since 1978
Bruce B primary pool	Ontario-Hydro Bruce	CANDU 37 elements	Bundles: 42,432 (806 MTHM)	Bundles: 21,200 (405 MTHM)	40 years	In operation since 1985
Bruce B secondary pool	Ontario-Hydro Bruce	CANDU 37 elements	Bundles: 333,888 (6,344 MTHM)	Bundles: 228,750 (4,370 MTHM)	40 years	In operation since 1987
Darlington primary pool	Ontario-Hydro Bowmanville	CANDU 37 elements	Bundles: 350,000 (6,650 MTHM)	Bundles: ~ 80,000 (1,520 MTHM)	40 years	In operation since 1990

TABLE 1-B CANADIAN DRY SPENT FUEL STORAGE FACILITIES  
(Status of wet storage facilities as of August 1997)

<i>Name of Facility</i>	<i>Location and Owner</i>	<i>FUEL TYPE (Natural U unless specified)</i>	<i>Design Capacity (tHM)</i>	<i>Built Storage Capacity (tHM)</i>	<i>Current Inventory (tHM)</i>	<i>Planned Operating Life</i>	<i>Operating Period</i>
Whiteshell	Pinawa, Manitoba / AECL	CANDU WR-1 (enriched and natural 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 (enriched and natural U)	75 (12 silos)	75 (12 silos)	75 (11 silos)	50 years	1989 to present
Point Lepreau	Point Lepreau, New-Brunswick / New-Brunswick Power	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)	456 (2 modules)	401 (1.75 modules)	50 years	1995 to present
Pickering - Phase 1	Pickering, Ontario / Ontario-Hydro	CANDU 28 elements	1,375 (180 DSC's)	~ 535 (~ 70 DSC's)	~ 381 (52 DSC's)	30 years	Constructed, Licensed, first loading Jan./96
Pickering - Phase 2	Pickering, Ontario / Ontario-Hydro	CANDU 28 elements	5,376 (682 DSC's)	0	0	30 years	Currently under design and licensing
Bruce	Bruce, Ontario-Hydro	CANDU 37 elements	9,120	0	0	30 years	Currently under design

Note: All facilities are of AFR (RS) type except for Chalk River which is an AFR (OS) facility

## 2.0 STATUS OF 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 AFR(RS) type dry storage facilities. Between 1974 and 1989, concrete canisters (silos) were used 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. The single unit CANDU reactors switched to dry storage technology rather than expanding their wet storage capacities. Between 1991 and 1995 non transportable silos were built. Since 1995, vault and cask type storage methods are also in use. The various dry storage methodologies are summarized in Figure 2.0.

### 2.1 The Gentilly 2 Experience

This 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 MASTOR 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 Hydro Québec's Gentilly 2 Generating Station 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 2.1) 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.

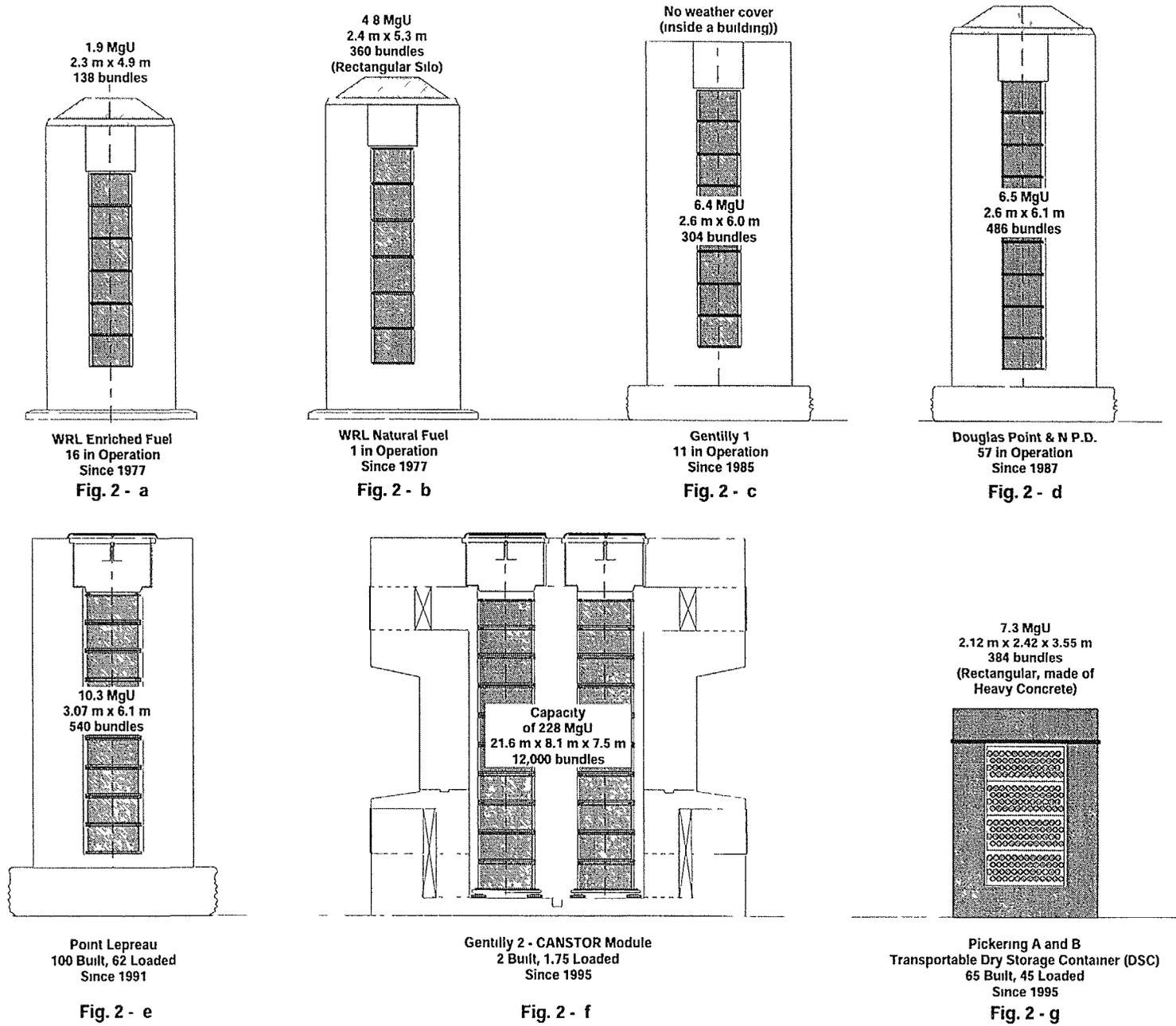
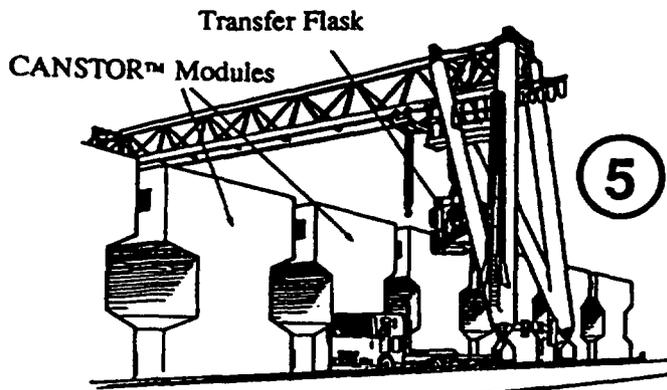
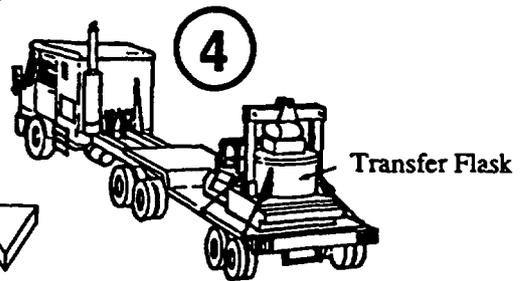
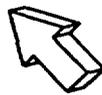


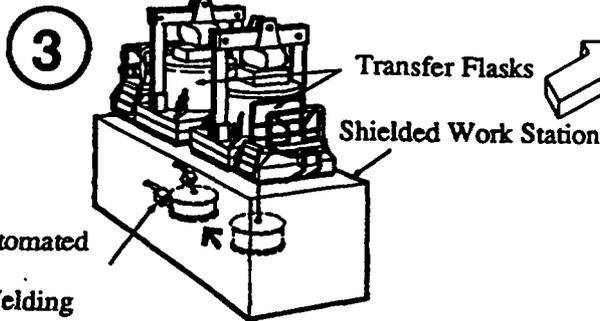
FIG. 2.0. Canadian concrete silos, module and cask  
 (Data as of July 1997).



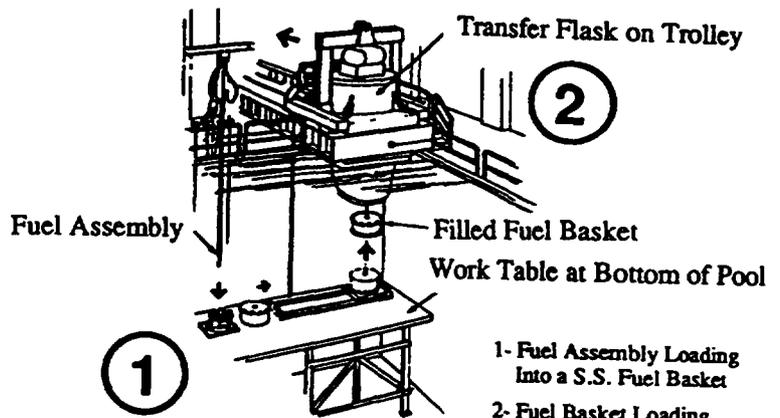
Storage of a Fuel Basket in a CANSTOR™ Module at the Storage Site



Fuel Basket Transfer To Storage Site



Fuel Drying and Basket Sealing in Shielded Work Station



- 1- Fuel Assembly Loading into a S.S. Fuel Basket
- 2- Fuel Basket Loading in Transfer Flask



FIG. 2.1. Fuel handling operations at Gentilly 2.

## 2.2 The Pickering Dry Storage Experience

The first phase of the Pickering Used Fuel Dry Storage Facility (UFDSF), located at the Pickering NGS (8x 550 MWe CANDU) was completed in 1995. The facility consists of a building which houses the operations area (workshops, services, utilities, offices etc.) and a storage area with capacity for 185 Dry Storage Containers (DSC's), each containing 10 t of CANDU spent fuel. Following the commissioning of all individual systems, operation of the Pickering Dry Storage Facility (UFDSF) started in November 1995. The first DSC was loaded underwater and back-filled with Helium following draining, sealing and drying operations. The DSC, also fitted with safeguard seals, was then placed in the Storage Area Building.

The Pickering DSC is a rectangular section container made of a double carbon- steel shell filled with reinforced high density concrete. The container design is shown in Figure 2.2. The payload cavity is designed to hold four standard fuel storage modules (96 fuel bundles each) and measures 1.34 m x 1.05 x 2.52 m in height. The loaded DSC weighs about 63 t. The UFDSF will provide additional on-site storage until the end of the station operating life (year 2005).

## 2.3 Bruce Used Fuel Dry Storage Project

By 1999, Ontario Hydro's Bruce Generating Station A will have filled all of its current wet pool storage capacity and by 2003, the Bruce Generating Station B will have filled all of its pool storage space. Ontario Hydro is planning to provide additional storage capacity utilizing a dry storage container (BDSC 600) similar to that used at its Pickering NGS. The changes made to the Pickering DSC include:

- fuel stored in trays instead of modules;
- container capacity increased to 600 fuel bundles instead of 384 bundles;
- changed the container lid to a metal lid;
- changed the single concrete lid to a double metal lid system;
- container is not off-site transportable;
- the container would be dry loaded.

## 3.0 WHITESHELL LABORATORIES PRIVATIZATION PROGRAMME

AECL entered into negotiations with a selected consortium of four companies lead by BNF in order to agree on the conditions for turning the site over to the private industry. The objective of the consortium, called Canadian Nuclear Projects Limited (CNPL), is to maintain as much of the current activities as possible including operation of the Underground Research Laboratories. AECL would retain certain CANDU reactor related services with its own staff and responsibility for its nuclear waste stored at site mainly in concrete silos. Completion of the negotiations is targeted for November 1997.

## 4.0 TRANSPORTATION

A relatively small number of spent fuel shipments has been made in Canada since interim, at reactor, storage has been implemented. In general, small capacity road casks have been used such as the Pegase cask (originally licensed in France) and the NPD-25 cask.

In addition to a steel cask, Ontario Hydro has also licensed for transportation its Pickering Dry Storage Container (DSC). With impact limiters both containers can transport fuel from a reactor site to a disposal site via rail, barge or road.

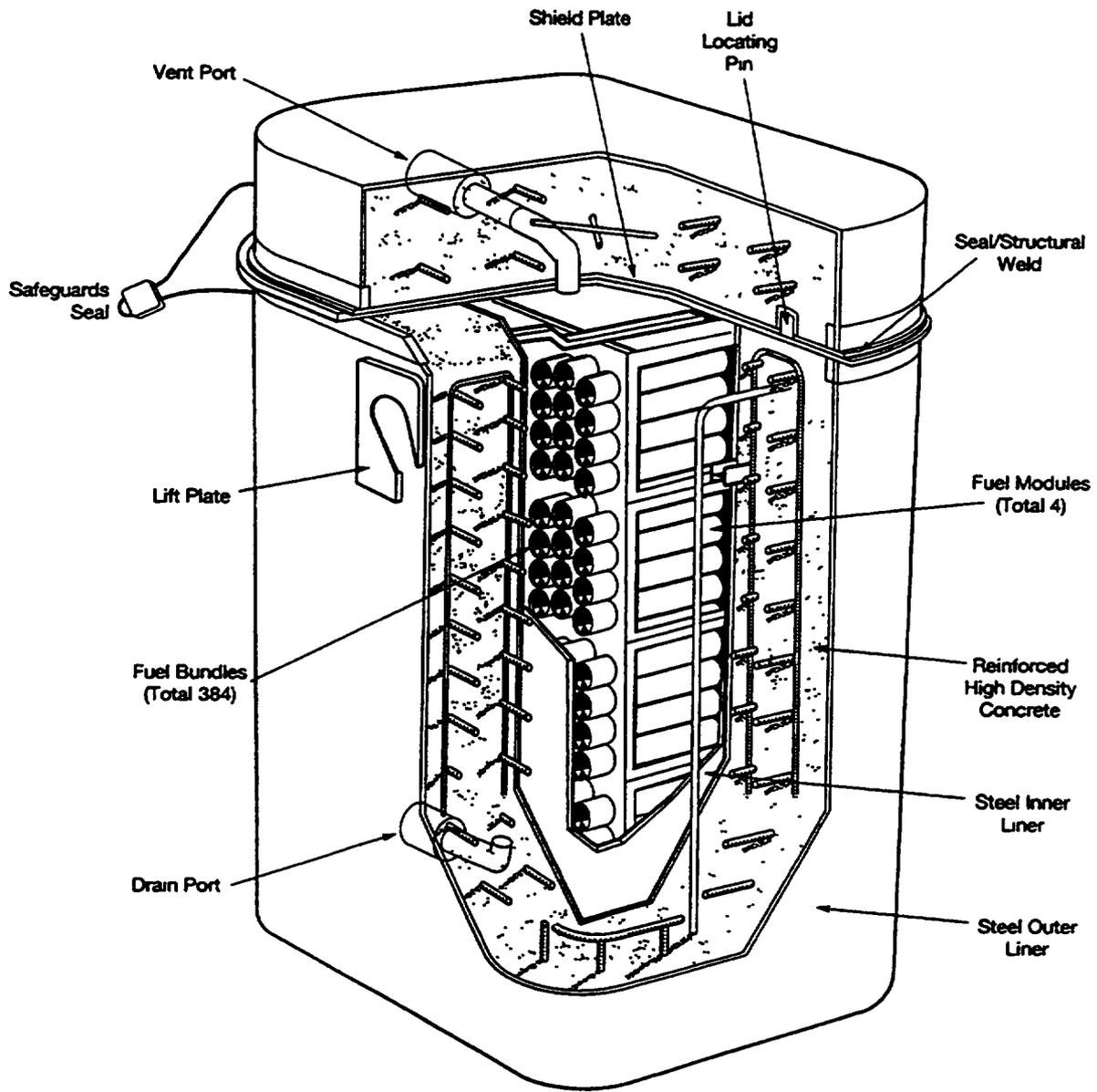


FIG 2 2 Pickering storage container (DSC)

Because of uncertainties surrounding the implementation of a spent fuel disposal facility, and for economical reasons, the DSC presently is not being considered for other Ontario Hydro reactor sites.

## 5.0 SPENT FUEL DISPOSAL

The environmental assessment and review process for the disposal concept developed under AECL's nuclear fuel waste management programme is nearing completion. The following events took place:

- Environmental Impact Statement (EIS) on the concept for Disposal of Canada's nuclear fuel waste was submitted by AECL to the government of Canada in 1994;
- The Minister of the Environment set up an Environment Assessment Panel for the purpose of establishing EIS guidelines, reviewing the disposal concept and holding public hearings on the acceptability of the proposed concept. These public hearings were held in 1996 and 1997;
- The Environment Assessment Panel is expected to submit its findings and recommendations to the government of Canada before the end of 1997.

The disposal concept is a method for geological disposal of nuclear fuel waste. Multiple barriers would protect the environment from both radioactive and chemically toxic contaminants in the waste. The barriers would be the container; the waste form; the buffer; the backfill, and other vault seals; and the geosphere (the rock, any sediments overlying the rock below the water level). Institutional controls would not be required to maintain safety in the long term.

During the public hearings the scientific community generally accepted the disposal concept although the research methodology and technical groundwork were greatly discussed. The general public representation emerged mainly from the antinuclear organizations. They voiced their opposition to any kind of disposal, preferring interim spent fuel storage with the ultimate aim of phasing out the nuclear power generation. Petitions were also made to couple the disposal review with acceptance of nuclear power by the public.

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# SPENT FUEL MANAGEMENT IN CHINA: CURRENT STATUS AND PROSPECTS\*

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## Abstract

In this paper, the development of nuclear power in China, its status of operating nuclear power plants and progress of on-going NPP projects are described. With the arising of spent fuel from NPPs, a national policy of a closed nuclear fuel cycle has been determined. Following storage at reactor sites for at least 5 years (generally maximum 10 years), spent fuel will be transferred to an away-from-reactor pool type centralized storage facility. Adjacent to the storage facility, a multi-purpose reprocessing pilot plant will be set up by the end of this century. An industrial scale reprocessing plant would be succeeded around the year 2020.

China's spent fuel management activities include at-reactor storage, transportation, away-from-reactor storage and reprocessing. Relatively detailed description of the work done up to now on spent fuel management and plans for the future are described. It should be noted that activities related to the management of high level radioactive waste are not included here.

## 1. CHINA'S NUCLEAR POWER PROGRAMME

The development of nuclear power in mainland China was initiated in the 80's. The first two nuclear power plants, Qinshan phase I and Daya Bay, were completed successfully at the early 90's and have been operating quite well until now. The next four successive projects are: Qinshan phase II, Lingao, Qinshan III and Lianyungang. Construction will start before the year 2000. Other potential nuclear power projects are being considered for further programme development. Efforts have been made with regard to site selection.

### 1.1. Status of operating nuclear power plants

#### ● Qinshan nuclear power plant, Phase I

The Qinshan 300 MWe nuclear power plant (NPP), the first self-designed and self-constructed NPP in China, was connected to the grid on December 15, 1991. It reached full load in July 1992. In its first year of commercial operation (in 1994), an annual load factor of 68% was attained and increased further to 84% in 1995 and 1996. The plant finished the third refueling and maintenance programme, and now it is in its fourth cycle operation. Its performance as a prototype plant has been satisfactory. It is characterized by its good design, manufacturing and construction as well as qualified personnel and good management. The completion of this plant represents the end of a history without nuclear power in the mainland China and the beginning of a new epoch of nuclear power development in China. The experience gained in the design, research and development, construction, commissioning and operation of the Qinshan NPP can be widely benefited by experts engaged in nuclear power development.

#### ● Daya Bay nuclear power plant

The Guangdong Daya Bay NPP (2x900MWe PWR) is owned by a joint venture company the Guangdong Nuclear Power Investment Company and its Hongkong partner. Its main constructors are three European firms: Framatome, as the supplier of the nuclear island; Alsthom GEC, as the supplier of the turbine island, and EDF, as the consultant to the owner. Commercial operation of the first unit of the Daya bay NPP began on 1 February 1994 and that of the second unit on 6 May 1994. The operation records have been kept well up to now, and the average annual load factor of two units was 70.5% in 1995, and 70.1% in 1996.

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\* This paper was contributed after the Meeting.

## **1.2. Progress of ongoing NPP projects**

- **Qinshan NPP, phase II**

The second phase of the Qinshan NPP (2x600 MWe PWR) has been started. The preliminary design was approved in November 1992 by China National Nuclear Corporation (CNNC). The detailed design and the excavation work started in 1993. The first pouring of concrete for Unit 1 was carried out in June 1996. According to the schedule Unit 1, will be connected to the grid in 2002.

- **Lingao NPP, Daya Bay Phase II**

Guangdong Lingao NPP (2x900 MWe PWR ) will take the Daya Bay plant as the reference plant, with the same capacity, same vendors and same suppliers. Its site is about one kilometer from Daya Bay. The project proposal was approved in April 1995, and the contracts of component supply, engineering consultation, and export credit were signed between France and China on October 25 1995. The civil construction has been started in May 1997 and according to the schedule the first unit will be finished in year 2003.

- **Qinshan NPP, Phase III**

In November 1994, the CNNC and AECL signed minutes on a joint construction of 2x700 MWe CANDU-6 in China. With the support of AECL, Chinese side finished the feasibility study for the construction of CANDU NPP at Qinshan as Phase III. The project proposal was approved in October 1995. The main contract was signed on 26 November 1996. It is expected to have the first concrete pouring in 1998, and to get the plant connected to the grid in 2003.

- **Lianyungang NPP**

The Lianyungang NPP (2x1000MWe VVER) is a Sino-Russian co-operative project. Its site was finally decided in October 1996. The Joint Leading Group for this project is composed of Jiangsu Province and CNNC. The feasibility study report has been completed and the draft contract preparation has been finished. The contract negotiation between Russia and China has been carried on. The construction of this project is planned to be started by the end of 1998. The first unit is planned to be completed by 2004.

## **1.3. Other potential nuclear power projects**

The programme of additional nuclear power plants is being reviewed for other provinces and cities, especially those in the south-east coastal areas such as Shandong, Guangdong, Zhejiang, Liaoning, Fujian, etc. Those areas are quite well developed regarding their economy but short of coal and hydropower. At present, efforts are made with regard to site reconnaissance and selection and raising of funds.

According to the electricity demands for the developing national economy, it can be predicted that nuclear power will be developed in China on a fairly large scale after the year 2000. Experts estimate, by the year 2010 the nuclear power capacity will reach 20~23 GWe and by the year 2020 will be 40~50 GWe.

## **2. PRODUCTION AND ESTIMATED PRODUCTION OF SPENT FUEL**

The spent fuel arisings will continue to increase while China's nuclear programme is advancing (Table I). The figures corresponding to the year 2005 are considered to be firm, since the relevant nuclear power plants are already under construction or construction soon will be started. There are some uncertainties in the rest of the figures since they are based on the forecasted nuclear power plant development, the realization of which will depend on many factors: economic and social conditions

including the nuclear power programme development strategy. The spent fuel accumulation figures have not taken the reprocessing into account.

TABLE 1. PROSPECT OF ANNUAL ARISING AND ACCUMULATION OF SPENT FUEL IN CHINA'S UNTIL 2020

Year		1995	2000	2005	2010	2015	2020
Nuclear installed capacity	GWe	2.1	2.1	9.7	19.7	29.7	39.7
Annual spent fuel arisings	tHM	60	60	343	582	822	1,062
Cumulative arisings	tHM	107	406	1,378	3,810	7,442	12,272

### 3. REPROCESSING OPTION AS THE BACK END OF NUCLEAR FUEL CYCLE STRATEGY

Based on China's concrete conditions and in order to make the best use of nuclear resources and to dispose radioactive waste with a view to protecting the environment, China has opted for a reprocessing strategy for the backend of the nuclear fuel cycle. All spent fuel from nuclear power plants or from various research reactors will be reprocessed.

For this purpose, it is necessary to construct a Centralized Wet Storage Facility (CWSF) away from reactors to provide buffer storage of spent fuel for the reprocessing plants. The CWSF, at the first stage, will have a storage capacity of 550 tHM by the end of this century. Meanwhile, a nearby reprocessing pilot plant (RPP) with a throughput of 300 kgHM/d will be constructed and put into run around the same period.

A larger scale reprocessing plant with a throughput of 400 tHM/a will be built around 2020, while the CWSF will expand its storage capacity up to 1000 tHM. Therefore, all civilian plutonium will not be reprocessed and will remain in spent fuel stored in individual reactor pools until turning to the next century.

Regulations regarding the management of spent fuel have been drafted and submitted to the State Council in 1996 awaiting for final approval.

### 4. AT-REACTOR STORAGE OF SPENT FUEL

There is always an pool for interim storage of spent fuel and for unloading the whole irradiated core fuel in case of emergency present at any Chinese reactor site. Spent fuel discharged from reactors has to be stored at the reactor pool for at least 5 years in order to reduce its radioactivity significantly. In fact, this interim storage period would be likely extended to 10 years or more since the reactors' owner would like to put-off delivery of spent fuel to postpone payment or in case spent fuel could not be accepted by reprocessing contractor under unavailable conditions at that time. For example, there are two pools with a capacity of 15 years' fuel discharge in the Qinshan-I plant. However, in respect of most of present and planning NPPs in China, the maximum 10 years of at-reactor storage of spent fuel has been generally defined, even with a compact storage pattern.

### 5. AWAY-FROM-REACTOR RECEIPT AND STORAGE OF SPENT FUEL

A Centralized Wet Storage Facility (CWSF) project located in Lanzhou Nuclear Fuel Complex (LNFC) is being constructed with a capacity of 550 tHM for the storage of spent fuel, among which 500 t for PWR fuel and 50 t for the others, at the first stage. Next to the storage pools, there is a set of systems, including a receipt and monitoring hall for casks with a overhead crane of 130 t capacity, cooling and purification for pool water to be recycled and the auxiliary systems such as water make-up, power and ventilation etc.

It is anticipated, that the CWSF being constructed currently would be put into active operation in 1998, and be extended with an additional capacity of 500 tHM or more in the early next century. In

farther future, the facility's storage capacity could be again expanded double and interlinked with the industrial-scale reprocessing plant through a designed channel if necessary.

## **6. TRANSPORT OF SPENT FUEL**

China's NPPs (including existing and planning NPPs) are mostly situated on the southeast coastal area of China while the reprocessing establishment in the northwest, which is very far away, at least 3000 km of distance, from all NPPs. For this reason, an issue on spent fuel transport has to be dealt with. A feasibility study on transport of spent fuel from the Daya Bay plant was completed a few years ago. The study result has shown that owing to no rail access to the existing NPPs, a combined transport option by both sea and rail would be preferable, using the large loading fuel casks and making two round trips a year. Alternatively, a gate-to-gate transport option by road is desirable due to very limited business in the near future. However, it is necessary that a completed spent fuel transport system, including casks and its maintenance facility, a purpose-built marine terminal, ships, wagons etc., should be set up in the far future.

### **6.1. Transport by sea**

There is a dock for unloading huge and massive equipment at the Daya Bay NPP bordered on the sea but no fixed crane on the dock. It is considered that installing a costly gantry crane with a heavy duty capacity on the dock is not costeffective since handling spent fuel casks is not frequent. Consequently, cask handling has to be carried out by crane attached to the transport ship or a rental floating crane.

In view of the fact that a specific purpose-made ship with dual-hulls and -bottoms is more reasonable and safer than renting a general ship, a conceptual plan of the ship fitted with a fixed crane, carrying a maximum of 5 cask packages, with a 3000 displacement tonnage has been put forward.

There are two ports, perhaps Shanhaiguan or Lanshan in the near middle of east coast of China, could be used for transit. A specially purpose-built marine terminal at the port which has a wharf for ship docking and be equipped with a rail-mounted cantilever crane of 150 ton capacity, enabling cask to be loaded from the ship onto the special rail wagons. A rail spur links the terminal with the main rail line.

The sea route of about 3000 nautical miles away from the Daya Bay plant has been identified. The navigation conditions have been carefully investigated, including proper sailing season, haven choice and prevention of striking on the rocks, as well as the emergency response and salvage access in the event of a ship sinking.

### **6.2. Transport by rail**

Type B (U ) of package for spent fuel, which must comply with all the regulations laid down by the IAEA, will be transferred to the concave-type wagons with a 150 ton loading capacity and twelve axles. China possesses the ability of fabricating the wagons but there would be a possibility of importing them from abroad alternately.

After being marshaled and short staying, a specific train composed of two cask-loaded wagons and some necessary auxiliary cars will be driven according to a designed transport scheme.

A running route between the marine terminal and LNFC has been preliminary selected on the principle of avoiding big-and middling-cities, densely populated regions and rail routes with heavy traffic. The transport distance is more than 2600 km while it would take about a week for single trip to travel.

### 6.3. Transport by road

Two shipments of spent fuel assemblies of Heavy Water Research Reactor (HWRR) from China Institute of Atomic Energy (CIAE), Beijing to LNFC for interim storage and following reprocessing were actually carried out by road with 6 small sized casks (RY-1A) (each accommodates maximum 12 HWRR fuel assemblies serving as dual-purpose application to both transportation and dry storage) in 1995, and 9 RY-1A casks in 1996 respectively. A total of 180 HWRR fuel assemblies was transported which contain 756 kgHM. The result has shown that this movement is quite efficient, there was no incident during the implementation process neither industrial or radiological.

## 7. TRANSPORT CASK

R&D on transport cask for spent fuel has been carried out for more than 10 years. A small-sized cask, RY-1A with a 5t loaded-weight has been developed in China. A medium-sized cask made of nodular cast iron with some 20t weight will finish its R&D stage within one or two years. The selection of large-sized cask loading more than 10 PWR fuel assemblies is being considered. Meanwhile, a purpose-built facility capable of handling cask up to 50 ton weight to carry out various monitoring and tests, e.g. covering shielding, containment, dropping, penetration and fire etc., has been set up.

As for pre-cooled spent fuel from NPPs, a type of large capacity cask, capable of holding more than 20 PWR fuel assemblies with a weight of about 120 tons would be preferably chosen in view of economy. There could be 3 options to realize it: (a) domestically R&D---cheap but spending longer time and less certain; (b) imported from abroad---mature and available sooner but expensive; (c) R&D in collaboration with foreign enterprise---assured, moderate in the cost and the term needed, and also favorable to transfer to domestic production. It seems that the last one should be the most desirable. However in case of no sufficient time to fulfill this task, purchase from overseas would be a more realistic approach.

## 8. SPENT FUEL REPROCESSING

A multipurpose reprocessing pilot plant (RPP) is under construction for the purpose of:

- demonstration of the processes, equipment and instrumentation under hot conditions;
- accumulation of design, construction and operating experience;
- training of the operation personnel;
- recovery of enriched uranium from spent fuel of the High Flux Engineering Test Reactor;
- R&D of reprocessing technology for MOX or FBR fuel in the future.

It is composed of a CWSF mentioned above, a main reprocessing facility with a maximum throughput of 300 kgHM/d, a hot cell laboratory with nearly 1 kgHEU/d and a machinery testing workshop as well as other auxiliary facilities. RPP basic design was accomplished at end of 1991.

Test rigs of the plant, completed in advance in 1992, were designed for simulated tests of some key equipment and instrumentation, and remote operation, such as a fuel bundle shear, a set of pulsed sieve extraction columns and their monitoring instruments, and a few specific glove-boxes for plutonium tail-end process etc. Other complementary R&D are being actively progressed in laboratories. Meanwhile, the Reprocessing Pilot Plant's detailed design is now underway. Construction of main reprocessing facility building will be started within two years. Commissioning of the whole plant is scheduled by early next century.

After extensive experience of the pilot plant obtained and the sufficient amount of spent fuel accumulated at CWSF, a large-scale, perhaps 400 tHM/a, commercial reprocessing plant will be expected in commissioning around 2020.

## 9. CIVIL PLUTONIUM RECYCLING THE FAST REACTOR (FBR) PROGRAMME

China's FBR development plan has been included in the National Development Programme for High Science and Technology. FBR development is divided into three phases :

- First phase: An experimental FBR with a thermal capacity of 65 MWth (25MWe) will be constructed by the beginning of next century.
- Second phase: Construction of a modular FBR is planned around 2010.
- Third phase: A large scale FBR with a capacity of 1,000-1,500 MWe is expected around 2025.

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This paper describes spent fuel management in the Czech Republic (CR). It describes both the recent status and the possible future development. The paper also briefly discusses the State Office for Nuclear Safety (SONS) last steps in the licensing procedures of the Interim Spent Fuel Storage Facility (ISFSF) at the NPP Dukovany. The dry ISFSF has been commissioned in 1997. The approach to construct an additional storage capacity is briefly mentioned. The recent situation in the introduction of new legislation in the Czech Republic is discussed as well.

**1. INTRODUCTION**

Nuclear power remains a significant source of the CR electricity production. The four VVER 440 units at NPP Dukovany contribute to the electricity production in Czech Republic by about 25%. NPP Temelín will increase the ratio of nuclear power on total electricity production to 45%, but completion of NPP Temelín is still a political issue.

A new legislation has been adopted during 1997. First of all, the "Atomic Act" - law No. 18/1997, issued in January 1997, entered into force in July 1997. The Atomic Act is based on the internationally adopted principles of nuclear safety and radiation protection, which are implemented in the recommendations of the IAEA, ICRP and WHO. The Atomic Act includes, in compliance with the Vienna Convention, provisions declaring 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 direct whether to reprocess or dispose the spent fuel. According to this law, spent fuel is not considered to be 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 to establish a Radioactive Wastes Repository Agency. This Agency will gradually overtake the activities connected with the waste management in the Czech Republic, including research and development in this area. The activities are financed mainly from the utility money, derived from the electricity production. The atomic Act will be followed by 13 Regulations, nowadays 9 of them have already been issued.

**2. SPENT FUEL MANAGEMENT****2.1. Spent fuel arisings**

Spent fuel from the NPP Dukovany represent currently the main source of spent nuclear fuel arisings in the Czech Republic. After its commissioning in 1998, the NPP Temelín will be another significant source. There are four PWR reactors in operation at Dukovany (VVER-440/213, the first unit started operation in 1985) and two PWR reactors are under the construction at Temelín (VVER-1000). The amount of spent fuel generated during a 30 years period of operation, is estimated to be 1,500 t for the NPP Dukovany and 1,350 t for the NPP Temelín. Thus, the total spent fuel arisings from the Czech NPPs will be about 2,850 t.

When the independent Czech and Slovak Republics were established on 1 January 1993, there were 1,176 spent fuel assemblies at the wet storage facility in the Slovak Republic. The transport of these spent fuel assemblies, back to the Czech Republic started in August 1995, so that recently there are only 132 spent fuel assemblies from the Dukovany NPP in the pools of Jaslovské Bohunice

interim storage facility. These remaining assemblies will be shipped back to the Czech Republic by the end of 1997. In September 1997, there were 2,588 spent fuel assemblies in the pools of the NPP Dukovany of which 1764 in 21 CASTOR - 440/84 casks in the ISFSF Dukovany.

## 2.2. Research and training reactors

After decommissioning of the research reactor at SKODA Plzen this year, only two research reactors at NRI Řež and a training reactor VR-1 at the Faculty of Nuclear Engineering in Prague remained. Two of them produce no spent nuclear fuel, because of their zero power output. Only the reactor LVR-15, operating at the Research Institute at Řež, produces spent fuel. This reactor uses the fuel type IRT-2M, mainly with an enrichment of 36%  $^{235}\text{U}$ . This spent fuel is partly stored at the reactor pool and later in wet pools: either in the old storage facility in the reactor building (status in September 1997: 119 EK-10 assemblies), or partly in one of the two new pools (status in September 1997: 16 EK-10 and 37 IRT-2M assemblies). The latter pools are placed in the building for both spent nuclear fuel and high radioactive waste, which was commissioned in December 1996. In September, there were also 71 drums with EK-10 stored in one of the dry cells of this facility. A more detailed description of research reactor fuel storage can be found in [1].

## 2.3. The fuel cycle back-end concept

VVER-440 fuel, after being discharged from reactors and a five years cooling period at the reactor pools of the NPPs, is stored in the interim storage facility ISFSF Dukovany. In December 1995, this facility commenced a trial operation and its permanent operation has been approved by the SONS this year. The capacity of the ISFSF Dukovany was originally limited to 600 metric ton of heavy metal by a political decision of former Czech government. Such storage capacity could cover spent fuel arisings from the operation of NPP Dukovany only until the year 2005. Consequently new storage capacity at other sites than Dukovany would be needed. The utility has devoted an intensive effort to solving this problem. About ten sites have been examined as a possible central interim storage facility site. Finally, the most promising site has been chosen, i.e. at Skalka, as a location for a near-surface dry cask storage facility (surprisingly without any significant public resistance to the project). CEZ has also prepared a conceptual study offering the following possibilities:

- two separate interim storage facilities at the site of each NPP, i.e. a new storage facility at Temelín and an enlargement of the existing storage facility at Dukovany;
- a central interim storage facility at the site of one of the NPPs;
- a central interim storage facility at the new site, both surface and underground options.

The cask technology has been chosen for any of these storage facilities by the utility. There are four vendors on the short list: GNB, NAC, Transnucléaire and ŠKODA Plzen. Expert's evaluations have preferred two separate interim storage facilities at the site of each NPP, or a central interim storage facility at the Skalka site(near-surface). A conceptual study has been submitted to the Czech government by the Ministry of Trade and Industry.

The government decided to abolish previous limits of spent fuel to be stored at the Dukovany site, which most probably will result in the concept of two separate interim storage facilities at the site of each NPP and the near-surface Skalka project as a back up possibility.

The commissioning of the first VVER-1000 unit is officially announced for the year 1999. The at reactor pool capacity will be sufficient from ten to twelve years of operation. After approximately ten years (not sooner than in 2010), CEZ will also have to store VVER 1000 spent fuel in the new (dry cask) interim storage at the NPP Temelín.

Reassuming the above-described situation, new storage capacity will be needed in the CR after the year 2005 for VVER-440 spent fuel and after 2010 for VVER-1000 spent fuel. Neither the possibility of returning the spent fuel back to the country of the fuel supplier is not considered by the

CEZ nor the reprocessing option is not foreseen. Consequently the concept of fuel cycle back-end will have to assure the storage and following disposal of spent fuel. Some research described in [2] has already been started and still continues.

### 3 .FINAL STEPS OF LICENSING ISFSF DUKOVANY

On 23 January 1997, SONS issued its decision No. 29/97: operation approval for a 10 year period has been given for the Interim Spent Fuel Storage Facility at Dukovany under fulfillment of several requirements (e.g. only storage of VVER 440 spent fuel in the licensed CASTOR-440/84 casks, full implementation of IAEA safeguards, etc. ). Before issuing this approval some major steps in commissioning had to be taken:

- EVALUATIONS of the final SAR of ISFSF Dukovany had to be carried out. It represented evaluation of 15 chapters of the SAR content (site characteristic, basic safety criteria, spent fuel data, QA, storage components and systems, equipment, operational procedures, design basis accidents, initiating events emergency plans, radiation protection, dose monitoring, waste management, safeguards, physical protection, concept of decommissioning, operational limits and conditions). Not only a team of SONS staff, but also external independent experts have been involved.
- EVALUATIONS of RESULTS of more than one year trial operation represented other significant resource for assessing the safety of this storage. During this period every month were the results of monitoring various parameters closely watched by the SONS. Namely the monitoring of temperatures both of containers and cooling air, gamma and neutron dose rates in the storage and its vicinity, pressure between lids of the containers were the most important objects of the inspections and evaluations. The monitoring has proved the reliability of the ISFSF components and systems, the monitored values were mostly lower than the preliminary calculated values. But for example independent periodical measurements of neutron and gamma fields spectra resulted in the SONS order to introduce personal neutron dosimetry in the Dukovany interim storage.
- INSPECTION activity of the SONS was foregoing the issuing of the operational license - during more than 12 month of the trial operation inspections were focused especially on the implementation of the operational limits and conditions and operational reliability of the monitoring systems. FINAL SONS inspection January 8-9, group of inspectors - focused on preparedness of the operator for ISFSF permanent operation - the compliance with the requirements of CSKAE Decree No. 6, namely:
  - Operational documentation and instructions,
  - Personnel qualification,
  - Emergency preparedness,
  - Testing of safety significant components,
  - Design changes,
  - Implementation of previous SONS decisions and protocols.

After taking all these steps and after receiving declaration of the operator concerning preparedness of the ISFSF components, personnel, documentation etc. SONS issued its license for ten years operation.

### 4. RESEARCH AND DEVELOPMENT

During last two years several projects focused on the dry storage challenges started, some of them - such as the DATA BASE of the spent fuel in CASTOR-440/84 casks has already been almost finished giving SONS the possibility to calculate the activity of isotopes, neutron and gamma sources in each cask at any time etc.

Two major (three years) projects sponsored by the Czech government on the VVER cladding (Zr1%Nb) behavior are being carried out:

- Under “normal” storage conditions;
- Under “accident” conditions in the storage (temperatures above 500°C, internal pressures above 4 MPa) in the SKODA DIAMO Zbraslav.

Some effort has also been spent on spent fuel computer codes qualification.

## 5. INTERNATIONAL CO-OPERATION

A significant assistance for countries operating VVER type reactors represents the IAEA research activity. Especially the following two IAEA projects:

- Mechanical properties of VVER cladding material during long-term dry storage, VNIPIET, St. Petersburg, Russian Federation;
- Benchmarking of American code COBRA-SFS (thermal-hydraulic of spent fuel storage and transportation systems), KFKI, Atomic Energy Research Institute, Budapest, Hungary.

CEZ and SONS have contributed to the latter - KFKI Budapest.- benchmarking of COBRA SFS project with collecting and releasing some data concerning ISFSI Dukovany.

IAEA TCM/Workshops, both the representatives of Czech regulators and operators participated in the project focused on storage technologies. The way of open exchange of information among experts from developed countries and those from recipient countries on important topics (choice of storage technology, quality assurance etc.) has proved to be very efficient.

In 1995, the Co-operation Forum of VVER Regulators established the Working Group - Licensing process of Dry Spent Fuel Storage Facilities. During the last three years, three meetings of this WG have been organized (Prague 1995, 1997, Budapest 1996). The members of the WG had significant opportunity to exchange their opinion concerning major problems connected with the licensing of SF Facilities.

The first meeting was more or less devoted to gain basic information about the spent fuel management problems in the respective countries, situation with national standards, regulations, requirements etc. concerning dry storage, and identification of common and specific problems.

At the following meeting, suggestions for possible co-operation in licensing, research activities, etc. could already be formulated and the preparation of some guidance documents started.

At the last (until now) meeting of the WG, a Table of Contents for the SAR of a dry storage facility based on the Appendix of IAEA Safety Series No. 118 "Safety Assessment for Spent Fuel Storage Facilities" was prepared. This "Table of Contents" in general reflects the current situation in the countries operating VVER type reactors but the group tried to modify it taking into account the some specific conditions. List of available and applicable documents, expert evaluations, results of experiments and research concerning the problems related to the dry storage of spent fuel in some countries participating in this working group has been prepared as well.

Bilateral co-operation in the area of spent fuel facility licensing has been realized between SONS and US NRC. In December 1996, brief consultation of some final licensing steps of dry cask storage facility were carried at the NRC. This consultation helped SONS experts to precise the operational limits and conditions of the Dukovany interim storage and – especially in the cask handling requirements (criteria for drying etc.).

## 6. CONCLUSIONS, FUTURE TASKS AND CHALLENGES

Since 1995, a fast development in the area of spent fuel and waste management in the Czech Republic can be observed. First of all, some experience with the evaluation of SARs for dry storage facilities has been gained by the SONS and a new dry – cask interim storage in Dukovany has been successfully constructed and commissioned. A significant and more general step represents the newly adopted nuclear legislation and establishment of the Radioactive Wastes Repository Agency. Some development has also been achieved in the research. In spite of the fact that the basic research for studying the behavior of spent fuel during long-term (dry) storage, including experiments in hot cells, is missing, some work has been initiated to overcome this problem. The announcement of SKODA Nuclear Machinery Plzen, concerning the development of its own dry storage technology (the dual-purpose cask SKODA-440/84), is an interesting development.

Undoubtedly, many challenges remain in the area of the spent fuel. The research focuses on the behavior of spent fuel during long-term (dry) storage, the behavior of some storage components and burn up credit problems. Owing to the planned utilization of the dual casks in the CR, some problems will have to be solved in the future. Especially the relationship between the required operational inspections (of the cask components) and the renewal (issuing) of the (off-site) transport license (permit) after a more than 10 or 20 years storage period, proving the influence of both the radioactive inventory and the environment to the transport and storage cask components. If the near surface cask storage is the option of the utility, the establishment of acceptance criteria will become another significant task for the regulatory body.

Implementation of the Joint Convention on the Safety of Spent Fuel Management and on the safety of Radioactive Waste Management in the following years will certainly influence the activities in this area, not only in the Czech Republic.

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# STATUS AND PROSPECTS FOR SPENT FUEL MANAGEMENT IN FRANCE

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XA9846850

## Abstract

The 70's oil crisis has shown that the energy resource dependence of France was too high. The decision was made by the French government to accelerate the implementation of an ambitious nuclear power programme, based on Light Water Reactors, and to do the utmost to reuse the energy bearing material included in the spent fuel. The French nuclear policy has not changed since then. This paper is aimed at describing the present status of implementation of this policy, and the associated prospects. It will first sum up the presentation made in 1995 to the Regular Advisory Group of IAEA on Spent Fuel Management. Then, it will update the situation of the main actors of the spent fuel management policy in France: EDF, the national utility; COGEMA, world leader on almost all the steps of the fuel cycle; CEA, the national research body in the field of nuclear science and its applications; ANDRA, national body in charge of the management of the waste arising from the nuclear activities in France, final disposal included.

## 1. INTRODUCTION

Since the end of the second world war, France has endeavored the development of nuclear activities in both military and civilian applications. This was specifically the purpose of the creation of the Commissariat à l'Énergie Atomique (CEA) in 1945.

When EdF, the national utility, began to use Gas Cooled Reactors to produce electricity, the question of the management of the spent fuel was raised immediately because the long time storage of these fuels was technically difficult then. The previous experience gained in operating the first industrial-scale reprocessing facility at Marcoule, since 1958, drove the French government to decide the construction of a new plant dedicated to the management of the EdF fuels at La Hague: UP2, started up in 1966. At this first stage of the UP2 activities, the purpose was only to handle properly and safely the spent fuel elements; recycling of the valuable material was not yet in the French policy, but many research works in CEA were preparing this next step.

The 70's oil crisis showed to the French government that the energy resource dependence of France was too high. Thus, the decision was made to accelerate the implementation of an ambitious nuclear programme, based on Light Water Reactors, and to get all the possible means to reuse the energy bearing material included in the spent fuel. This resulted in several technical choices : the first reactor resulting from this new policy started up in 1976 at Fessenheim; the pilot FBR Phénix started up in 1973; the first important generic type of EdF's 900 MW PWR (16 reactors) was designed for the used of recycled plutonium; and the first facility to handle uranium oxide fuel started up at La Hague under the responsibility of the newly created COGEMA company.

In many other countries, similar political decisions were made. Several countries asked COGEMA to reprocess their spent fuel, and licensed their reactors for the use of MOX fuel. The French nuclear policy has not changed since then. For example, the French government approved the decision made by EdF and COGEMA, in the mid-80's, to launch a large-scale plutonium recycling programme into MOX fuel. This programme included the licensing of 16 reactors, the increase of the La Hague reprocessing plant capacity and the erection of a large-scale MOX fuel production plant, MELOX. In simple words, the French policy can be summed up as :

- France takes care of their spent fuel by reprocessing them, recycling of the valuable materials, and safe and efficient conditioning of the final waste;
- Interim storage of spent fuel can occur as long as it results from technical constraints and planning requirements.

## 2. FRENCH ENERGY POLICY

The main points of the French nuclear energy policy presented in 1995 were:

- France masters all the fuel cycle steps, i.e. mining, conversion, enrichment, UO<sub>2</sub> fuel fabrication, reprocessing, MOX and reprocessed UO<sub>2</sub> fuel fabrication;
- France has a at-reactor storage capacity consistent with its nuclear programme (55 reactors, mainly PWRs of three types —900, 1,300 and 1,450 MWe— and two FBRs);
- France's away from reactor storage is done in the COGEMA's ponds at La Hague, in accordance with the EdF needs;
- Transportation of spent fuel is a routine activity (more than 8,000 t of fuel safely transported);
- Gradual increase of reprocessing/recycling activities based on the La Hague reprocessing plant, the MELOX/Cadarache/Dessel MOX fuel fabrication plants and the EdF MOX fueled reactors;

France prepares for the future: R&D works on advanced storage means, reprocessing process improvements and next generation MOX fuel.

Recent evolution confirm the global axis of French spent fuel management policy while emphasizing flexibility as the keyword for tomorrow's strategy. This is mainly reflected in the 1991 French law on "research on nuclear waste management". Dealing with the final disposal of the nuclear waste, this law asks the Government to propose to the French Parliament, before 2006, the best way among three options : the geological disposal in a specific repository designed after studies made in at least one deep laboratory; the separation of long-lived nuclides by enhanced reprocessing and their transmutation in facilities to be defined; the long-term on-surface storage.

The following paragraphs will describe the situation and prospects of EdF, COGEMA, CEA and ANDRA regarding the spent fuel management policy.

### 3. EDF

The implementation of the French policy in EdF strategy can be seen in the long-term plan of EdF on spent fuel management published by its top-level management at the end of 1996. EdF confirms that reprocessing and recycling is the very essence of its nuclear strategy. Today, annual spent fuel arisings from EdF's reactors are about 1,100 tHM. This value should reach 1,300 tHM around the year 2000. From this amount, EDF intends to reprocess about 1,000 tHM of spent UO<sub>2</sub> fuel, and recycle the corresponding separated plutonium.

More precisely, Bernard ESTEVE, head of EDF's Fuel Department, stressed the strategy implemented by the French utility [1]:

- "recycling of separated plutonium in 28 x 900 MW PWRs and in Superphénix;
- adaptation of the reprocessed quantity of UO<sub>2</sub> fuel to the recycling possibilities;
- maximum extraction of plutonium during reprocessing and provisional storage of the excess of used fuel prior to a future treatment;
- minimization of the final waste volume."

This strategy entitles EdF maximizing the use of existing facilities, both from an economic and an industrial point of view, while keeping a necessary flexibility with regard to market prices of uranium and enrichment services, or to legal requirements regarding the future of HLW programmes.

#### 3.1. Spent fuel interim storage

The EdF's strategy in spent fuel management is reprocessing for recycling of the valuable materials and for proper conditioning of the real waste. In this strategy, it is needed to let enough time

between unloading of the fuel elements from the core and reprocessing, allowing mainly for the decrease of the irradiated fuel elements thermal power. This duration includes the necessary cooling time before transportation away from reactor. Thus, EdF has to deal with interim storage at-reactor, as every other utility, and with extended interim storage at-reactor or away from reactor. In 1995, EdF was studying the construction of a centralized interim spent fuel storage facility away from reactors. In fact, this project has been abandoned in 1996 since La Hague site, along with EdF's own at-reactor storage capacity, appeared to be sufficient to handle spent fuel arisings till the year 2010, taking into account the recycling of about 1,000 tHM each year of cooled enough fuel assemblies.

Nevertheless, reracking is contemplated by EdF to gain some space in its reactor storage pools. Studies showed that gains should be really significant (for instance, about 6,700 tHM in the 900 MW PWRs). Such reracking would also be possible at La Hague.

### **3.2. MOX loading**

As explained sooner in this paper, many EdF's reactors were initially designed to accept 30% of MOX fuel in the core, and 16 are licensed to do it at present. The loading of these reactors is on progress, the rhythm being driven by the MOX fuel production rate. Two new reactors have been loaded in the first half of 1997 (Tricastin 4 and Gravelines 1), so that 12 reactors are presently loaded with MOX. Due to the high throughput of MELOX, 14 reactors should be loaded with MOX fuel at the end of 1997.

In line with their recycling strategy, EdF decided to increase the number of reactors licensed to receive MOX fuel: 28 x 900 MW PWRs is the short term target. EdF has already filed the licensing procedures for the 12 reactors needing to get this license (Chinon 1 to 4, Cruas 1 to 4, Blayais 3 and 4, Gravelines 5 and 6). In addition, the public inquiry, preceding licensing procedure for MOX loading, has been conducted early 1997 for the Chinon's four units.

Current French experience in MOX fuel industrial utilization represents more than 600 assemblies, i.e., about 300 tHM. According to the granted license, MOX ratio in the core is limited to 30% and reloading is performed on a 3-cycle basis.

The projected programme implies annual supplies of MOX fuel growing from 52 tHM in 1995 to 130 tHM in 2000 and afterwards. Prospects are numerous regarding MOX use in France. For instance, it deals with burnup increase beyond the current limits : the MOX operating license already allows a burnup of 39 GWd/tHM for a third core fuel reload cycle; soon, MOX fuel performance will be brought to the levels currently achieved with UO<sub>2</sub> fuels which have an average burnup of 45 GWd/tHM (maximum 47).

## **4. COGEMA**

### **4.1. Reprocessing**

1996 and 1997 confirmed the reliability of the La Hague reprocessing plant. Overall reprocessed quantities, as for oxide fuels, exceed 11,100 tHM. The site is now operating at the nominal throughput of 1,600 tHM per year.

With a licensed capacity of 14,400 tHM, the La Hague's ponds can store the fuel assemblies from many utilities. La Hague site is licensed to receive various kinds of fuel, such as high burnup UO<sub>2</sub> fuel, MOX fuel and research reactors' fuel. The service offered by La Hague regarding fuel management not only includes the interim storage; La Hague is able to reprocess all the fuel elements received in its ponds, giving an efficient way for conditioning the ultimate waste and letting the utility free to recycle the energetic material included in their fuel elements or not.

Spent fuel reprocessing at La Hague features numerous advanced techniques, designed to recover plutonium and uranium (that have a highly energetic content, even after using the fuel at a very high burnup), and to minimize the volume and the radiotoxicity of the final waste. The recovery ratio of plutonium and uranium is about 99,9%, that helps to reduce the waste volume without having trouble with the fissile contents. As practiced today at La Hague, reprocessing of one tHM of fuel yields less than 0.5 m<sup>3</sup> of High and Intermediate level waste and 1.4 m<sup>3</sup> of Low and Very low level waste. These very low volumes result from technical choices taken by COGEMA in the past few years, including the discontinuation of cementation of hull and end fittings, the discontinuation of bituminization of the liquid waste and the high-force compaction of technological waste and of fuel metallic structures. This policy permitted COGEMA to develop a new strategy in solid waste conditioning: the standardization of the waste package for handling and storage interfaces, called Universal Canister. This allows for reduction of cost of transportation, interim storage and final disposal of High and Intermediate level waste. Although external dimensions remain the same on every Universal canister, the constituting materials and the internal fittings may vary according to the type of waste contained. The Universal Canister relies on the long-known glass canister which specifications have been approved by many countries, and which size is common for BNFL and COGEMA's customers.

Consequently, reprocessing appears to be the most efficient way for safe and cost saving spent fuel management : storage of the fuel is limited in time to the technical needs of reprocessing and can be handled by the reprocessing facility; after proper conditioning, the final waste has a volume more than four times lower than the final volume of the fuel conditioned for direct disposal, and is delivered to the customer under the form of small standardized packages rather than big bundles of several fuel elements.

#### **4.2. MOX fabrication**

French spent fuel management policy plans recycling of plutonium into MOX fuel. In this regard, one of the most significant achievements of the recent years is the construction and the commissioning of the MELOX plant, located on the Marcoule site, southeast of France, the first commercial high-throughput MOX fuel fabrication plant in the world.

In 1995, the MELOX plant came online. Since then, the production of the plant has increased as planned, and has reached the licensed throughput of 10 tHM of MOX fuel per month in the first half of 1997.

The fabrication process implemented in MELOX is derived from well-proven techniques qualified in the Cadarache plant and in the Dessel plant. This process is called Advanced-MIMAS. Rather straightforward, it can be summarized as follows : the PuO<sub>2</sub> powder is micronized with a part of UO<sub>2</sub> powder to form a primary blend of about 30% Pu content, which is then mechanically diluted and mixed with a free-flowing UO<sub>2</sub> powder to get the specified Pu content of the MOX fuel. The pellets are pressed, then sintered, then adjusted with a grinding machine and then introduced in the cladding tubes. The rods are then used to make the fuel assembly.

To take full advantage of the MELOX characteristics for the benefit of all potential customers, COGEMA is implementing adjunctions to the present plant to let it be able, from 1999 onwards, to produce every kind of LWR fuel.

#### **4.3. Transportation**

Transportation is a significant aspect of spent fuel management. As soon as the fuel elements are no longer stored at-reactor, transportation means take the premium importance in the safety of the spent fuel management. In some cases, the transportation means feature both characteristics of transport and interim storage (even final disposal conditioning, sometimes).

In France, transportation of EdF's spent fuel to La Hague is a routine activity, with no difficulty. The transportation services are fulfilled by Transnucléaire, a COGEMA's subsidiary. Transport by rail and by truck is used. Transportation is also a key activity for the recycling industry: uranium and plutonium from EdF's fuel reprocessing are daily conveyed to the plants where they are converted into fuel, which are, in turn, dispatched to the power plants. Transnucléaire is able to transport any kind of fuel or nuclear material, by road, rail, sea and air, in many countries. This is done with various types of casks and transportation means, for example:

- TN 12 for spent fuel, TN 17 for fresh MOX fuel, TN 28 for vitrified waste, FS 47 for PuO<sub>2</sub> powder;
- special secured trailer-trucks for PuO<sub>2</sub>, ships adapted to nuclear material, wagons for long distance heavy load transportation.

In 1996, Transnucléaire performed 223 spent UO<sub>2</sub> fuel and 10 spent MOX fuel transports, and two overseas transports of vitrified waste to Japan and Germany.

## 5. CEA

CEA skills are multiple and cover almost all the technical areas related to the nuclear fuel cycle. CEA is also involved in the definition of spent fuel management practices. As the French nuclear Research and Development body, CEA brings technical expertise in two domains defined by the 1991 law :

- separation and transmutation of long-lived radionuclides;
- conditioning and final disposal of nuclear waste.

The first set of research projects is called SPIN. Under this project, for example, special fuel elements made of Neptunium have been fabricated by COGEMA for experiments to conduct in Superphénix, and enhanced reprocessing is studied by CEA teams with the support of COGEMA. Similarly, CEA deals with new types of reactors for recycling plutonium more efficiently, such as CAPRA, an acronym for " increased consumption of plutonium in fast neutron reactors ".

## 6. ANDRA

ANDRA, the national radwaste agency, is in charge of the last part of the French spent fuel management, which means operating Low Level Waste repository sites and finding, under the terms of the 1991 law, an appropriate disposal site for High and Intermediate Level Waste. After years of research, ANDRA selected three sites that could host an underground laboratory. One site is granite (in the Center of France), the two others are clay (northeast and southeast).

Public inquiries were conducted at each site. After validation by the Safety Authority, and formal acceptance by the political authorities, one or two sites will be chosen in 1998. Construction of the laboratories and implementation of the R&D programme by ANDRA will be conducted between 1998 and 2006 at the latest. At the end, the National Evaluation Committee will propose to the Parliament, through a " scientific statement " a solution for the management of ultimate waste in France. The decision then will be made by French policy makers as to the preferred option for the HLW final disposal.

## 7. CONCLUSION

As can be seen from the paper, the French strategy in spent fuel management has been focusing on recycling for long time. Thanks to this strategy, a strong industry has been developed, able to serve not only French needs, but also international needs even when the supporting policy is not consistent with the French one.

For instance, France has developed particular techniques in the dry storage area such as the TN 24 dual-purpose cask (transport and storage), already in use in the USA or the CASCAD (vault type storage facility) for non-reprocessed natural uranium fuels. Other applications, outside the French immediate needs, include the international programme for in-excess weapon-grade plutonium: COGEMA, associated with SIEMENS and MINATOM launched a programme aimed at building a weapon-grade plutonium dedicated MOX fabrication plant in Russia.

Rooted in the strategic situation of France and in the skills of the French nuclear research centers and companies, French spent fuel management policy will not undergo fundamental changes in the years to come. Spent fuel reprocessing-recycling is already a mature industry, mastering all the fuel cycle steps, and existing industrial tools are flexible enough to integrate new progresses, whether they are in the technical or in the legal domain.

Besides the acquired industrial maturity, future improvements are being prepared steadily. They will lead to improved performances regarding industrial flexibility, resources management, environmental aspects as well as economic results.

#### **REFERENCE**

- [1] Core Issues, n°2, The Uranium Institute (1997).



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**Abstract**

The spent fuel management strategy in the Federal Republic of Germany is based alternatively on interim storage and subsequent reprocessing of spent fuel or on extended storage and direct disposal of spent fuel. By economic and strategic reasons the spent fuel burnup is presently achieving 50 GWd/tHM and will targeting 55 GWd/tHM batch average. Recently the CASTOR V/19 license is issued to store spent fuel assemblies (SFAs) with up to 55 GWd/tU burnup (batch average) for 40 years. The integral pool storage capacity in Germany is 5600 tHM without the necessary full core reserve. The AFR spent fuel storage sites of Ahaus ( 4200 tHM) and Gorleben (3800 tHM) are in operation. The PKA pilot-facility to condition the SFAs is in the final state of erection and alternative approaches for SFAs with a higher burnup and/or MOX fuel are under investigation. The underground exploration of the Gorleben salt dome is in progress. Presently the non heat generating waste is disposed in the former Morsleben salt mine. Licensing of the larger Konrad iron mine for that purpose is under treatment.

**1. INTRODUCTION**

The safe management of waste and/or spent fuel („Entsorgung“) from nuclear power plants and particularly the orderly disposal of radioactive waste and/or spent fuel are of paramount importance to the peaceful use of nuclear energy. The German Federal Government continues to maintain its policy that the safe management of the back end of the nuclear fuel cycle is a precondition for the operation of nuclear power plants.

The basic principles for waste management are established in the Atomic Energy Act and in the waste management concept of the German Federal Government which gives greater substance to the statutory requirements and the principles of the waste management provisions („Entsorgungsvorsorge“) for nuclear power plants.

Recently, some modifications in the backend of the nuclear fuel cycle are created with the „Artikel-Gesetz“ (Energy policy amendments to the Atomic Energy Act). The „Artikel Gesetz“ allows (see Fig. 1):

- to use both strategies such as reprocessing or direct disposal of spent fuel;
- that the strategy selection is now with the utilities based upon criteria such as technical availability and economical advantages.

The new situation after the release of the „Artikel Gesetz“ is characterized by:

- MOX fuel assembly manufacturing outside Germany in Belgium or France;
- reprocessing services provided from France and UK;
- open decision between reprocessing and direct disposal.

**2. STRATEGIC DECISIONS CONCERNING THE NUCLEAR FUEL CYCLE AND ITS CONSEQUENCES FOR THE BACK-END OF THE NUCLEAR FUEL CYCLE**

In the late 1970s/early 1980s, important strategic decisions concerning the nuclear fuel cycle were taken against a background of high natural uranium prices. Other motivations included earlier plans for a fast breeder program as well as peak burnups attainable at that time for LWR fuel assemblies of slightly over 30 MWd/kgU. In view of these circumstances it is understandable that considerable importance was attached to recovery of the fissile materials still present in spent fuel

assemblies through reprocessing, and to the recycling of these materials in fast breeders and light water reactors by using them in the fabrication of MOX fuel assemblies (i.e. fuel assemblies containing mixed uranium/plutonium oxides) and RRU fuel assemblies (i.e. fuel assemblies containing re-enriched recovered uranium).

Today's situation, on the other hand, is characterized by relatively low natural uranium prices, large quantities of plutonium and uranium that have been recovered by reprocessing, and the absence of a fast breeder program. Under these conditions, the costs for the back end of the fuel cycle makes up the largest share in the total nuclear fuel cycle costs.

**Former Situation**( Atomic Energy Act)

**New Situation**( Artikel Gesetz / amendment to the Atomic Energy Act )

The basic principles are based on

- Atomic Energy Act
- Principles of Waste Management Provision

Spent Fuel has to be reprocessed and valuable parts of the fuel has to be recycled. Exemption are only allowed if

- this is technical not feasible
- this is economical not recommendable
- this is to risky

The new, additional "Artikel-Gesetz" allows:

- to use both strategies
  - reprocessing
  - direct disposal
- that the strategy selection is with the utilities based upon
  - technical availability
  - economical advantages

*Fig. 1. Changes in the German Back End of fuel Cycle Strategy.*

Fig. 2 illustrates the relative development of fuel cycle costs at current money values for a typical German nuclear power plant. The curves clearly show that, since the beginning of the 1980s, disposal has accounted for a steadily growing proportion of fuel cycle costs, while natural uranium, conversion and enrichment have merely been subject to currency and market fluctuations. The share made up by the cost of manufacturing uranium fuel assemblies has remained virtually constant over the period considered.

Nuclear fuel supply and disposal in Germany is characterized by a situation whereby key cost components are assessed in specific terms, i.e. per kg of processed fuel. 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 disposal costs.

The key to any reduction in the mass of fuel in circulation is the fuel assemblies. The most important part of optimizing fuel performance - among others - lies in the possibility for achieving higher burnups, which have an inversely proportional effect on the mass of spent fuel to be disposed of per generated kWh. Advances made in fuel assembly design and manufacture have now made discharge burnups of 55 MWd/kgU - or even more - achievable.

Fig. 3 shows the corresponding development in the average discharge burnups 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 burnups 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 kWh since the mid-1980s, in spite of disposal costs that continue to rise.

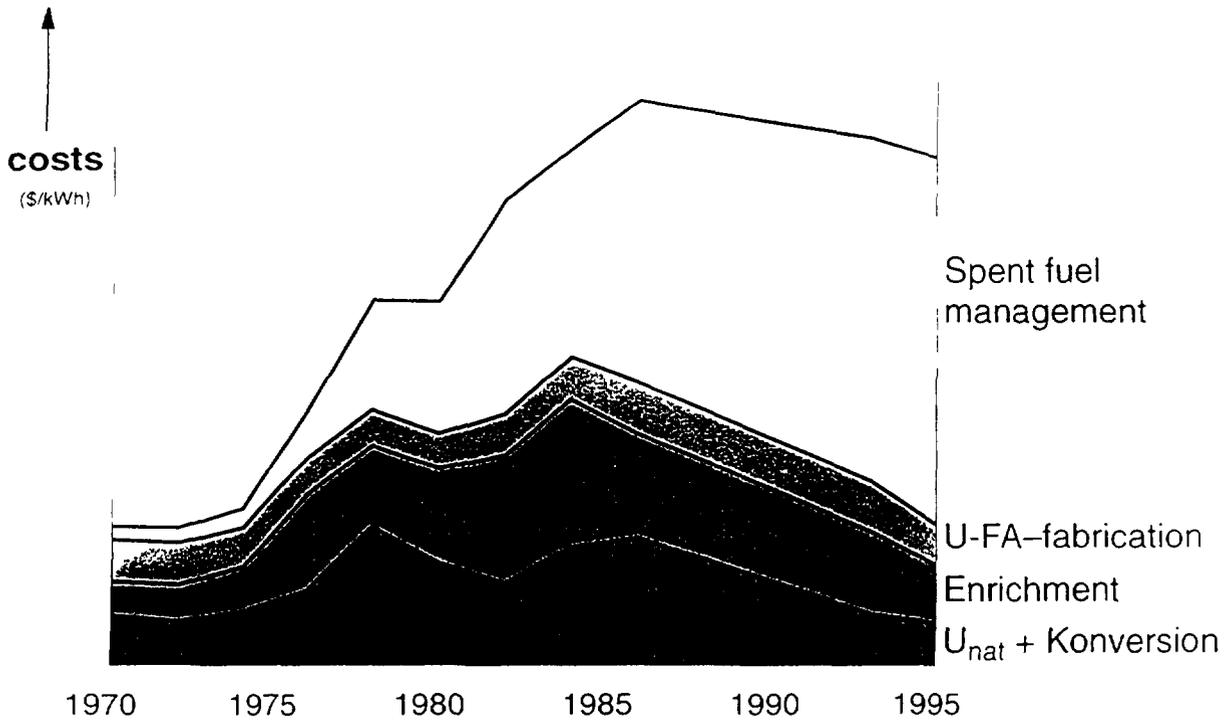


Fig. 2. Spent Fuel Management has become the dominating Domain of the Fuel Cycle Costs.

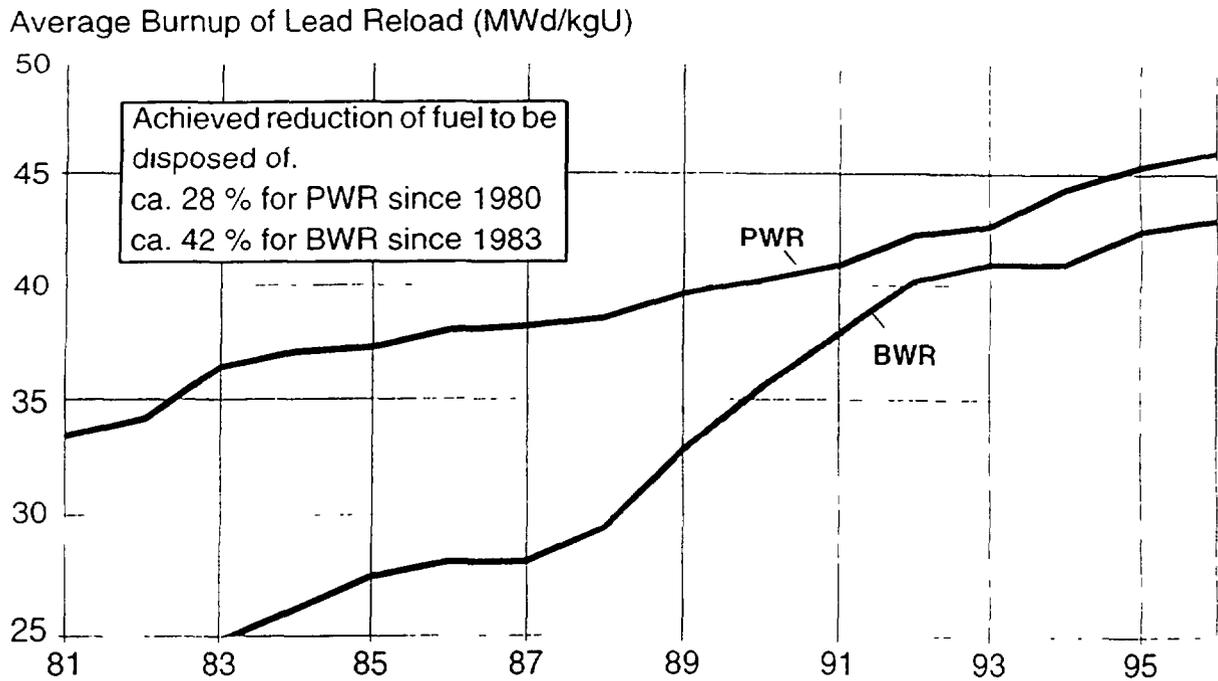


Fig. 3. Burn-up Increase reduces the circulating Fuel Quantity and thus reduces primarily the Spent Fuel Management Costs.

### 3. THE BACK-END OF FUEL CYCLE STRATEGY

Fig. 4 exhibits the scheme of the nuclear fuel cycle in Germany. The back end of fuel cycle management concept embraces 3 significant steps:

I. Interim storage of spent fuel in the nuclear power plants and in offside interim storage facilities.

II. Reprocessing of spent fuel and re-use of the fissile material in the stages:

- Recovering fissile material from spent fuel (recycling);
- Interim storage of the separated fission products, first as liquid and later on as vitrified material;
- Final disposal of the vitrified fission products; or,

Direct disposal of spent fuel in the stages:

- Extended spent fuel storage;
- Spent fuel storage conditioning for final disposal;
- Final disposal.

III. Disposal of radioactive wastes in the stages:

- Conditioning;
- Interim storage at nuclear installations, in offsite facilities or in regional collection centers;
- Final disposal.

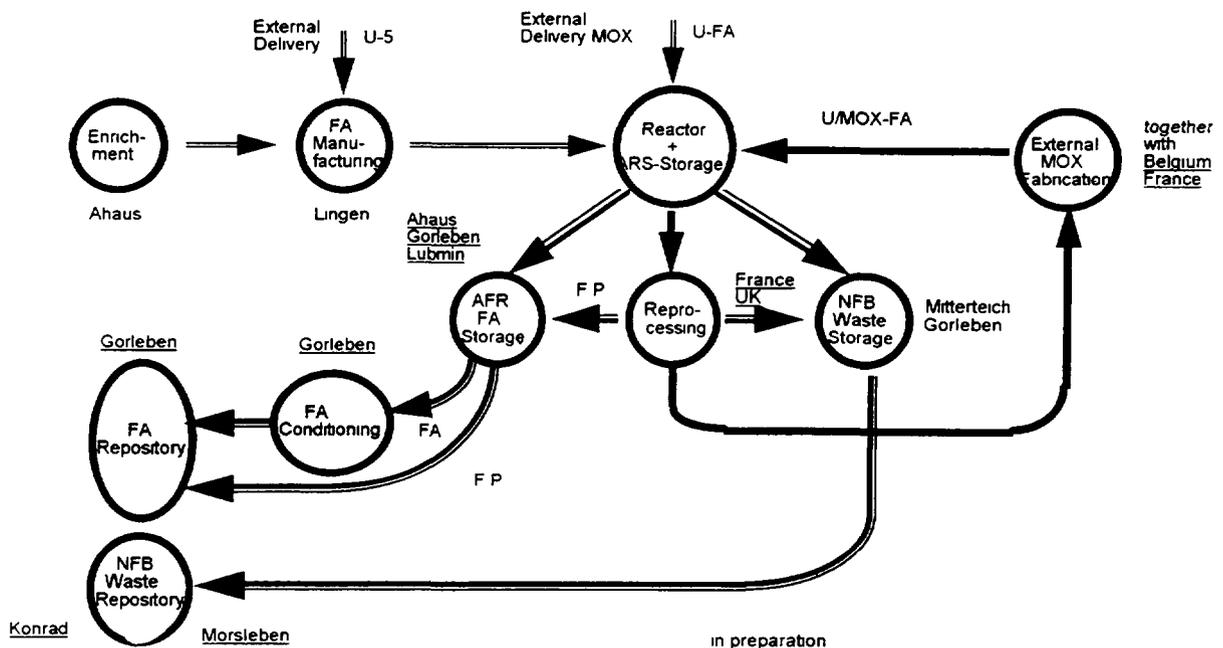


Fig. 4. Nuclear Fuel Cycle in Germany (overview).

### 4. FUEL PERFORMANCE OPTIMIZATION AND BACK-END OF THE FUEL CYCLE INTERFACES

#### 4.1. General aspects of U and MOX SFAs with increased burnup

For spent U, MOX and RRU fuel assemblies exhibiting higher burnup the feasibility of interim storage prior to direct disposal is of particular importance. Fig. 5 shows the decay heat - an important factor for this route - of a spent U- and of a spent MOX fuel assembly in W/kgHM and an average-

burnup of 60 GWd/tHM. In the medium and long term, the decay heat of a MOX fuel assembly is significantly higher than that of a uranium fuel assembly, and exhibits a flatter downward curve. This characteristic behavior of spent MOX fuel assemblies is highly significant for the direct disposal route, as poor choice of storage technologies may lead to very long decay times, both in the fuel pool and in away-from-reactor interim stores. Fig. 6 illustrates the consequences of a burnup increase and/or the use of MOX on the number of required rack positions in a reactor pool.

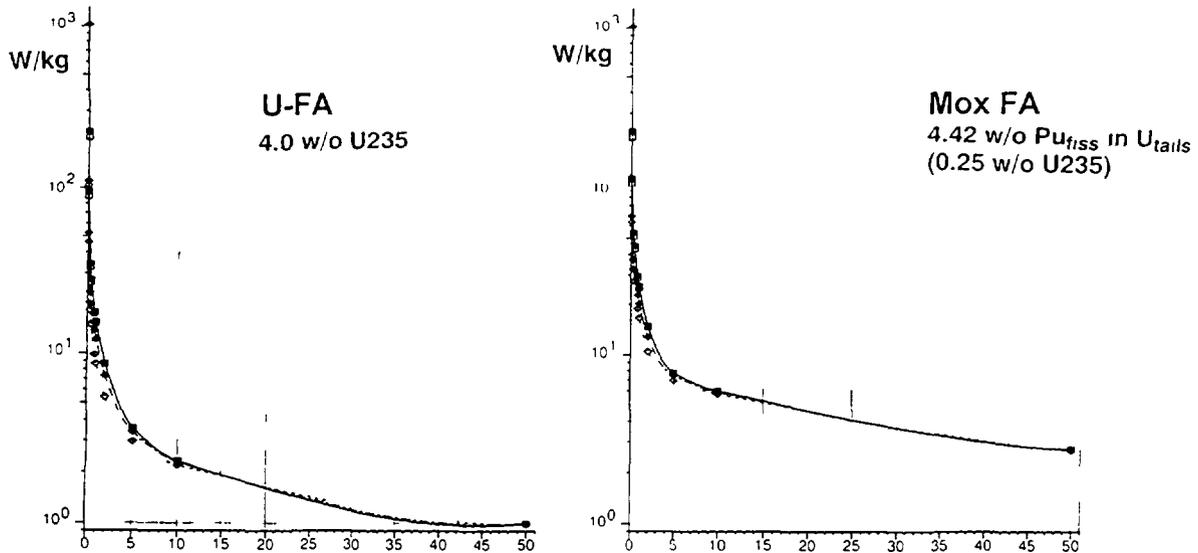


Fig. 5. Decay heat generated from a U and MOX FA after a burnup of 60 GWd/tHM.

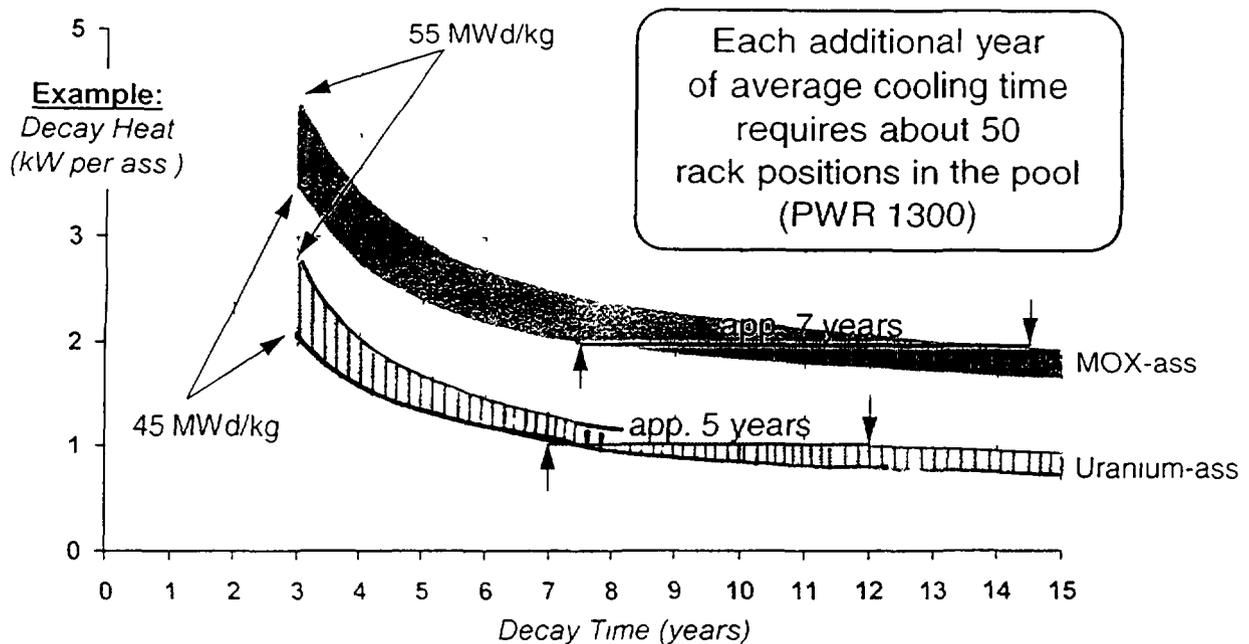


Fig 6 New Developments in the front end have significant consequences for the Spent Fuel Management

## 4.2. Wet storage of SFAs with increased burnup

In all cases interim storage ARS occurs in Germany so far in water filled storage ponds. Assessment of the ability of fuel assemblies to be safely stored has been based above of all on the evaluation of corrosion mechanisms:

- oxidative and electrochemical cladding tube corrosion: Pure oxidative corrosion is of no importance. Electrochemical corrosion can be suppressed through the selection of appropriate materials and by appropriate control of the pool water chemistry;
- corrosive attack of structural components and the so-called crevice corrosion: This kind of corrosion can be suppressed through the selection of appropriate materials. Furthermore, the fuel assemblies are for the most part passivated after irradiation in the reactor.

Especially, all defect mechanisms associated with stress and strain in the Zr-cladding can be neglected since all stresses - also in fuel with increased burnup - are far less than the yield strength. Therefore, wet spent fuel storage of fuel assemblies is of no concern when increasing the burnup.

## 4.3. Dry interim storage of SFAs with increased burnup

Higher burnup of fuel assemblies leads to higher decay heat levels and higher n- and  $\gamma$ -source terms. This correlation is not generally linear in nature, but is frequently characterized by a disproportionate increase with rising burnup levels. While the decay heat power levels in the interim storage cask affect, in particular, the temperature-dependent behavior of the fuel rod cladding tubes, the n- and  $\gamma$ -source terms dictate the requirements to be met by the shielding. 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  $\gamma$ -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. 7). 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. Fig. 7 compiles all noteworthy fuel rod degradation mechanisms under dry storage conditions:

- Oxidation of the Zircaloy 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 oxidizing substances.
- Iodine-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, iodine is not present in a form which could trigger stress corrosion cracking, and also the stress conditions required for the occurrence of stress corrosion cracking are absent.
- Creep is the enveloping criterion for consideration of cladding integrity during storage. At the temperatures of between 300 and 400°C which prevail at the start of dry storage, the cladding undergoes strain, the numerical value of which is largely determined by the fuel rod internal pressure and the temperature time history during 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.

Typically the higher dry storage temperature and the increased fission gas release - originating from the higher decay heat from fuel assemblies with higher burnup and/or MOX-fuel - 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 a longer residence time of a fuel assembly in the core tend to decrease the residual wall thickness through in-reactor corrosion, stress and strain is furthermore increased (Fig. 8). in order to stay within the licensed limits of stress and strain longer residence time of the fuel in wet storage is required or other measures - e.g. specially optimized storage cask loading patterns - are required.

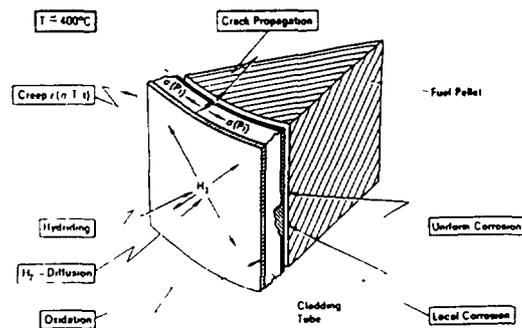
### 1. Source terms

- 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. 7. Source terms of spent FA and FA-Integrity are essential criteria for Dry Storage.

#### • Total Strain

Basis:

< 1 % to avoid creep rupture ==> 2%

Creep investigation post pile (unirradiated) creep

Total strain from irradiated cladding

#### • Max stress

Basis:

< 100 N/mm<sup>2</sup> to avoid SCC-rupture ==> 130 N/mm<sup>2</sup>

Worldwide experience that there is no SCC for stresses below 130 N/mm<sup>2</sup>



#### • CASTOR I & II

cladding  $\leq 390$  °C - 410 °C

burn up  $\leq 45$  GWd/tU

#### • CASTOR V

Cladding  $\leq 350$ °C - 370°C

Burn -up  $\leq 55$  GWd/tHM batch average

< 65 GWd/tHM single FA

Fig. 8. Spent LWR FA Dry Storage licensing criteria.

#### 4.4. Tentative estimate of dry interim storage periods of SFAs with increased burnup and/or mox fuel

The tentative estimate of interim spent fuel storage periods cannot give precise figures. It must be reminded that world-wide neither a specification nor other acceptance criteria for the final disposal of spent fuel are officially released. However a tentative estimate might show relative tendencies comparing different parameters of influence. The following parameters had been considered:

- Rock formation: salt or hard rock
- Maximum bed rock temperatures: salt = 200°C, hard rock = 100°C
- Kind of final SFA disposal package: multiple (3) or single SFA / package

The tentative estimate of spent fuel intermediate storage period needed before the final repository might be able to accept the spent fuel results in the following figures:

Kind of package	SFA	Rock formation	Minimum intermediate storage period
single SFA	U	salt	< 50 a
	MOX	salt	≤ 50 a
	U	hard rock	> 50 a
	MOX	hard rock	> 100 a
multiple SFA	U	salt	< 50 a
	MOX	salt	≤ 50 a
	U	hard rock	≥ 100 a
	MOX	hard rock	>>100 a

It might be discussed if the estimated figures of the necessary interim spent fuel storage periods might be right or wrong. However the result that the necessary interim storage time is positively influenced when the number of fuel assemblies per package is decreased and that salt as bed rock has the better thermal performance in comparison to hard rock cannot be overlooked. Care must therefore be taken to ensure that the economic advantages gained through optimization of fuel performance in the front end of the fuel cycle are retained at the interface between in-core fuel performance optimization and the back end of the fuel cycle when selecting specific strategies to close the nuclear fuel cycle. The best way to achieve this is to consider the life of a fuel assembly in an integral fashion that takes requirements at both the front and back ends of the fuel cycle into account, from design to manufacture, in-core operation and interim storage, right through to direct disposal or recycling of the fissile materials (plutonium and uranium) recovered by reprocessing.

## 5. BACK-END OF FUEL CYCLE INSTALLATIONS IN GERMANY

### 5.1. Spent fuel arising

In Germany back end of fuel cycle provisions must be made for 19 operating power reactors with an integral installed electrical power of 22,174 MWe. The yearly arising of spent fuel is decreasing from nearby 500 tHM to 400 tHM depending on the actual burnup achieved in the individual power reactors. Fig. 9 gives an overview of the kind of spent fuel in Germany. Originally all spent fuel was foreseen to be reprocessed. After the amendment to the Atomic Energy Law by the „Artikelgesetz“ the management of the spent nuclear fuel in Germany has changed. The present situation is shortly summarized in Fig. 10. It can be seen that both back end strategies - reprocessing together with recycling and final disposal - are presently practiced in Germany.

Reactor Type	FA-Type	Burn-up	Fuel	Comments
Gas/Graphit	Graphit-Sphere		U/Th	from AVR and THTR * total: 10 tSM
SWR	8 x 8 9 x 9 10 x 10 SVEA ATRIUM	40-50 max.	U MOX	burn up will increase towards 55 GWd/tM
DWR	WWER-2 WWER 440 14 x 14 15 x 15 AKA 15 x 15 FOCUS 16 x 16 FOCUS 16 x 16 HTP 18 x 18 FOCUS 18 x 18 HTP	< 40     45-50 max.	U     U MOX	ca. 500 tSM *     burn up will increase towards 55 GWd/tSM

\*: reactor operation closed, constant amount of spent fuel

*Fig. 9. Kind of Spent Fuel in Germany (Overview).*

- **Reconsideration of the Reprocessing Contracts:**
  - COGEMA Contracts
    - 1. contracts expiring in 2000 to reprocess 4755 tU: will be fully used
    - 2. contracts 2000-2010: will be used partly
  - BNFL Contracts
    - contracts expiring 2005 will be used partly
- **Direct Disposal**
  - 305 Castor THTR/AVR in intermediate store in Ahaus
  - KKK and KRB decided to use CASTOR-storage to comply for back end of fuel cycle requirements
  - an increasing number of reactors are sending spent FA in CASTOR CASKS to the AFR storage sites

*Fig. 10. Consequences of the "Artikel Gesetz" in the Back End of the Fuel Cycle in Germany.*

## 5.2. Wet spent fuel storage at ARS and transportation

The pools for storing spent fuel in the reactor buildings of nuclear power plants allow to store the down-loaded fuel until the fuel can be shipped for reprocessing or to ARS in dry transport and storage casks. Consequently, these storage pools were designed in Germany in the past so as to accommodate the volume of SFA unloaded over approximately 2 to 11 years of reactor operation, depending on the power plant in question. Integrally onsite storage capacity of approximately 5600 tHM for spent fuel is available in the pools of all power reactors in Germany. This capacity does not contain the full core reserve of about 500 tHM. Storage capacity equal to one core charge has to be

kept in reserve in every nuclear power plant. On principle, onsite storage capacity may not be used for fuel from other plants.

The most common spent fuel pool storage technology is the compact storage rack in Germany using borated SS as absorber and structural material. As shown in Fig. 6 the burnup increase of the spent fuel requires longer residence time in the wet storage pools and, as a consequence, more storage capacity in the ARS-pools of a given geometrical size. In countries as Spain and Korea Siemens equipped storage pools ARS with an innovative 2-region compact store. Recently also 2 German utilities ordered this kind of pool storage. The licensing is presently in progress. The 2-region compact storage racks provides in one region a design of the storage racks which can receive SFA without or only with limited burnup. This region is e.g. foreseen to receive the full core reserve. The other - much larger and more compact storage region - allows to store SFA exceeding a minimum burnup which depends from the initial U-235-enrichment. The 2-region compact storage permits to increase the storage capacity of a reactor from 11 to 15 reloads together with the full core reserve without changing the pools size.

The spent fuel shipping casks presently in use are designed for transporting the fuel a few months after it has been discharged from the core. Therefor their feature is a good heat dissipation capacity in the range of up to 100 kW and suitable neutron and gamma-shielding. Those cask however have a limited payload. The transport 5 tHM of spent fuel needs a cask of approximately 120 t cask weight. These kind of transport casks are used to ship fuel from reactors with a limited pool storage capacity to a reprocessing plant. SFA foreseen for the direct disposal are shipped in the transport and storage cask of CASTOR-type after a suitable time of decay of 5 to 10 years to the AFR storage sites in Ahaus or Gorleben.

Fig. 11 summarizes the transport and storage capabilities available in Germany. It might be surprising that routinely 80 to 100 spent fuel transport occur in Germany. Most of those transports had been bound for the reprocessing plants in France or UK. Also the transportation of 305 CASTOR/THTR/AVR casks occurred without any public attention. Only the few transports to Gorleben faced a major public resistance.

### Storage of Spent Fuel

Kind of Storage	Storage Mode	Location	Capacity	Licensed	Commissioned
wet storage	compact	ARS	6107 t	yes	yes
dry storage	CASTOR CASK	Lubmin	560 t	in progress	no
dry storage	CASTOR Cask	Ahaus	1500 t	yes	yes
		increase	4200 t	in progress	
dry storage	CASTOR Cask	Gorleben	3800 t	yes	yes

### Transportation of Spent Fuel

Cask	Destination	Mode	transports/a
various	Ahaus	rail	~ 80-100
	Gorleben	car	
	Cap La Hague		
	UK		

Fig. 11. Transportation and Storage of Spent Fuel in Germany.

### 5.3. Interim spent fuel storage at AFR

The AFR-storage in the sites of Ahaus and Gorleben is practiced in the transport and storage cask of CASTOR type. CASTOR casks type I and II are licensed for fuel assembly burnup of < 40 GWd/tHM. In order to meet with increasing burnup requirements and to achieve, for economic reasons, larger amounts of spent fuel in a single cask, the CASTOR V cask was developed which can accommodate 19 pressurized water reactor (PWR) fuel assemblies or 52 boiling water reactor (BWR) fuel assemblies. The CASTOR V cask series is designed for spent fuel assemblies with thermal and nuclear source terms typically for fuel assemblies with a rod burnup up to 65 GWd/tHM. Therefore, suitable combinations of fuel assemblies from a reload batch with an average burnup of 55 GWd/tHM can be loaded into the CASTOR V cask even if individual fuel assemblies from such reloads have higher fuel assembly burnup. However, fuel assemblies in the burnup range under consideration - e.g. MOX fuel assemblies - might even limit cask capacity due to the higher nuclear and thermal source terms in comparison to uranium fuel assemblies. The CASTOR V cask, like all other CASTOR cask types, is designed by applying the double barrier principle in order to safely and reliably retain all activity under normal and off-normal conditions. Licensing requires that no systematic fuel failure occur in the cask under the given storage conditions.

The AFR-spent fuel storage plant Ahaus consists of a large vault with the dimensions of length = 196m x width = 38m x height = 20m. The storage vault contains a cask reception area and a storage area. The decay heat is exhausted passively by natural convection through slots in the roof of the building. The total storage capacity for LWR-SFA in CASTOR-casks type I and II presently licensed is 1500 tHM. Additionally the storage of 305 CASTOR/THTR/AVR is licensed and those casks are in store. A further license application to use CASTOR V casks is in progress. The total storage capacity in Ahaus is - after receiving the CASTOR V license - 4200 tHM (Fig. 11).

The AFR spent storage plant Gorleben is of identical design and lay-out as that of Ahaus. The storage plant is licensed for CASTOR-Types I, II and V. Additionally there is a storage license for transport and storage casks containing vitrified fission products received from reprocessing. The licensed storage capacity of LWR-SFA is 3800 tHM. Since the Gorleben AFR-spent fuel storage plant is licensed to receive the CASTOR V cask LWR-SFA with a fuel rod burnup  $\leq 65$  GWd/tHM and MOX-SFA this storage site can presently receive such fuel assemblies for interim storage. Actually 8 casks are in store, thereof 3 filled with vitrified fission products received back from the French reprocessing plant and another 5 casks containing spent LWR-SFA.

### 5.4. Spent LWR conditioning for final disposal

Spent fuel has to be conditioned for final disposal. With respect to salt as bedrock 3 modes are under discussion to prepare the fuel :

- The disposal of spent fuel in POLLUX casks (reference case);
- the disposal of spent fuel cylindrical storage containers (GEISHA-study);
- early encapsulation of individual SFA (Siemens project).

The POLLUX-cask is designed to fulfill all requirements simultaneously for transport and storage as well as for final disposal in a gallery-type salt repository. This cask is able to receive the fuel rods after its removal from the skeleton e.g. from 10 U-PWR-SFA together with the compacted structural parts of those SFAs. Alternatively the POLLUX cask can be loaded with the fuel rods of 7 spent U-SFA and 3 spent MOX-SFA. Altogether the POLLUX payload is 5.5 tHM. The POLLUX cask has to a great deal the same technical features as the CASTOR cask since it has to fulfill the type B requirements by its design criteria. The license application for that cask is given to the competent authorities. However a final decision still needs the results from the exploration in mining of the *Gorleben salt dome*.

The storage containers considered in the GEISHA-study contains the fuel from 3 spent PWR-SFA in a densely packed form. After filling the container it is transported below surface in an revolving transport and shielding equipment. Down in the gallery of the final disposal it will be positioned in an overpack to protect the containerized fuel rod from the impact of the salt rock..

Both approaches - final disposal in the in POLLUX-cask and in the storage container considered in the GEISHA-study - needs a conditioning system operational at the end of the interim storage lasting 50 -100 years or more. Therefor the PKA-plant (Pilot Konditionierungs Anlage ) is erected at the Gorleben-site to demonstrate the conditioning technology in realistic operation. The basic approach is to remove from all SFAs the fuel rods, pack them densely in canisters which fits into the POLLUX cask or in storage container. In the reference case (use of POLLUX-cask) the structural parts of the emptied SFAs will be compacted and loaded in an space foreseen in the POLLUX-cask. The PKA is presently in the final stage of erection and will be ready for cold testing within one year. The license for hot operation is expected in 1998. the plant license allows a yearly throughput of 35 tHM and processing PWR-SFAs  $\leq 55$  GWd/tHM, BWR-SFAs  $\leq 45$  GWd/tHM.

Both approaches discussed above will provide packages containing fuel from 3 to 10 SFAs. As discussed under 4.4 those fuel packages will request for SFAs with a burnup above 50 GWd/tHM and for MOX-SFAs with the typically higher decay heat (Fig. 5) interim storage periods exceeding a 100 years period in the worst case. If, as discussed under political aspects, hard rock will be considered as bedrock of a final repository (see GEISHA-study) the situation may get even worse. That's why Siemens took the initiative to provide a technology to encapsulate individual spent SFAs within a project entitled "Early Encapsulation". This concept is shown in Fig. 12. After an appropriate decay time in the wet storage pond, the spent fuel assemblies are individually encapsulated (Fig. 13), directly in the fuel pool. The canister is internally dried, inerted and then seal-welded. Fig. 14 shows the individual steps in the encapsulation process. The design criteria selected assure a flexible use of the technology. The process can be combined with all transport and storage confinements available and licensed. The encapsulation process is based upon proven technologies from spent fuel pool services. The reliability of the encapsulation process is assured by designing each individual process step reversible, repeatable and repairable. The SFAs-capsule now takes on the barrier function for the fuel rod cladding tubes, while at the same time serving as a "waste package" for the spent fuel assembly. Immediately after sealing, the canisters are loaded into a dual-purpose cask, and removed from the pool inside the cask in the usual manner. *Early encapsulation* is being designed for all PWR and BWR fuel assemblies, in the latter case including the fuel assembly channels. As regards burnup, as well as the possibility of also encapsulating MOX and RRU fuel assemblies, margins are being allowed for which go far beyond current requirements. Even defective fuel can be encapsulated.

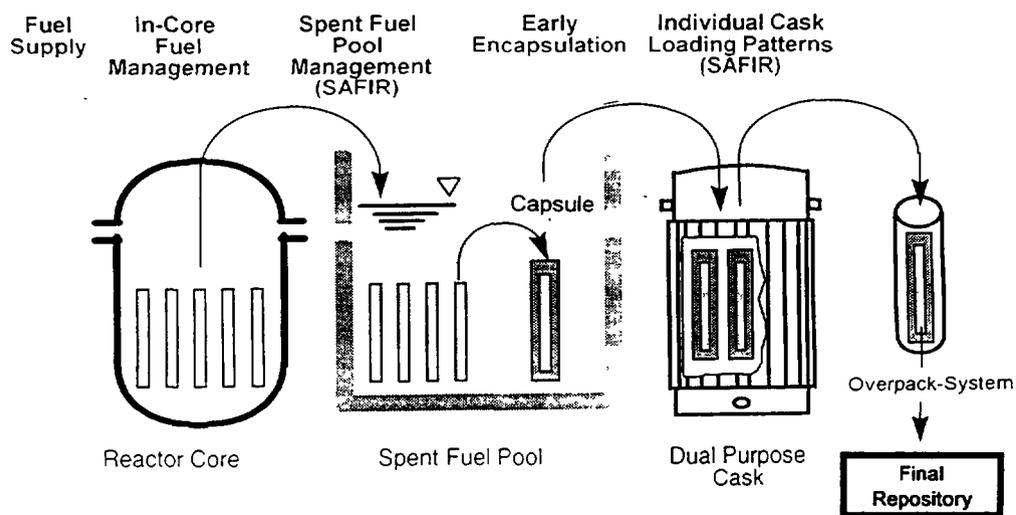
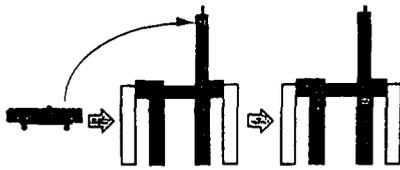
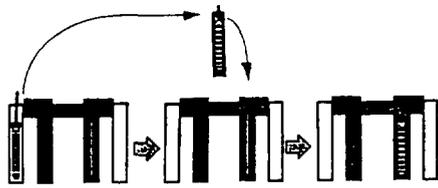


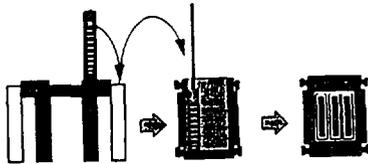
Fig. 12. Optimization of the entire process - not only of single steps - assures a reliable management of the Back End of the Nuclear Fuel Cycle.



The capsule is inserted into the encapsulation device



The FA is removed from the storage racks and is inserted to the encapsulation device, dried, placed into the capsule and seal welded



The encapsulated FA is removed from the encapsulation device and loaded to the transport and storage cask

*Fig. 13. Early Encapsulation of a Spent FA occurs in a movable service equipment within the pool of a Power Reactor.*

no longer extended ARS pool storage, extremely long interim storage AFR and complex interfacing between front end & back end of the fuel cycle even for FA with increased burn up



*Each spent FA is loaded into a capsule. The capsule interior is dried and made inert, and the capsule is seal welded. The encapsulation is performed just prior to dual purpose cask loading as a service*

- the capsule assumes the barrier function instead of the cladding  
 → *the FA EOL condition plays no longer a major part if the cask loading is performed*
- the capsule is designed to take all loads from handling processes and in storage  
 → *the capsule minimizes the number of interfaces between front end and back end and allows for separate optimization*
- the encapsulation provides an early conditioning of the spent FA for the repository  
 → *adequately designed SS capsules provides a better long term storage performance than a high burn up FA in its EOL condition*
- individually encapsulated spent FA provides the smallest decay heat per package  
 → *this is the only chance to limit the the interim storage for U-MOX-FA even with higher burn up to < 100a for all kind of geological rock formations discussef for the final repository*

*Fig. 14. Early encapsulation creates benefits throughout the entire Back End of the Nuclear Fuel Cycle.*

## 5.5. Final repositories

The salt dome in Gorleben was selected for the final disposal of heat generating wastes and SFAs which are not foreseen for reprocessing. After a positive statement assessing the above-ground salt dome investigations it was decided to start the underground exploration. Presently 2 shafts are completed reaching the -800 m level within the salt dome. The preparation of the galleries is in progress. The underground investigation of the salt dome is planned to be completed by 2003. By the results available and assessed today a positive final statement is expected at that time. Based on such a time schedule the licensing might be completed at 2011 at the earliest. Based on the necessary

interim above ground storage time periods of 50 years and longer - depending on the kind of SFAs to be disposed off - the general timing meets the requirements for an orderly management of heat generating wastes and the spent fuel selected for direct disposal in Germany.

Different from the decision of other countries Germany selected for the disposal also of non heat generating wastes deep geological repositories. Presently those wastes are disposed in the Morsleben salt dome repository. The caverns to receive the wastes are on a level of 400 m and the available volume accounts to 40,000 m<sup>3</sup>. The present license is exhausting in the year 2000. If the license will be extended is not yet decided. As a second possibility the former iron mine Konrad in a geological hard rock formation is considered for non heat generating waste disposal since 1976. This mine provides on a level of 800 to 1300 m a potential volume of 660,000 m<sup>3</sup>. The licensing procedure is in progress. From the present status of work the mine Konrad could be commissioned in the year 2002.

## 6. INTERNATIONAL CO-OPERATION

Through bilateral working agreements with other countries and through international organizations Germany co-operates in the development of methods and processes with respect to all steps of the management of the back end of the nuclear fuel cycle. Co-operation agreements covering activities from regular working visits and exchanges of experiences to joint research projects exists with a number of countries.

Organizations and experts from Germany are playing leading roles in the planning and execution of research and development programs within the European Community. An international exchange of information's and experiences takes place regularly within the Nuclear Energy Agency (NEA) of the OECD in Paris and the International Atomic Energy Agency (IAEA) in Vienna. Germany participates in several agreed research programs from the NEA as well as from the IAEA. Germany also supports the work done in the IAEA establishing rules and guidelines in the field of waste management by sending of expert delegates.

# SPENT FUEL MANAGEMENT IN HUNGARY: CURRENT STATUS AND PROSPECTS

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## Abstract

Paks Nuclear Power Plant is the only NPP of Hungary. It has 4 VVER-440 type units. Since 1989, approximately 40-50 % of the total yearly electricity generation of the country has been supplied by this plant. The fresh nuclear fuel is imported mostly from Russia and the spent fuel assemblies are shipped back to Russia for later reprocessing after 5 years of decay storage in the spent fuel pools. Spent fuel transports continue to take place, but to provide assurance of the continued operation, Paks NPP's management decided to implement an independent spent fuel storage facility (ISFSF) and choose the GEC-ALSTHOM's MVDS design. The construction of the ISFSF started in March 1995 and the facility was commissioned in 1997. The paper gives general information about the spent fuel arisings, the storage at the site, the shipments to Russia, the implementation process of the ISFSF and about situation with the spent fuel at the Hungarian research reactors.

## 1. INTRODUCTION

Nuclear electricity in Hungary is generated by the Paks Nuclear Power Plant. Four units of the NPP are now in operation, each with a VVER-440 type a Pressurized Water Reactor. The first unit was commissioned in 1982. The next units have been started with a time delay of about 1-2 years with the last unit, No. 4, commissioned in 1987. Since 1989, approximately 40-50% of the total yearly electricity generated in the country has been supplied by this plant. (40.8% in 1996) The average load factor at Paks for the year 1996 was 87.7%. Two units had load factors over 90%.

The fresh fuel is imported mostly from Russia (previously from the Soviet Union). There was a shipment of slightly used German fuel assemblies from the Greifswald NPP. On the basis of an international agreement, after 4 years of preparatory work, 235 assemblies were shipped from Germany to the plant. Although anti-nuclear groups tried to stop the shipment, it was accomplished successfully and 109 assemblies were loaded at Unit 4. A change in the fuel procurement policy was the conclusion of a contract for the development of an alternative VVER-440 fuel for Paks, together with the Finnish Utility IVO. The contract was awarded to BNFL (UK), the Lead Test Assemblies will be tested in Finland at Loviisa. If the test results are satisfactory, Paks will be in the position to choose from different suppliers.

According to the agreement between the 2 Governments (Russia and Hungary), the spent fuel from Paks can be shipped back to Russia after 5 years of decay storage in the spent fuel pools of the plant. The first shipment according to the agreement was carried out in 1989, and the last shipment in 1997. The development of the national strategy for the back-end of the nuclear fuel cycle for Hungary was motivated by the possibility to send the spent fuel back to the former Soviet Union for reprocessing so that no waste is returned to Hungary.

Although the national strategy is the same today, taking into account the uncertainties of the political situation in the CIS and other aspects, the NPP's management made a decision in 1990 to start studying the implementation of an independent spent fuel storage facility (ISFSF) at the Paks site. On the basis of the investigations made during 1991 and 1992 and of the evaluation of different proposals, the final decision of the Paks NPP was to choose the GEC-ALSTHOM's MVDS. The construction started in March 1995 and the facility was commissioned in 1997.

## 2. SPENT FUEL ARISING

There are 312 operating and 37 control assemblies in the reactor core. Fuel assemblies are of hexagonal cross section. One assembly encloses 126 fuel rods. The cladding of the fuel rods and the assembly shroud are made of zirconium alloy. The maximum enrichment of the assemblies is 3.6% U-

235. This provides for reactor operation with about one third of the core being reloaded every year. A reactor charge is about 42 t of UO<sub>2</sub>.

In average 120 spent fuel assemblies are produced in each reactor of Paks NPP annually that equal to 14.4 t of HM. Since last year the Paks management works on introducing a "more than 3 years" cycle for the assemblies. This will result in higher burnup values and a corresponding lower amount of the spent fuel discharges. In the 1995/96 fuel cycle year there were 72 assemblies in the core for the fourth year at two units. Evaluation of the results and further extension of the programme is expected in the near future

### 3. SPENT FUEL STORAGE IN THE AR STORAGE OF THE NPP

A three-year storage of spent fuel assemblies removed from the reactor was envisaged by the original design of NPP Paks, after which they would have been shipped back to the Soviet Union for reprocessing. Consequently, spent fuel racks were designed with a capacity for 349 bundles. This was achieved by a subcritical lattice structure without built-in absorbers. Since it became known that the spent fuel needs a longer cooling, for at-least 5 years, the storage pools were reconstructed to provide more space through installation of compact storage. In the new compact storage pool there are 706 cells, of which 650 pcs are normal storage cells and 56 pcs are hermetic casings at each of the four units.

The fuel bundles are placed into a 3050 mm long, hexagonal stainless steel absorber tube with a 3 mm wall thickness containing 1.05-1.25 % natural boron. The rack structure holding the absorber tubes is made of stainless steel. In case of an emergency, spent fuel can be unloaded from the reactor and placed in a second, so called "emergency rack". This rack which is erected on top of the normal rack, is normally kept in the Reactor Hall. Presently (August 1997) some spent fuel is being stored using these racks too, until it can be transported either to MVDS or away from Paks.

### 4. SPENT FUEL SHIPMENTS TO RUSSIA

So far, the spent fuel assemblies were shipped for reprocessing in Russia, and all the products (radwastes of different activity levels, plutonium, uranium) stayed ultimately in Russia. The quantity of the spent fuel assemblies to be sent to Russia and the price of the services have been negotiated annually.

The quantity of the spent fuel assemblies shipped back to the Soviet Union (Russia) until now is listed in Table I. The shipments were carried out using the standard Russian (Soviet) Railway Transport Unit, which includes containers of the TK-6 type. The total amount of spent nuclear fuel returned to Russia until August of 1997 is 2151 fuel assemblies, i.e., 258 tU.

TABLE I. SPENT FUEL ASSEMBLIES RETURNED TO RUSSIA

<b>Year</b>	<b>Assemblies</b>
1989	116
1990	235
1991	210
1992	240
1993	180
1995	480
1996	240
1997	450
<b>Total</b>	<b>2151</b>

Since the middle of 1992 the return of spent fuel is has an occasional character because of legal problems associated with the return of spent fuel to Russia. To provide a solution to this legal problem, a supplement was added in 1994 to the Intergovernmental Agreement of 1966. It should be noted however, that all transports are on an exceptional basis only, and it is always possible that they will stop if

Hungary does not take back the waste of reprocessing. It can be stated on the base of the above that the possibility for continuing with the former practice of return shipment is questionable.

## 5. IMPLEMENTATION OF THE AFR STORAGE AT PAKS

Taking into account the uncertainties of the political situation in the CIS and the economic conditions, the Paks NPP's management made a decision to study the implementation of an AFR storage facility at Paks. Seven foreign companies were invited to submit proposals, which was evaluated by all competent Hungarian organizations (operator, designer, regulatory body) and the IAEA. The final decision of the management of Paks NPP was to choose the GEC-ALSTHOM's (UK) Modular Vault Dry Storage (MVDS) system. The basic requirement to the MVDS system is the medium term (50 years) storage of VVER-440 reactor fuel assemblies and followers at a site, adjacent to the site of Paks NPP.

In this system the fuel assemblies are stored vertically in individual fuel storage tubes, the storage tubes are housed within concrete vault modules that provide biological shielding. To prevent the development of eventual corrosion processes, the fuel assemblies are in an inert nitrogen atmosphere 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 vaults and the outlet stack. (See Figure 1.)

According to the contract, the MVDS will provide 4950 spent fuel storage positions in the first stage. This first stage will be carried out in three phases. The first phase, with 3 modules, was completed at the end of 1996. The final phase of commissioning is scheduled for August 1997, with fuel loading in September. Construction of the whole facility will be implemented in 2 further phases, each of those will consist of 4 vaults with a storage capacity of 1800 assemblies. Thus, 11 vault modules will be constructed for storing 4950 fuel assemblies. The storage capacity can eventually be increased up to 14 850 storage positions by adding further vaults. This number would be sufficient to store all spent fuel generated throughout the lifetime of Paks NPP.

The licensing process took place in parallel with the construction works. The (trial) Operational License was issued by the Hungarian Atomic Energy Commission in February 1997. A decision was made by the management of Paks NPP to continue the construction with Phase 2, to have available storage space by the year 1999.

## 6. THE BUDAPEST RESEARCH REACTOR AND THE TRAINING REACTOR AT THE TECHNICAL UNIVERSITY

The first research reactor started operation in 1959. The type of the reactor was WWR-S, it was supplied by the Soviet Union. The reactor operated until 1967 at 2 - 2.5 MW<sub>th</sub> power level. After the first reconstruction, the reactor operated until 1986 with a new, 36% enriched fuel. The reactor reached its criticality after the second reconstruction in December 1992. The authorized thermal power is 10 MW<sub>th</sub>, while the available cooling capacity is 20 MW<sub>th</sub>. In the first period EK-10 (Soviet made) fuel was used. This is a rod type, rectangular assembly, with 10% enrichment. Since the first reconstruction WWR-SM fuel is used with 36% enrichment.

During the second reconstruction, all spent fuel (862 EK-10 and WWR-SM assemblies) were removed to an external spent fuel storage facility, located at the site of the Research Institute, 100 m from the reactor. The reconstructed inner storage has built-in absorbers, thus providing a compact storage array. Its capacity is enough to accommodate spent fuel from 4 - 5 years of operation and a total core unloading.

The training reactor at the Technical University is used mainly for teaching purposes. As its capacity is about 100 kW<sub>th</sub>, it is often not listed in the reactor statistics. This reactor uses EK-10 fuel, similar to what was used first by the Budapest Reactor. No spent fuel has been discharged since its commissioning in 1971.

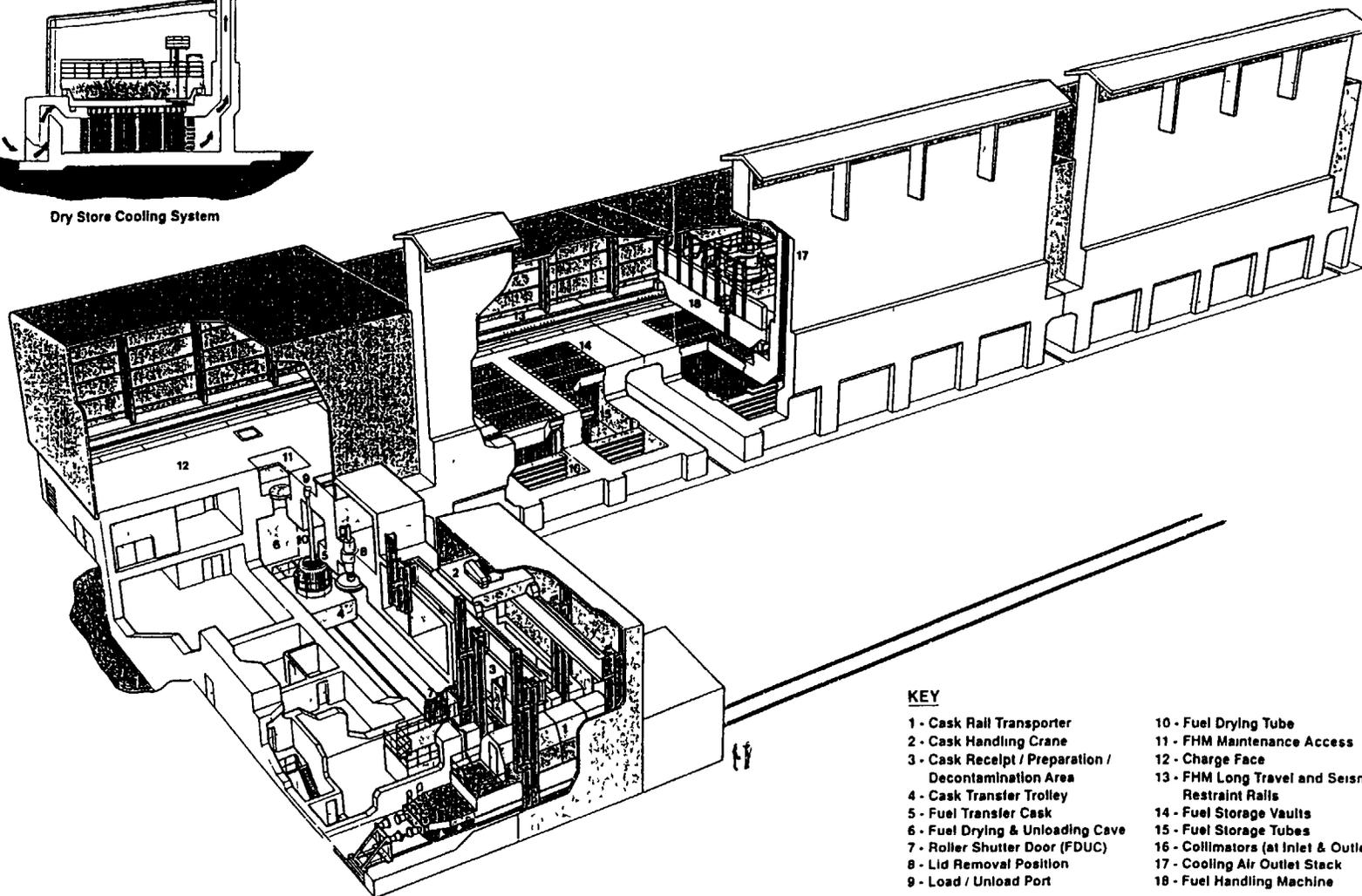
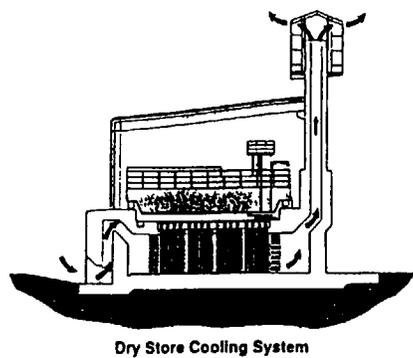


FIG. 1. Paks modular vault dry store.

A total of 1973 research reactor fuel assemblies are in Hungary, approximately 65% of it is irradiated. Unlike from the nuclear power plants, the return of spent fuel from a foreign country to its country of origin (Russia) has never taken place so far. In the investigations made, there seems to be two alternative, interim solutions for Hungary:

- Interim storage and final disposal;
- Reprocessing.

Since some of the spent fuel is being stored for 35 years in water, the subject of long-term behavior of the cladding is also becoming an important issue. The present tasks are related to these issues, i.e. improvement of storage conditions, investigations of long-term interim storage and review of alternative solutions. For a small country, like Hungary, it makes sense to review the future alternatives in co-ordination for the NPP and the research community. The study of the possibility of loading research reactor spent fuel in the Paks MVDS, is one step in this direction.

## 7. FUTURE TASKS RELATED TO THE BACK END OF THE FUEL CYCLE

The strategy for the back-end of the fuel cycle for the Hungarian Republic related to the nuclear power plant is shipping the fuel back to Russia for reprocessing under the given "conventional" conditions, and the "wait and see" for the decision between reprocessing and direct disposal in the case, where reprocessing (return shipment) would no longer be possible.

The basic objective can be formulated as follows:

The final disposal facility for high-level radioactive wastes produced during spent fuel reprocessing, or for assemblies packed in containers in the case of their direct disposal, should be put into operation in Hungary by 2040.

In February 1992 the National Atomic Energy Commission made a decision to develop a project for the solution of the final disposal of radioactive wastes produced in the nuclear power station. Within the framework of the project, in its first phase a complex radioactive waste management strategy was elaborated. It seems that there will be two tasks, namely

- the disposal of low- and intermediate level radioactive wastes, as a more urgent issue, and
- the selection of the site and technology for the disposal of high level radioactive wastes and spent fuel.

In the new Atomic Energy Law, and the related decrees published this summer, a new non-profit organization is named as the responsible organization to take care of all relevant issues. All nuclear operators (NPP Paks and the research reactors) will be contributing to the so called Nuclear Waste Fund starting January 1998. Discussions are still going on about the status and responsibilities of the company managing the Fund.

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**Abstract**

From Indian point of view, the spent fuel management by the reprocessing and plutonium recycle option is considered to be a superior and an inevitable option. The nuclear energy programme in Indian envisages three stages of implementation involving installation of thermal reactors in the first phase followed by recycling of plutonium from reprocessed fuel in fast breeder reactors and in the third phase utilization of its large thorium reserves in reactor system based on U-233-Th cycle. The Indian programme for Waste Management envisages disposal of low and intermediate level radioactive waste in near surface disposal facilities and deep geological disposal for high level and alpha bearing wastes. A Waste Immobilization Plant (WHIP), employing metallic melter for HLW vitrification is operational at Tarapur. Two more WIPs are being set up at Kalpakkam and Tarapur. A Solid waste Storage Surveillance Facility (SSSF) is also set up for interim storage of vitrified HLW. Site investigations are in progress for selecting site for ultimate disposal in igneous rock formations. R&D works is taken up on partitioning of HLW. Solvent extraction and extraction chromatographic studies are in progress. Presently emphasis is on separation of heat generating short lived nuclides like strontium and alpha emitters.

**1. BACKGROUND**

The light water reactors (LWR) have been the mainstay of nuclear power programme in many parts of the world, with PHWRs and FBRs contributing significantly to this effort in a few countries. Over the past few decades, the operation of these uranium based power reactors and the various research reactor systems have led to increased fissile inventories of plutonium in the spent fuel. During the first generation nuclear fuel cycle activities, reprocessing and recycle of uranium and plutonium for power generation was perceived by many countries to be among the best of long term strategies for the management of spent fuel. In this context, from the Indian point of view, the reprocessing and plutonium recycle option is not only considered to be superior option but also to be an inevitable one.

**2. FUEL CYCLE STRATEGY**

The nuclear energy programme in India envisages three stages of implementation involving installation of uranium fueled thermal reactors in the first phase followed by utilization of plutonium in fast breeder and other types of reactors and in the third phase, utilization of reactor systems based on U233- Th cycle. The first phase of the programme is essentially based on the utilization of PHWRs for power generation with fuel reprocessing, plutonium recycle and efficient waste management as the strategies for the back end of the Fuel Cycle. The choice of the Reprocessing and Plutonium Recycle option has endowed the programme with a variety of mid course options in both uranium and thorium fuel cycle with plutonium forming the vital link between the two.

**3. POWER PROGRAMME**

Besides the two BWRs at Tarapur, there are several operating PHWRs with a design capacity of 220 MWe each. Few more reactors of similar type including two reactors each of 500 MWe are under different stages of planning, construction and commissioning. Under Fast Breeder Reactor (FBR) technology development programme, a 40 MWth Fast Breeder Test Reactor(FBTR) is operational at Kalpakkam and the design of a 500 MWe Prototype Fast Breeder Reactor (PFBR) is in progress. In addition to PHWRs and FBRs, it is proposed to include LWRs and Advanced Heavy Water Reactors (AHWR) in the power programme. (see Table I). Over the years, in tandem with the increase in spent fuel arisings from the growth of nuclear power, the reprocessing and nuclear waste management capabilities have been augmented. There are now two reprocessing facilities to treat

spent fuels from PHWRs. (see Table II). Based on the plutonium based fuel fabrication experience at pilot plant scale in Trombay, a sophisticated industrial scale Advanced Fuel Fabrication Facility (AFFF) has been setup at Tarapur to meet the MOX fuel fabrication requirements. These facilities will meet the fuel requirements of MOX for thermal reactors, FBTR and the initial startup requirements of FBR. A larger facility to cater to the fuel requirements for FBR is planned at Kalpakkam.

TABLE I. INDIAN NUCLEAR POWER PROGRAMME

Station	Reactor type	Capacity (MWe)	Year of Criticality
<b>I. Units in Operation</b>			
1. TYPS -1&2	BWR	2 X 160	1969
2. RAPS-1	PHWR	220	1972
3. RAPS-2	PHWR	220	1980
4. MAPS-1	PHWR	220	1983
5. MAPS-2	PHWR	220	1985
6. NAPS-1	PHWR	220	1989
7. NAPS-2	PHWR	220	1990
8. KAPS-1	PHWR	220	1993
9. KAPS-2	PHWR	220	1994
<b>II. Units under Construction</b>			
1. RAPP-3	PHWR	220	1998
2. RAPP-4	PHWR	220	1999
3. KAIGA-1	PHWR	220	1998
4. KAIGA-2	PHWR	220	1999
<b>III. Units Planned</b>			
1. KAIGA-3&4	PHWR	2 x 220	2006
2. TARAPUR-3&4	PHWR	2 x 500	2008
3. KUDANKULAM	VVER	2 x 1000	2010
4. KALPAKKAM	FBR	500	2010

#### 4. WASTE MANAGEMENT

The Indian programme on safe management of radioactive wastes envisages two distinct modes of final disposal in respect of radioactive wastes; near-surface engineered, extended storage for low and intermediate-level radioactive wastes and deep geological disposal for high-level and alpha bearing wastes. A waste immobilization plant for the treatment of HLW is operational at Tarapur. It is a semi continuous pot glass process involving calcination followed by melting in the process vessels. Two more waste immobilization plants are being set up at Trombay and Kalpakkam. Use of joule heated ceramic melters is under development. A solid storage and surveillance facility (SSSF) has also been set up for interim storage of vitrified HLW. As regards ultimate disposal, the Indian choice is focused on igneous rock formations and some selected sedimentary deposits. Investigations are in progress for evaluation of candidate sites for a repository .

TABLE II. SPENT FUEL ARISING FROM PHWRs (Fuel Burn-up Average : 6,600 MWd/tU)

Status Capacity	Power	Spent fuel generation (GWe/yr)	Reprocessing arisings	
	(MWe)		Plant 1 (tHM/yr)	Plant 2 (tHM/yr)
Existing	8x220	1.76	290	100
Under construction/ design	4x220	0.88	145	100
Under planning	2x500	1.00	165	300
	2x200	0.44	73	
<b>Total</b>	<b>16</b>	<b>4.08</b>	<b>673</b>	<b>500</b>

## 5. ACTINIDE SEPARATION

The main objective of partitioning high level waste is that it shall lead to a safer waste, more acceptable to the public. The removal of long-lived alpha emitting actinides from these wastes under the partitioning and transmutation (P&T) option would greatly reduce their long-term radiological hazards. Removal of shorter lived fission products like <sup>90</sup>Sr can reduce the heat generation from these wastes. Further, recovery of useful nuclides from this waste will make the waste management with P&T more economical and viable. From the Indian context, the present studies are limited to the partitioning of the long lived actinides from the HLW as any reduction in the alpha burden of these wastes would render them safer with respect to long term disposal. At appropriate time, a long term policy on the final utilization / transmutation of the recovered actinides would be evolved, based on the available state of the art technology at that point of time.

## 6. R&D WORK

Studies are in progress for the quantification of the PHWR spent fuel arisings, the radiological source terms of the relevant actinides and fission products in Purex HLW after reprocessing and evaluation of their hazard ranking. CMPO based solvent extraction and extraction chromatographic studies with HLW are in progress to propose suitable flow sheets for partitioning of the relevant actinides from these wastes and to reduce the alpha burden to very low levels. Other extractants are also being explored in this context. From reprocessing angle, improvements in Purex process to reduce Pu losses to the waste, recovery of neptunium and possibly <sup>99</sup>Tc in the process are some of the aspects receiving attention. Alkaline trapping for <sup>129</sup>I is also envisaged. However, an advanced fuel cycle with complete recycle of uranium and transuranium elements will have many additional steps like efficient minor actinide separation from HLW, MOX fuels or target fabrication for fast or thermal reactors or transmutation facilities, reprocessing of spent FR/MOX fuel with quantitative and multiple recycling for transuranium elements depletion and final storage/ disposal of highly radioactive spent fuels from Fast reactor. All these technologies are still to be evaluated and mastered.



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### **Abstract**

In Japan 52 commercial nuclear power units are now operated, and the total power generation capacity is about 45 GWe. The cumulative amount of spent fuel arising is about 13,500 tU as of March 1997. Spent fuel is reprocessed, and recovered nuclear materials are to be recycled in LWRs and FBRs. In February 1997 short-term policy measures were announced by the Atomic Energy Commission, which addressed promotion of reprocessing programme in Rokkasho, plutonium utilization in LWRs, spent fuel management, backend measures and FBR development. With regard to the spent fuel management, the policy measures included expansion of spent fuel storage capacity at reactor sites and a study on spent fuel storage away from reactor sites, considering the increasing amount of spent fuel arising. Research and development on spent fuel storage has been carried out, particularly on dry storage technology. Fundamental studies are also conducted to implement the burnup credit into the criticality safety design of storage and transportation casks. Rokkasho reprocessing plant is being constructed towards its commencement in 2003, and Pu utilization in LWRs will be started in 1999. Research and development of future recycling technology are also continued for the establishment of nuclear fuel cycle based on FBRs and LWRs.

## **1. INTRODUCTION**

Japan has scarce energy resources and depends on foreign resources for most of its energy needs. Therefore, Japan has made efforts to utilize nuclear power since mid-1950s, by carrying out research and development, and to promote commercialization of peaceful use of nuclear energy. Since its initial stage, the development and utilization programme has consistently called for recycling of nuclear fuel. Today nuclear energy plays an important role as a key energy source and the nuclear power generation accounts for about 34 % of the total electric power generation.

This report shows Japan's basic policy on nuclear energy in view of long and short terms briefly, and then describes the current status and prospects of generation, storage and transportation of spent fuel. Some explanation is given on the research and development of spent fuel storage technology. Current status and future plan on reprocessing and recycling of U and Pu, and furthermore, brief description of the progress in radioactive waste management are also given.

## **2. BASIC POLICY, CURRENT STATUS & FUTURE PROSPECTS ON SPENT FUEL MANAGEMENT**

### **2.1. Long-term programme for research, development and utilization of nuclear energy**

Basic policy on nuclear energy is defined in Long-Term Programme for Research, Development and Utilization of Nuclear Energy (hereinafter referred to as Long-Term programme). Long-Term programme is determined by Atomic Energy Commission of Japan (the AEC) and revised approximately every five years. According to the Atomic Energy Basic Law, it is required that research, development and utilization of nuclear energy are limited to peaceful purposes and that assurance of safety is the foremost consideration in them.

In 1994, the AEC revised the Long-Term Programme. The programme intends to guarantee 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. Furthermore, recycling of nuclear fuel contributes to preserving resources and the environment, and to improved management of radioactive waste.

In the basic concept of the Long-Term Programme, the fast breeder reactor (FBR) is the kernel of nuclear power generation in the long-term prospect, together with Light Water Reactors (LWRs). Research and development is to be undertaken, in cooperation with the government and the private sector each other, to establish a commercial system of nuclear fuel recycling based on FBRs by around 2030. Also, construction of a commercial reprocessing plant and Pu utilization in LWRs are steadily promoted. Experience with nuclear fuel recycling with LWRs is considered important in order to establish a comprehensive technological system of plutonium utilization which is indispensable to the above system based on FBRs.

Nuclear fuel recycling is promoted on the principle of not possessing plutonium beyond the amount required to implement the programme, i.e. the principle of no surplus plutonium, as well as having very strict management of nuclear materials, coupled with transparency so as to provide assurances regarding adherence to non-proliferation of nuclear weapons.

## **2.2. Policies to promote the nuclear fuel cycle in the short term**

After the sodium leak accident in the secondary system of the FBR prototype reactor "MONJU" in December 1995, the government made 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.

The AEC has deliberated and decided short-term concrete measures of policy concerning the nuclear fuel cycle at the end of January 1997, taking into account the outcomes of the discussion by the Advisory Committee for Energy which is an advisory body to the Minister of International Trade and Industry (MITI). The measures were consented by the Cabinet in February 1997 and are commitment 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 the utilization with three or four reactors loading MOX fuel by 2000, expanding the 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 local public.
  - Initiate a study aiming at an early conclusion on the development of necessary environment to enable spent fuel storage at away-from-reactor sites by around 2010, in addition to the storage at reactor sites, given the increasing quantities of stored spent fuel in the long-term prospect.
- 3) **Backend 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 Special Committee on FBRs established under the AEC.

In March 1997, a fire and explosion occurred at the Bituminization Demonstration Facility of Tokai reprocessing plant of Power Reactor and Nuclear Fuel Development Corporation (PNC). Investigation of this accident is being intensively carried out. PNC will be reorganized and formed into a new body. In June 1997 chairman of the AEC announced a statement of reconfirming the Cabinet Consent in February 1997, in which establishment of nuclear fuel cycle should be promoted as before.

### **2.3. Current status and future prospects of spent fuel management**

Reprocessing service will be provided by the Tokai reprocessing plant, the Rokkasho reprocessing plant and reprocessing contracted to BNFL and COGEMA. Tokai reprocessing plant, which has an annual treatment capacity of around 90 tU, will shift its major role to research and development of future reprocessing technologies after Rokkasho reprocessing plant begins operation. The Rokkasho reprocessing plant, Japan's first commercial reprocessing plant, will have an annual treatment capacity of 800 tU, and is scheduled to go into operation in 2003. The reprocessing capacity and technology of the second commercial reprocessing plant will be decided around 2010. Regarding spent fuel exceeding the available reprocessing capacity, it will be properly stored and managed as an energy stockpile, until reprocessed.

## **3. SPENT FUEL ARISING, STORAGE AND TRANSPORTATION**

### **3.1. Spent fuel arisings/transportation**

As of the end of August 1997, 52 commercial nuclear power units are in operation in Japan, and the total electric power generation capacity is about 45 GWe. According to the report written by the Electric Utility Industry Council, a government advisory organization, nuclear power generation capacity will increase to 70 GWe in 2010. The cumulative amount of generated spent fuel was about 13,500 tU (LWR: about 12,300 tU, GCR: about 1,200 tU) as of March 1997. It is estimated from the projected power generation capacity that the annual generation rate of spent fuel for the years 2000 and 2010 will be 800 to 1,000 tU/y and 1,000 to 1,500 tU/y, respectively (see Fig. 1). In the near future spent fuel with higher burnup up to 55,000 MWd/t and spent MOX fuel from LWRs will also be generated.

Part of spent fuel generated in Japan has been transported to the reprocessing plants. This cumulative quantity is about 7,800 tU as of March 1997. About 6,800 tU of spent fuel has already been shipped to overseas reprocessors (LWR spent fuel: about 5600 tU, GCR: about 1200 tU) as of March 1997, and the rest to Tokai reprocessing plant. From this year to around 2003, about 300 tU of spent fuel is scheduled to be delivered additionally to overseas reprocessing plants.

### **3.2. Storage**

#### *3.2.1. Current status and prospects*

The total amount of spent fuel stored in nuclear power station pools is approximately 5,800 tU as of the end of March 1997. Part of the spent fuel will be shipped to the Rokkasho reprocessing plant, when the pool of spent fuel in the plant goes into operation. The spent fuel also continues to be shipped to the Tokai reprocessing plant and to overseas reprocessing plants according to the existing contracts.

Total controlled-storage capacity (capacity available for spent fuel storage) at LWR sites is about 9,920 tU as of March 1997. Expansion of storage capacity at reactor sites has been carried out and is being planned by alternation of racks in existing storage pools (re-racking), common use of

pools at the reactor site (use of a pool by two or more reactors), building additional pools or dry-cask storage facilities. These activities will ensure sufficient storage capacities required towards 2010. Examples of recent activities for expansion of storage capacity are shown below.

- Re-racking:  
Tsuruga nuclear power station (No.2), Japan Atomic Power Company  
Hamaoka nuclear power station (No. 3), Chubu Electric Power Company
- Common use of pools inside the reactor site:  
Onagawa nuclear power station (No. 2, 3), Tohoku Electric Power Company  
Kashiwazaki-Kariwa nuclear power station(No. 3, 4, 6, 7), Tokyo Electric Power Company
- Expansion of pool:  
Fukushima-Dai-ichi nuclear power station (No. 1-6), Tokyo Electric Power Company (construction of additional pool for common use)  
Ohi Nuclear power station (No. 3, 4), Kansai Electric Power Company (Use of reserve pits)
- Construction of dry storage facility inside the reactor sites:  
Fukushima-Dai-ichi nuclear power station (No. 4, 5, 6), Tokyo Electric Power Company (operated since 1995)

In addition, corresponding to the Cabinet consent in February 1997, a study was initiated, by MITI, Science and Technology Agency (STA) and Japanese electric utilities in March 1997, on practical issues such as safety, new technologies, management of storage facility and siting, so that spent fuel storage at away-from-reactor sites can be facilitated by around 2010, in addition to the storage at reactor sites.

### 3.2.2. Research activities on spent fuel storage

Research and development on spent fuel storage has been conducted by Central Research Institute of Electric Power Industry (CRIEPI) and Japan Atomic Energy Research Institute (JAERI).

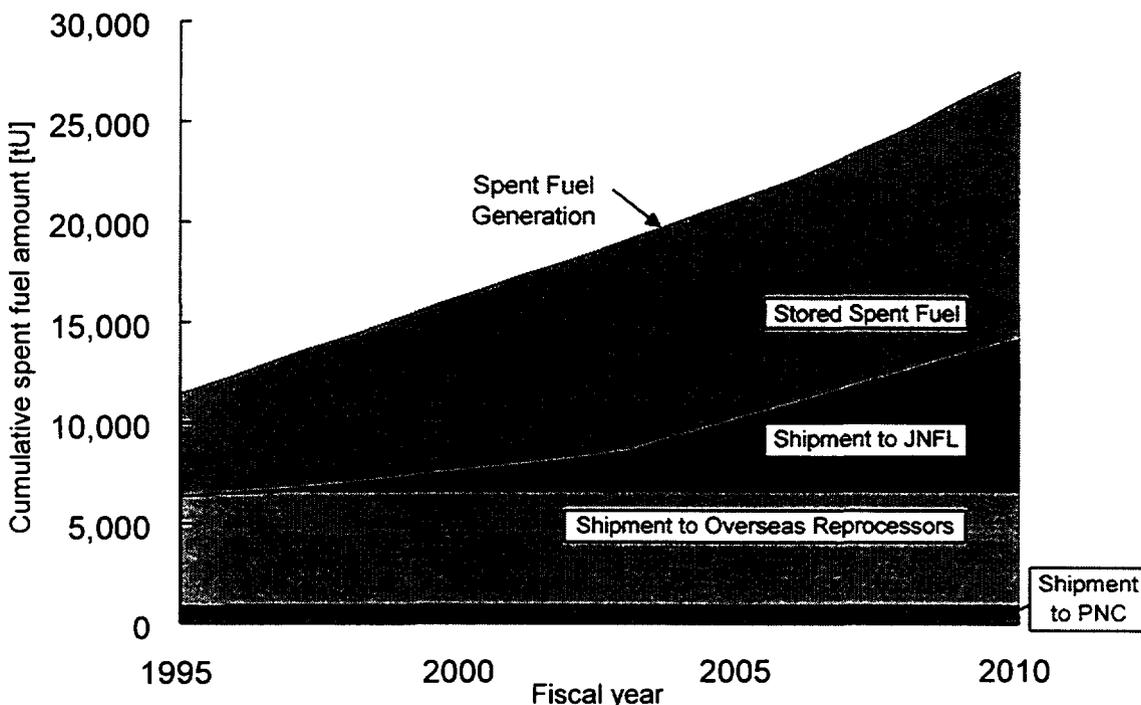


Figure 1. Prospect of LWR-spent fuel management

### ***Dry storage***

Research and development on spent fuel dry storage technology has been carried out mainly by CRIEPI under contracts with the government. In a study programme until 1996, safety and cost evaluations for various types of dry storage system applicable to the storage at reactor site were carried out for spent high burnup fuel and MOX fuel. Especially, for metal cask storage technology, the safety of the total system was confirmed. Some of the outcomes were incorporated in the safety review guide on "Dry Cask Storage of Spent Fuel in Nuclear Power Plants" (August 1992, Nuclear Safety Commission Japan (NSC)). They were used in the licensing process of the first dry cask storage with a capacity of about 73 tU (9 casks) at the reactor site of Fukushima-Dai-ichi nuclear power station of Tokyo Electric Power Company. The storage facility has been operated since 1995 and will be expanded as far as 20 casks.

For the storage programme away from reactor site, heat removal characteristics of the cask, vault and horizontal concrete silo storage systems have been made clear, and they are expected to cut down excessive safety-margin included in the conventional designs. In 1997, a new study programme of demonstrative tests has started, which is mainly related to concrete module storage technology, such as a horizontal concrete silo and a concrete cask. The concrete cask is considered to essentially have economical advantage. The following items are addressed in the programme.

- tests on long-term performance related to carbonation and salt damage of concrete material.
- tests on dynamic strength for concrete material to evaluate the integrity of concrete structures in case of accidents.
- tests on fracture toughness and corrosion resistance for the welded lid of a stainless steel canister installed in the cask.

### ***Burnup credit***

JAERI has conducted fundamental studies to implement the burnup credit into the criticality safety design of storage and transportation (S/T) casks for high burnup fuel under auspices of Science and Technology Agency Japan. Computer codes and data which are vital to cope with the above implementation have been developed.

Some spent fuels discharged from Japanese PWRs have been measured with nondestructive and destructive methods so that their burnup and nuclide composition data are obtained. Subsequently, burnup analysis computer codes are validated with these experiment data, and combination of particular codes for criticality, shielding and thermal analyses are studied by comparing calculated results with corresponding safety criteria in applying to the pragmatic S/T cask design by taking into account the burnup credit concept.

In addition to the fundamental study on the burnup credit, *The Nuclear Criticality Safety Handbook of Japan* was published in 1988, and its English translation was issued in 1995. The data and information in this handbook were used for criticality safety design of Rokkasho reprocessing plant. In particular burnup credit concept was introduced into the criticality safety design of spent fuel storage rack of the plant. After the publication of the handbook, supplemental work has been conducted. Assessment of criticality safety margin for chemical process taking into account burnup credit was included, together with recommended nuclide composition values to be taken for criticality safety evaluation by assorting data described in open literature concerning post irradiation examination (PIE) of LWR spent fuel. The second version of the handbook will be published by arranging these new data with the first version.

## **4. REPROCESSING OF SPENT FUEL**

### **4.1. Tokai reprocessing plant**

The Tokai reprocessing plant, the first reprocessing plant in an industrial scale in Japan, is owned and operated by PNC. The plant with a capacity of 0.7 tU/day went into the hot test operation

in September 1977 and was commissioned in January 1981. Although the plant has experienced several troubles with long interruptions, the total amount of 936 tU of spent fuel including 10 tHM of spent MOX fuel from Advanced Thermal Reactor (ATR) "FUGEN" has been reprocessed by the end of March 1997 (see Fig. 2).

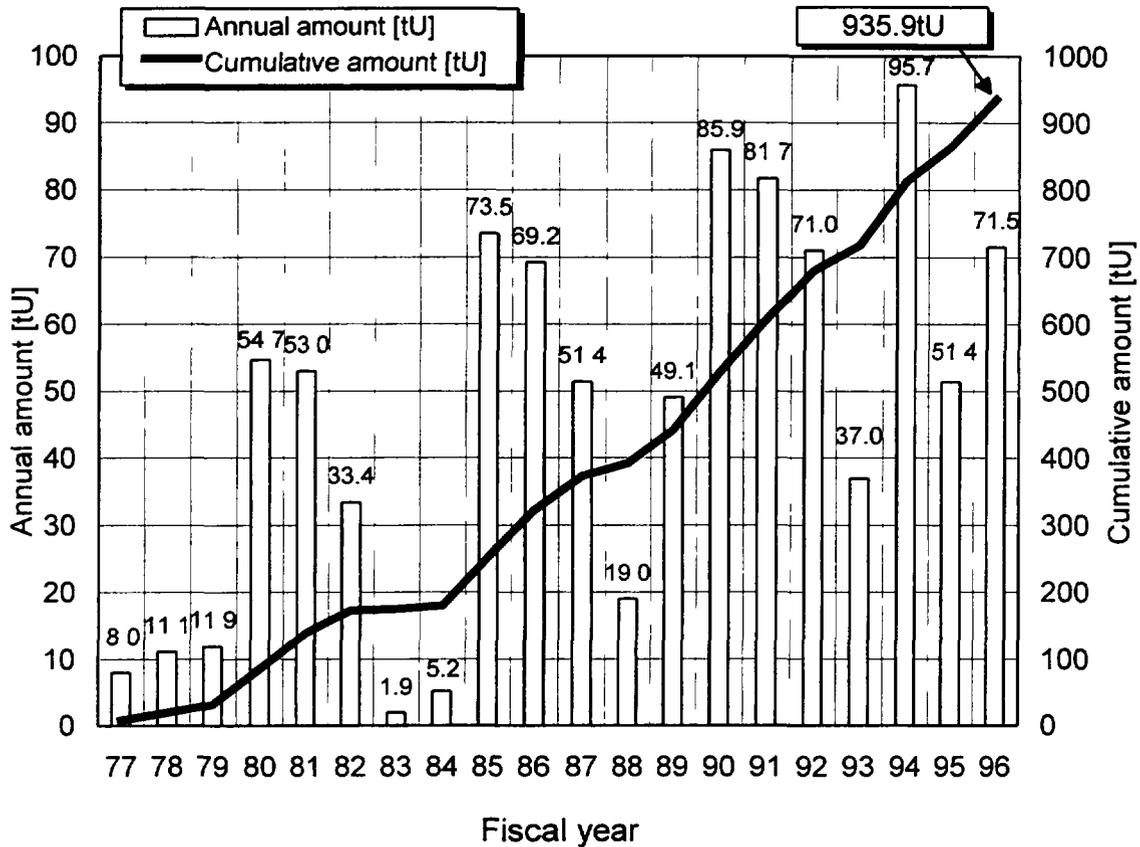


Figure 2. Operating history of Tokai reprocessing plant (as of 3/31/1997)

#### 4.2. Rokkasho reprocessing plant

Japan Nuclear Fuel Ltd. (JNFL) has started construction of a reprocessing plant with a capacity of 800 tU/year in April 1993, in Rokkasho Village, Aomori-Prefecture. Principal facility specification is shown below.

##### Reprocessing plant

Method: PUREX method  
 Capacity: 800 tU/year  
 4.8 tU/day (maximum)

##### Spent fuel storage pool

Spent fuel: 3,000 tU  
 Residual enrichment: less than 3.5 wt.%  
 Cooling time: over 1 year before receipt  
 over 4 years before reprocessing  
 Burnup: 55,000 MWd/tU (max.)  
 45,000 MWd/tU (average)

JNFL applied for the authorization of reprocessing business through the STA to the Prime Minister in March 1989. The first-step review by STA for reprocessing was completed in August 1991, and the second-step review by the AEC and NSC was finished in December 1992.

JNFL intended to modify mainly the purification process in order to rationalize the plant design. The modification required the first- and second-step reviews again. The first-step review was finished in December 1996 and the second-step review was completed in July 1997. Figure 3 shows construction / operation schedule presently planned. Spent fuel storage pool is planned to be operated in October 1997. The reprocessing plant will go into operation in January 2003.

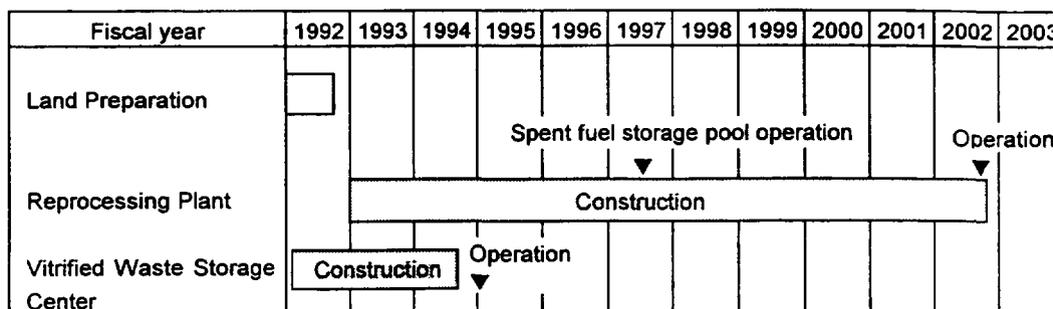


Figure 3. Construction/operation schedule for Rokkasho reprocessing plant and Vitrified Waste Storage Center

### 4.3. R&D activities in fuel reprocessing

#### 4.3.1. LWR fuel reprocessing

The Tokai reprocessing plant has a role as the pilot plant for establishing reprocessing technology. Two kinds of R&D programmes in PNC are carried out: short- and medium-term programmes to improve plant operation, and long-term programmes to pursue advanced technology of reprocessing.

JNFL promoted various R&D works, in order to ensure stable and reliable operation of the Rokkasho reprocessing plant. Recently, JNFL is working on the development of operation and maintenance technologies.

JAERI has promoted researches for safety criteria and evaluation, advanced reprocessing and waste treatment. In order to support these research activities, NUCEF (Nuclear Fuel Cycle Safety Engineering Research Facility) commissioned in 1994 the hot operation of STACY (Static Experiment Critical Facility) and TRACY (Transient Experiment Critical Facility) for criticality safety experiments, and of alpha and gamma cells for reprocessing studies.

#### 4.3.2. FBR fuel reprocessing

For the FBR fuel reprocessing, PNC has developed its own process and equipment as well as remote handling technology through large scale cold mock-up tests and laboratory scale hot tests. The Chemical Processing Facility (CPF) has been used for hot process tests since 1982. The Recycle Equipment Test Facility (RETF) is for hot engineering-scale equipment test and is under construction.

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

### 5.1. Future nuclear fuel recycling programmes

For a certain period, LWRs continue to be a major source in Japan's nuclear power generation programme and some of them will use recovered plutonium as stated in section 2. FBRs will play in the future a central role in the nuclear fuel recycling system and will be the principal reactors to use

recovered plutonium in combination with LWRs. Rokkasho reprocessing plant will be the main source to supply plutonium to the LWRs and FBRs in the future. The reprocessing capacity and technology of the second commercial reprocessing plant will be decided. Recovered plutonium in overseas reprocessing plants will be fabricated in overseas fuel manufacturer and utilized as MOX fuel in LWRs of Japan. It is necessary to construct a domestic commercial MOX fabrication plant for LWRs, taking account of the operation plan of Rokkasho reprocessing plant.

## 5.2. Project of fast breeder reactor

The construction of MONJU, a loop type LMFBR of 280 MWe output, was completed late in April 1991. The pre-operational test started in May in the same year. The test consisted of a function test and a start-up test. The function test was finished in 1992. The reactor had reached initial criticality in April 1994 and started the generation of electricity in August 1995. With regard to the sodium leak accident occurred in December 1995, STA announced the final report in February 1997. And PNC completed the investigations of the cause of the accident in March 1997. A total safety evaluation of the MONJU plant is now being performed in order to improve its safety. As the Cabinet consent in February 1997 showed, further strategies for development of fast breeder reactor, including future management of "MONJU," are under discussion of the Special Committee on FBRs established under the AEC.

## 5.3. Programme for utilization of MOX fuel in LWR

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 shown in 2.2, the Federation of Electric Power Companies of Japan announced a programme for the MOX utilization in LWRs in February 1997 as listed in Table I.

TABLE I. JAPANESE UTILITIES' MOX UTILIZATION PROGRAMME

Year	1999	2000	Shortly after 2000	2010
Cumulative number of reactors	2	4	9	16 - 18

## 5.4. MOX fuel fabrication

Development of MOX fuel fabrication by PNC started at the Plutonium Fuel Development Facility (PFD) in 1965. To fabricate fuels for FBRs and ATR, PNC has been operating the Plutonium Fuel Fabrication Facility (PFFF) and the Plutonium Fuel Production Facility (PFPF). A domestic LWR MOX fuel fabrication plant for commercial operation will have a capacity of around 100 t of MOX fuel per year.

## 5.5. Utilization of recovered uranium

Recovered uranium can be converted to uranium hexafluoride followed by re-enrichment and re-conversion, can be mixed with enriched uranium, or can be mixed with plutonium to be recycled as MOX fuel. Re-enrichment is considered to be the best method of recycling uranium in terms of the economy and the amount of usable uranium recovered. About 240 t of recovered uranium will be converted to uranium hexafluoride by PNC under a contract with Japanese utilities.

## 5.6. R&D for advanced nuclear recycling technology

For the future nuclear 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. Long-term research and development will be conducted 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. MANAGEMENT OF RADIOACTIVE WASTES

### 6.1. High-level radioactive wastes

High-level radioactive waste (HLW) that results from the reprocessing of spent fuel is solidified into a stable form, known as vitrification, and after storage for a period of 30 to 50 years to allow cooling, and it is to be disposed of deep in the ground. With regard to the development of technology for treatment of HLW, at PNC's Tokai Vitrification Facility (TVF), PNC started its operation after receiving its full operational license from the government in December 1995. By the end of July 1997, 62 vitrified wastes have been produced.

The Vitrified Waste Storage Center was constructed by JNFL, where high-level radioactive waste in glass form returned from overseas reprocessing plants is stored, after the necessary license was obtained from the government in April 1992. Operation was started in April 1995, as indicated in Fig. 3, with a capacity of 1,440 canisters. In March 1997 the second returned shipment of 40 canisters of vitrified HLW from France was completed. As of the end of August 1997, the number of cumulative vitrified HLW returned from overseas reprocessing plant in the Vitrified Waste Storage Center are 68 canisters.

In regard to the HLW final disposal, the Special Committee on High-Level Radioactive Waste Disposal of the AEC is discussing social and economic aspect of the disposal, in order to promote the public understanding and acceptance. Technical aspect and the R&D programmes for the disposal are discussed by the Advisory Committee on Nuclear Fuel Cycle Backend Policy of the AEC. In addition, Steering Committee on High-Level Radioactive Waste Project (SHP), under the Council for Promoting High-Level Waste Disposal, was established in May 1993. The purpose of SHP is to promote the preparation for the project for the safe and environmentally acceptable disposal of high-level radioactive waste, with public understanding and cooperation, by conducting research and investigation.

## 7. CONCLUSIONS

Japan intends to guarantee 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 Pu and U to allow the reuse of these materials as nuclear fuel in LWRs and FBRs.

The AEC announced the short-term policy measures with regard to the promotion of nuclear fuel cycle. Concerning spent fuel management, the policy measures included the expansion of storage capacity at reactor sites and the study on the option of storage in the facilities at away-from-reactor sites in addition to the storage at reactor sites.

Research and development on reprocessing technology have been conducted. Also, research and development on dry storage technology and fundamental study to implement the burnup credit into the criticality safety design have respectively been carried out. Outcomes of those studies were incorporated in safety review guide and safety design.

Based on the Long-Term Programme and the short-term policy measures, Japan will steadily promote the reprocessing programme for the plant under construction in Rokkasho and make efforts to establish nuclear fuel cycle.



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**Abstract**

Korea has selected nuclear energy as the major source for the electric power generation due to the insufficiency of energy resources in Korea. In compliance with the policy, Korea Electric Power Corporation (KEPCO) has expanded the nuclear power programme and faced the significant arisings of spent fuel. The interim At Reactor(AR) storage pools have very limited capacities and temporary expansion of this capacity has been taken such as re-racking and dry storage construction. There was a plan, to construct a centralized spent fuel storage facility, which was postponed officially by the government. Under the current situation, it is hard to establish the long-term spent fuel management strategy. 'Wait and See' is no more applicable to Korea. because of storage shortage. Within R&D, dry storage construction and DUPIC fuel cycle are being considered. In this paper, the spent fuel management programme of Korea is briefly reviewed.

**1. NUCLEAR POWER PROGRAMME**

The electricity demand in Korea has grown rapidly as high as 10% on the average every year during the last decade and Korea expects to have the similar or slightly less growth trend of electricity demand for a while. In order to cope with the future electricity demands, the long-term power development plan has been updated bi-annually. The total installed capacity in Korea was 36 GWe in 1996, 27% from nuclear, 22% from coal, 24% from LNG, 18% from oil and 9% from hydro and others. The generated energy in 1996 was 205,494 GWh. The share was 36% by nuclear, 28% by coal, 13% by LNG, 21% by oil and 2% by hydro and others.

According to the last updated plan in 1995, additional construction of generating units of 57 GWe from 1995 to 2010 with the number of added 105 units will be required. These facilities will be comprised of 19 nuclear units with a total capacity of 19.3 GWe, 29 coal-fired plants with 15.5 GWe, 40 LNG combined cycle units with 17.4 GWe, 29 pumped storage power plants and hydro plants with 3.5 GWe, and other type of plants with 1.3 GWe. After completion of the development programme, the total installed capacity in 2010 will be around 80 GWe. The share of nuclear power installed capacity will be 33.1% and about 45% of the generated electricity in Korea will be produced by nuclear power (because the nuclear power plants will be operated in the base-load mode due to their low cost of generation). Table 1 shows the long-term nuclear power development plan by the year of 2010.

Twelve nuclear power plants (10 PWRs and 2 CANDUs) have been in operation since 1978 when the first nuclear power plant started its operation at the Kori site. The total generation capacity is as much as 10.3 GWe. Two other units of 1000 MWe class Korean standard PWR and 2 units of 700 MWe CANDU are scheduled for commissioning by the end of this century. By then, a total of 16 nuclear stations with 12 PWRs and 4 CANDUs will be connected to the grid in Korea. As of September 1997, the nuclear share of the total electricity generating capacity is about 25%.

**2. SPENT FUEL ARISING AND STORAGE IN KOREA**

The amount of spent fuels generated per year is different from reactor to reactor according to the reactor capacity, reactor type and reload cycle. Based on the current status, about 4,500 tU of spent fuels is projected to be accumulated by the year 2000. Moreover, the annual arisings of spent fuel discharged from the total of 16 units which are expected to be in operation by 2000 will be approximately 550 tU/yr. After the year of 2010, when 25 units (Kori 1 will be decommissioned in 2009) will be in operation, the annual discharge rate of spent fuel will be as much as 880 tU/yr.

Table 2 shows the storage capacity and the annual and cumulative spent fuel arisings as of 1997. From the table, it can be seen that soon a shortage of storage capacity will occur, if no measures are taken.

TABLE 1. NUCLEAR POWER PLANT CAPACITY IN KOREA

Status	Plant Name	Reactor Type	Commercial Operation	Capacity (MWe)
In Operation	Kori 1	PWR	1978. 4	587
	Kori 2	PWR	1983. 7	650
	Wolsung 1	PHWR	1983. 4	679
	Kori 3	PWR	1985. 9	950
	Kori 4	PWR	1986. 4	950
	Yeongkwang 1	PWR	1986. 8	950
	Yeongkwang 2	PWR	1987. 9	950
	Uljin 1	PWR	1988. 9	950
	Uljin 2	PWR	1989. 9	950
	Yeongkwang 3	PWR	1994. 9	1000
	Yeongkwang4	PWR	1995. 9	1000
	Wolsung 2	PHWR	1997. 7.	700
	Under Construction	Wolsung 3	PHWR	1998
Wolsung 4		PHWR	1999	700
Uljin 3		PWR	1998	1000
Uljin 4		PWR	1999	1000
Yeongkwang 5		PWR	2001	1000
Yeongkwang 6		PWR	2002	1000
Long-term Projection	Std. PWR 1	PWR(PHWR)	2005	1000
	Std. PWR 2	PWR(PHWR)	2006	1000
	KNGR 1	PWR	2007	1300
	KNGR 2	PWR	2008	1300
	KNGR 3	PWR	2008	1300
	KNGR 4	PWR	2009	1300
	Std. PWR 3	PWR	2009	1000
	Std PWR 4	PWR	2010	1000

TABLE 2. STATUS OF SPENT FUEL STORAGE IN KOREA (AS OF JUNE, 1997)

UNIT: tU

Name of Site	Storage Capacity	Annual Arisings	Cumulative Arisings
Kori	1,533	66.3	925
Wolsung	3,159*	193.6	1,325
Yeongkwang	1,271	80	491
Uljin	709	38	331
Total	6,672	377.9	3,072

\* included AFR storage(812t) under construction.

### 3. SERIES OF EFFORTS TO INCREASE THE STORAGE CAPACITY

Korea has been taking realistic measures to expand the storage capacity. There has been an attempt to increase the burnup of the fuels through the longer reload cycle, which results in less annual discharge. Table 3 shows the current status of fuel burnup and reload cycle for the operating nuclear reactors in Korea with the future fuel burnup and reload cycle to be implemented.

TABLE 3. RELOAD CYCLE AND FUEL BURNUP

Units	Reload Cycle (Month)		Discharged Burn-Up (MWd/tU)		Annual Discharge (t)
	Current	Future	Current	Future	
Kori 1	15	18	37,000	42,000	13.2
Kori 2	15	18	37,000	42,000	15.1
Kori 3	18	18	44,000	47,000	19
Kori 4	18	18	44,000	47,000	19
Yeongkwang 1	18	18	44,000	47,000	19
Yeongkwang 2	18	18	44,000	47,000	19
Yeongkwang 3	12	15 -> 18	43,000	45,000	21
Yeongkwang 4	12	15 -> 18	43,000	45,000	21
Ulchin 1	18	18	44,000	47,000	19
Ulchin 2	18	18	44,000	47,000	19
Wolsung 1	Daily Load	Daily Load	7,100	7,100	97.6
Wolsung 2	Daily Load	Daily Load	7,300	7,300	96

In addition to burnup increase, the storage capacity has already been expanded with replacing high density storage racks employing boron to absorb neutrons and allow closer spacing of the stored assemblies for PWR and constructing AFR dry storage for PHWR. The expansion history and the year to be full are summarized in the Table 4 based on the estimated annual spent fuel discharge rate.

TABLE 4. EXPANSION HISTORY AND THE YEAR TO BE FULL

Name of Site	Expansion Capacity (t)	Method	Year to be full
Kori	361	High Density Rack	2006
Wolsung	609 (Completed) 812 (Constructing)	Dry Storage	2006
Yeongkwang	344	High Density Rack	2006
Uljin	443	High Density Rack	2007
Total	6,672	-	-

As can be seen from Table 4, expansion for spent fuel storage would be the only temporary alternatives, far from accommodating the whole spent fuels discharging from the existing NPPs and the planning NPPs. Since 1980s, the national plan as a solution to cope with the shortage of the spent fuel storage capacity, was established to build a large interim storage facility. In practical, this facility can work as a central storage one to collect most of spent fuels discharged from all nuclear power plants. Besides, the interim storage facility can allow sufficient time to carry out the national strategy of spent fuel management and back-end fuel cycle, whether recycle or direct disposal.

As have been well known, site selection has been discouraged by a series of demonstration of local residents in several candidate sites since 1989, as encountered with the same situation in other countries. The last candidate site, named Guleop Island in the West Sea of the Korean Peninsula, was

close to have the support from the residents. However an active fault was found near the island and this candidate was canceled officially by the government.

#### 4. NEED OF A NEW APPROACH

Through the series of efforts, such as burnup extension, storage rack expansion and dry storage construction, Korea has delayed the point in time for the storage facilities to be full. Nevertheless, the spent fuel arisings from nuclear power plants in Korea are a tremendous problem. Unless some drastic measures are taken in time, it is feared that some of nuclear power plants might have to be shut down.

There are limited options to relieve this situation temporarily. The easiest way is to build a storage facility on site, when one can not find a permanent or interim AFR storage site in time. Through a preliminary study, possible storage techniques were compared and dry vault at site was found to be the best option because of better economics and flexibility to various situations. This vault construction is supposed to be started in 2003 to meet the year to be full.

As long as the non-proliferation can be guaranteed, the permanent solution for storage problem will be the spent fuel recycling. For Korea, within the context of securing transparency for non-proliferation, abroad reprocessing is another possible way to be considered to ease the storage problem. Since there is a long time lag between the spent fuel shipping date to foreign countries and the high level waste receiving date, approximately 8 to 10 years, the requirement for additional AR storage capacity would be greatly reduced. However whether this method could be used or not is not a pure technical problem rather it is political and economic ones.

#### 5. RESEARCH AND DEVELOPMENT ACTIVITIES

Direct re-fabrication is another way to fulfill the non-proliferation purpose and storage problem. The PWR-CANDU reactor mix strategy in Korea allows a unique way for spent fuel management. Taking advantage of the recognized fact that the content of residual fissile elements in spent PWR fuel is still sufficiently higher (about 1.5 %), almost double, than those of the new CANDU fuel, DUPIC (Direct Use of Spent PWR Fuels In CANDU) has been studied to secure a proper fuel cycle with the reuse of the spent PWR fuels to CANDU in a direct way. DUPIC employs the OREOX (Oxidation/Reduction of Oxide Fuel) process which is a kind of dry one for reconfiguring the enriched spent oxide fuels into the DUPIC fuel bundles. More details are described in the Appendix.

Related to the integrated spent fuel management, other important research activities have been carried out. Their main activities aims to back up the national spent fuel management strategy and policy. Important studies regarding spent fuel management are being carried out in the following areas such as 1) transportation cask, 2) long-term behavior of defected fuel in wet storage atmosphere, 3) transmutation of actinides and fission products, and 4) remote handling and maintenance, etc.

#### 6. CONCLUSIONS

The back-end of the fuel cycle has not been established in Korea. Both safe interim storage of spent fuel discharged from power plant and the reduction of the spent fuel arisings through improvement in uranium utilization through increased fuel burnup would be the realistic alternatives that Korea should take. This approach, however, is not so successful because of the lack of AFR interim storage facilities. Korea should select an interim storage site or find alternatives such as dry vault at site for short the term, abroad reprocessing for the mid-term and direct re-fabrication for the long term. Korea is eager to have an internationally accepted spent fuel management and back-end fuel cycle technology. "The sooner, the better."

**Appendix**  
**DUPIC CYCLE**

In this process, dense UO<sub>2</sub> pellets are broken down into a ceramic-grade powder by cyclic oxidation and reduction and resulting powder is pressed and sintered into CANDU-quality pellets that are assembled with CANDU bundles. The fissile content of around 1.5% in DUPIC fuels is well within the performance envelope of CANDU system. However, due to the fission products inherently left in the fuel by the DUPIC process, thereby depressing the core reactivity, the burnup level liable to be reached is not as high as in the TANDEM cycle. A value of 16 MWd/kgU is assumed to be achievable with DUPIC fuel, to compare with the 25 MWd/kgU foreseen for TANDEM fuel and the 7 MWd/kgU experienced with natural uranium fuel.

In accordance with a preliminary study, the DUPIC fuel cycle will allow to achieve the *optimized fuel economy and spent fuel management with respect to the mass of fissile material and the volume of waste produced per unit energy*. In terms of the reactor ratio, as a function of burnup of PWR fuel, the PWR/CANDU ratio corresponding to the best material balance for DUPIC would be 2.5:1 at the usual burnup rate of PWR fuel with 35 MWd/kgU. In case of high burnup of 50 MWd/kgU, the PWR/CANDU ratio of 4.8:1 was found to be the optimal ratio from the view point of material balance to the DUPIC fuel cycle.

However, due to some unknown factors that would affect the fuel cycle cost and resultant inevitable assumptions, it is in general very hard to assess the economics of a new fuel cycle like DUPIC. The preliminary study on the DUPIC fuel cycle which KAERI carried out showed that the DUPIC fuel cycle seems to be a competitive alternative to once-through fuel cycle.

Major benefits of the DUPIC fuel cycle are from the saving in natural uranium ore requirement, the substantial reduction in spent fuel and waste arisings, good safeguardability and so on. Especially, since the spent fuel material is directly and remotely re-fabricated into CANDU fuel in heavily shielded working cell without any separation of fissile elements, any access to and diversion from the DUPIC fuel process would be extremely difficult from the proliferation point of view.

There are still some uncertainties with the DUPIC fuel cycle technology. A series of questions in the DUPIC economics may rest on the remote fabrication technique of highly radioactive DUPIC fuel bundles, safeguards accountability, verification of fuel bundle performance, reactor physics and safety due to the perturbations from plutonium, the spent fuel disposal issues, the trapping and immobilization of semi-volatile radio-nuclides, and so on. Further studies should be carried out in this respect to obtain more realistic figures for the DUPIC fuel cycle.

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# CURRENT STATE OF SPENT FUEL MANAGEMENT IN THE RUSSIAN FEDERATION



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## Abstract

Twenty nine power units of nine nuclear power plants of total installed capacity 22 GW(e) are now in operation in the Russian Federation. They produce approximately 12% of electric power in the country. The annual spent fuel arising is about 790 tU. The spent fuel from VVER-440 and BN-600 is reprocessed at the RT-1 plant near Chelyabinsk. The VVER-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 because of its low fissile content. It is meant to be stored in intermediate storage facilities at the NPP sites and in a centralized storage facility during a period not less than 50 years and then to be disposed of in geological formations. State of the art of spent fuel reprocessing, storage and transportation is considered in the paper. Problems of nuclear fuel cycle back-end in Russia are taken into account.

## 1. STATUS OF SPENT FUEL MANAGEMENT

29 power units of 9 nuclear power plants are now in operation in the Russian Federation. The major reactor types are VVER-440 (6 units), VVER-1000 (7 units), RBMK-1000 (11 units), BN-600 (1 unit) and EGP (4 units) with total capacity about 22 GW. These nuclear power plants (NPPs) produce 12% of the total energy production. The annual spent fuel (SF) arising from Russian reactors amounts to 790 tU. In accordance with the scheduled commissioning of new power units, the SF arising will increase to >950t by 2000, and 1100 t by 2010.

At present Russia continues to realize the closed fuel cycle concept in the relation to the VVER and BN spent fuel. This includes SF reprocessing, U and partially Pu recycling in thermal and breeder reactors, and vitrified radwaste storage in ground-based storage facilities at reprocessing plants.

The VVER and BN spent fuel is cooled in AR pools for no less than 3 years. The spent fuel from VVER-440 and BN-600 is reprocessed at the reprocessing plant RT-1 near Chelyabinsk (enterprise MAYAK). VVER-1000 spent fuel is supposed to be reprocessed at the RT-2 plant which is under construction near Krasnoyarsk. A storage facility of 6000 t capacity has been built at the RT-2 plant site. At present it holds over 1700 t U of spent fuel.

The RBMK-1000 spent fuel is not reprocessed because of its low fissile content. An intermediate spent fuel storage is a necessary step of the fuel cycle. The RBMK SF is stored in AR and the interim storage facility at the NPP site for no less than 10 years. Wet storage remains and will be prevailing in the nearest years. RBMK spent fuel quantities present considerable difficulties. Dispatch of the spent fuel has not been carried out. The storage capacity of operating RBMK facilities (including that obtained with denser FA arrangement) will provide SF reception from NPPs up to 2005. The above situation points out at to the pressing problem of long-term storage.

TABLE 1. CAPACITY OF STORAGE FACILITIES AND SPENT FUEL ARISING FROM MAJOR TYPES OF POWER REACTORS

Type of reactor	VVER-440		VVER -1000				RBMK			
	At- reactor		At-reactor		Away-from-reactor		At-reactor		Away-from-reactor	
Storage facility type	Storage capacity	Arisings	Storage capacity	Arisings	Storage capacity	Arisings	Storage capacity	Arisings	Storage capacity	Arisings
Fuel assemblies, items	3900 (5200 <sup>*</sup> )	2670	2760	1047	900	104	31250 <sup>**</sup> )	23725	78750 <sup>**</sup> )	43540
Fuel quantities, tU	470 (630 <sup>x</sup> )	320	1200	460	400	46	3560	2700	8970	4960

\*) I and II units of NV NPP included

\*\*\*) Storage capacity with dense storage mode.

The status, as of 01.09.1997, of the spent fuel arisings from the major reactor types in the storage facilities are listed in Table 1.

At the previous meeting of the regular advisory group we informed about long-term storage technology versions considered in Russia. A decision was taken to construct a long-term centralized dry storage facility at the RT-2 plant site. The next technical proposals are considered:

- long-term storage of RBMK spent fuel in stainless steel canisters accommodated in massive concrete structures with residual heat removal by natural (air ) convection;
- placing sealed cans with fuel into metal tubes (SGN (France) design of CASCADE type).

Simultaneously dual-purpose metal-concrete casks are being developed. These can facilitate the intermediate dry storage of RBMK spent fuel at the NPP site.

The SF shipping from the NPP sites by railway remains the only transport means. TK-6, TK-10, TK-11, TK-13 are used. Many information about their design and practice is published.

A new generation of containers development has begun because of the end of service life of some types. The new ones will meet all modern requirements. A positive experience available is taken into account while developing the containers. The task is to ensure the maximum capacity for the cost lowering and to increase safety of transportation. At the same time new casks should allow their using at the existing and designed NPPs and nuclear industry facilities. As a rule the casks should be of dual purpose: for transportation and for storage.

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**Abstract**

Eskom, the South African utility, operates one of the largest electricity networks in the world. However, only 6% of the South African generating capacity is nuclear; the remainder is coal fired and hydroelectric. The nuclear component consists of the Koeberg Nuclear Power Plant, comprising two French supplied PWRs of 920 MWe each, situated approximately 45 kilometres from Cape Town. Construction started in 1976 and the two reactors reached criticality in 1984 and 1985 respectively. South Africa also has an Oak Ridge type research reactor, called SAFARI, operated by the South African Atomic Energy Corporation (AEC) at their Pelindaba site near Pretoria. This research reactor was commissioned in 1965, and has been in operation ever since.

South Africa has a National Radioactive Waste Disposal facility called Vaalputs, some 600 km north of Cape Town. The facility, operated by AEC, is presently licensed only for the disposal of low and intermediate radioactive level wastes. Vaalputs offers unique features as a potential interim spent fuel storage and final disposal site, such as favorable geology (granite), low seismicity, low population density, remoteness from industrial centres and arid conditions. Therefore, this site has been investigated by the AEC as a potential interim spent fuel storage site, but has not yet been licensed for this purpose. Hence, all spent fuel is currently stored on the two sites at Koeberg and Pelindaba respectively. The spent fuel storage pools at Koeberg have recently been enlarged to accommodate the lifetime spent fuel arisings of the plant. Since late 1997, the Safari spent fuel is stored in a pipe storage facility, constructed away from the reactor on the Pelindaba site.

**1. INTRODUCTION**

South Africa operates one of the largest electricity networks in the world. The national utility, Eskom, is the fifth largest in the world. However only 6% of the generating capacity is nuclear; most (92%) is fossil (coal) - fired, the remainder is hydroelectric and pumped storage generating plant.

The nuclear component is the Koeberg Nuclear Power Plant, consisting of two French supplied Pressurized Water Reactors (PWR) each of 920 MWe, situated approximately 45 kilometers from Cape Town. Construction started in 1976 and the two reactors reached criticality in 1984 and 1985 respectively. The plant, with a design life of 40 years, has been operating successfully ever since. Each reactor core is loaded with 157 fuel elements of the 17 x 17 type, 3.66m active fuel length. The fuel management strategy is 1/3 core refueling, 18 month cycles. The reload fuel enrichment was originally 3.25% which implies that periods of low power and stretch out or power coast down operation were required to achieve the 18 month cycles. Recently however the enrichment was increased to 3.9%.

South Africa also has an Oak Ridge type research reactor, called SAFARI, operated by the South African Atomic Energy Corporation (AEC) at their Pelindaba site near Pretoria. This research reactor was commissioned in 1965, and operated for much of its life at 5 MW. In the early 1990's the power level was increased to 20 MW, using MTR fuel produced locally by the AEC. The remaining life of the SAFARI research reactor is at least another 16 years. No decisions have been taken regarding another or a replacement research reactor. Nevertheless it is feasible that at some time in the future another or a replacement research reactor may be constructed.

South Africa has a National Radioactive Waste Disposal facility some 600 km north of Cape Town (100 km south east of Springbok). The facility, called Vaalputs and operated by the AEC, covers an area of about 10 000 hectare measuring 16,5 km from east to west and 6,5 km from north to south. Approximately 100 ha will be occupied as a waste disposal site, large enough for waste and decommissioned hardware from 3 nuclear power plants the size of Koeberg. This facility has generally received international recognition for its ideal location and very suitable geological conditions. Vaalputs is licensed with the South African regulatory authority - the Council for Nuclear safety (CNS) - only for the disposal of low and intermediate radioactive level wastes. No reprocessing facility exists in South Africa, and neither Eskom nor the AEC have plans to reprocess the spent fuel from Koeberg and SAFARI. Hence all spent fuel is currently stored on the two sites Koeberg, and Pelindaba, respectively.

Vaalputs offers unique features as a potential interim spent fuel storage and final disposal site, such as favorable geology (granite), low seismicity, low population density, remoteness from industrial centers and arid conditions. It has been investigated by the AEC as a potential interim spent fuel storage site, using various storage concepts. However none of these concepts have been licensed for Vaalputs.

South Africa has at present a surplus of generating capacity, which will continue into the first half of the next decade. Hence no new nuclear power plants of the size and type of Koeberg is likely to be constructed in the next 10 years. Thereafter economics, environmental considerations and public perceptions will influence the type of generating plant which will be constructed. Eskom's investigations into future electricity generating plant technologies includes conceptual studies on high temperature gas cooled reactors. The possibility does therefore exist that at some time in the future the nuclear power capacity in South Africa may increase.

The regulatory authority - the Council for Nuclear Safety (CNS) - is responsible for the licensing of all nuclear installations in South Africa. This responsibility also includes the disposal of radioactive waste and the storage of spent fuel. The CNS was established by an Act of Parliament and performs its functions independently of the operators of the nuclear installations and the generators of radioactive waste.

A number of initiatives have taken place over the last few years, for example the development of energy, environmental and radioactive waste management draft policies, a review of nuclear legislation, a review of the nuclear fuel cycle activities in South Africa and progress with specific spent fuel interim storage options. These initiatives have, and will continue in the next few years to shape the spent fuel management programme in South Africa. The current status of spent fuel, these initiatives and the likely direction for the future are discussed in this paper.

## 2. CURRENT STATUS

### 2.1. Koeberg nuclear power plant

When Koeberg was designed in the 1970's, the spent fuel pool of each unit, based on the French reference plant, had a capacity to store 382 fuel elements. These spent fuel pools were adequately sized for temporary storage for a plant which could transfer its spent fuel to a reprocessing facility after an appropriate cool down period. However, reprocessing was not an option open to Eskom at the time of commercial operation in 1984. So the pools were each re-racked in 1987, with high-density racks which increased their capacity to store to 728 fuel elements. This allowed storing of spent fuel for 11 to 16 years of operation with standard fuel, depending on the load factor (and outage duration).

Some years ago a decision was to purchase a number of dual-purpose transport and storage casks as the medium for interim storage (up to 50 years). This decision was based on the technology existing at the time (storage casks, dual-purpose transport and storage casks, storage vaults and

storage bunker systems - the latter two would still require transport casks) and a need to keep capital expenditure to a minimum for a number of years. At that time it was believed that another re-racking of the spent fuel pools would not be possible, due mainly to constraints associated with the spent fuel pool civil structures. The four casks originally ordered (in March 1992) were delivered to Koeberg in April 1996. The four casks were intended to be only the first step towards an interim spent fuel storage facility, and would have taken Koeberg only into the beginning of the next century. At some stage before that time it would have been necessary to review again the interim spent fuel storage technologies, and determine the cost effectiveness of continuing with storage in casks or to move towards some other form of storage.

During 1996 the final report of a study group, referred to as the "Nuclear Fuel Cycle Initiative (NFCI)" was produced. The objective of this study was to investigate and formulate appropriate policy options for the nuclear fuel cycle activities in South Africa. With respect to spent fuel, the utility Eskom had already indicated that it favored interim storage on the grounds of technological and economic considerations. The NFCI report defined four interim storage options for the Koeberg spent fuel, in terms of combinations of storage in the Koeberg spent fuel pools, away-from-reactor storage on the Koeberg site, and storage on the Vaalputs site.

A review by Eskom of the available technologies led to the decision to re-rack again the spent fuel pools at Koeberg with ultra-high density racks by the end of 1998. This decision was made possible by improved civil structure modeling leading to a revised position on the ability of the Koeberg spent fuel pool civil structures to carry an increased load, under normal and seismic conditions. Re-racking of the pools was determined to be economically the most favorable option. It also has the advantage of retaining the spent fuel from the normal operational life of Koeberg on the site, and hence avoids the need, cost and logistical problems associated with the transport of fuel to another interim storage facility, whether at Koeberg or at Vaalputs.

The four dual-purpose transport and storage casks already purchased will therefore not be used for interim storage purposes in the short-term, but may prove invaluable during the re-racking of the existing spent fuel pools, and are available as a contingency storage facility if the need should arise due to any unforeseen circumstances. At any time, after an initial period (normally 5 - 10 years), the spent fuel could be retrieved for reprocessing if required. After the 50 year interim storage period, the spent fuel could either be sent for reprocessing or could be disposed of permanently (if a licensed facility exists). Another possibility is that the spent fuel could be stored for a further interim period; however there is currently no international experience with this option.

The project to re-rack the spent fuel pools at Koeberg with ultra high density racks is proceeding. The contract has been awarded, and the design and safety studies are being performed. Licensing of the design and process to be followed during the re-racking with the Council for Nuclear Safety, is in progress. The re-racking project provides Koeberg with "interim spent fuel storage" for approximately 30 - 40 years, depending on the fuel management strategy, load factors and outage duration. Thereafter a licensed final disposal facility for the spent fuel (or the high level radioactive waste if reprocessing occurs) will be required.

The existing racks at Koeberg currently hold 800 spent fuel elements, with another 52 due to be placed in storage during a refueling outage in September/October 1997. Based on the present and likely future fuel management strategies it is estimated that approximately 3000 spent fuel elements will have been produced over Koeberg's operational lifetime.

## **2.2. SAFARI research reactor**

The SAFARI research reactor, situated at the Atomic Energy Corporation's (AEC) Pelindaba site near Pretoria is a US supplied ORR type reactor. Commissioned in 1965, the SAFARI reactor has been operating at a power level of approximately 5 MW for most of its life being primarily used for activation analysis and isotope production purposes, but also for some materials testing and physics

research. In 1994 the AEC embarked on a commercial molybdenum-99 isotope production programme, which led to a gradual increase in reactor power output currently at the 20 MW level.

SAFARI fuel was originally obtained from the US and limited quantities of spent fuel based on the original US fuel supplies have been returned to the US. From the mid 1980's onwards, the AEC has been producing its own MTR fuel. Spent fuel has been accumulating in the reactor storage pool since the start of reactor operation and although re-racked in 1990 the storage pool is presently almost filled to capacity. Consequently, various options for SAFARI spent fuel storage have been investigated by the AEC with the emphasis on dry storage in casks versus dry storage in a pipe storage facility. As the latter option appeared to be the more attractive proposition from a technical and economic point of view, it was decided to construct an away-from-reactor pipe storage facility on the Pelindaba site, called the Thabana Pipe Store, to accommodate all SAFARI's spent fuel arisings.

The geology of the selected pipe storage site consists of alternate layers of shale and quartzite of the Transvaal Sequence. From several boreholes drilled down to 100 m it was concluded that no large scale disturbance or concealed structures were present in this area. The probability of a large seismic occurrence at the site was considered to be acceptable. The static water level is approximately 60 m below the pipe store site, leaving at least 40 m of unsaturated rock between the bottom of the pipe storage holes and the water level.

The pipe storage building measures 21 by 11 metros and consists of two adjacent vertical storage sections; i.e. the spent fuel storage section and the hot cell waste storage section situated on either side of a central loading area inside the building. In the spent fuel storage section, thirty stainless steel storage pipes, each 125 mm in diameter and approximately 17 metros long are suspended in 20 meters deep boreholes below a 1 meter thick concrete floor slab. With each storage pipe capable of accommodating 20 fuel elements, the spent fuel section of the storage facility thus has a capacity of 600 elements. The hot cell waste section has a capacity of 840 canisters with a storage volume of 8 each.

Residual heat dissipation in the pipe store is achieved by means of natural convection inside the fuel pipes which are back-filled with an argon helium gas mixture. Heat is dissipated into the surrounding soil at a rate sufficient to ensure that the spent fuel element temperature does not rise above 200°C. The heat output per element 3 years after discharge is approximately 15 Watt.

The pipe store facility construction has been completed and an operating license conditionally granted. The facility is expected to be in use by September 1997. The spent fuel transfer operation from the reactor to the pipe store involves 280 elements and is scheduled to take approximately 10 months. Spent fuel is transferred by means of a bottom loading transfer cask into which a single spent fuel element is hoisted. The fuel element is maintained in a suspended condition in the cask using a cam lock at the top end of the cask. When inserted into the storage pipes the elements are lowered down the length of the vertical pipe by means of steel rods.

At the current rate of spent fuel generation of 60 spent fuel elements per year, 900 spent fuel elements would be produced over the next 15 remaining years of the reactor's life. Provision is made to extend the building design for additional storage area if required in future. No decision has yet been taken regarding the final disposal of SAFARI spent fuel.

### 3. FUTURE FRAMEWORK

Both the Koeberg and the SAFARI nuclear installations have been under IAEA safeguards since their commissioning. In July 1991, South Africa acceded to the Nuclear Non-Proliferation Treaty (NPT), and soon thereafter entered into a full-scope safeguards agreement with the IAEA. South Africa is a signatory to the International Nuclear Safety Convention, and has actively participated in the drafting of the Joint Convention of the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management.

South Africa is also currently establishing a national radioactive waste management policy. Although provision is made in the existing Nuclear Energy Act for control over the disposal of radioactive waste and the storage of spent fuel, it has generally been recognized in South Africa that a national policy on the management of radioactive waste is required. Hence the CNS was directed by the Government to facilitate a consultative process with a view to establishing such a policy. The process commenced in 1996 and is still in progress.

Environmental and Integrated Pollution Control & Waste Management policies are also under development. The integration of the intent of these different initiatives will ensure that the management of radioactive waste, and the storage of spent fuel are adequately addressed in future revised legislation.

All of these policy formulation processes will naturally result in revisions to existing nuclear-related legislation and the development of new legislation. Indeed the process of revising the Nuclear Energy Act of 1993, and the development of a Nuclear Safety Act to ensure the independence, in practice as well as in legislation, of the CNS, has commenced. An energy policy is also in preparation.

The Nuclear Fuel Cycle Initiative (NFCI) referred to above, also recommended that investigations into the suitability of Vaalputs for the final disposal of high level waste should continue. Presently it has not yet been established that Vaalputs meets the criteria for a final disposal site for spent fuel, although on the basis of initial bore hole results it is recognized that the probability of Vaalputs, or the surrounding regions, meeting the criteria is favorable. Hence the NFCI report contained a recommendation that the development of a concept for deep geological disposal and the required appraisal of geological structures and other environmental aspects in the Vaalputs region should be undertaken.

As indicated above, the re-racking project will provide Koeberg with "interim spent fuel storage" for approximately 30 - 40 years, depending on the fuel management strategy, load factors and outage duration. Similarly the interim pipe storage facility at Pelindaba will meet the needs of the SAFARI spent fuel. Hence sufficient time exists to develop a final disposal facility. Nevertheless, international experience to date suggests that the time required from preliminary investigation, facility design and licensing, obtaining public acceptance, through to initial operation of a deep repository for the final disposal of spent nuclear fuel or high level radioactive waste is in the order of 30 to 40 years. This implies that preliminary investigations into a final disposal facility for south Africa's spent fuel should commence within the next one to two years.

#### 4. CONCLUSION

It is evident from the above that South Africa is involved in a complex process of developing policy and revising legislation regarding nuclear activities, and specifically spent fuel and radioactive waste management. Although interim spent fuel storage options have been chosen and are being implemented, the need exists to commence the appropriate processes which will lead to the establishment of a final disposal facility.

However it is not South Africa's desire or intention to "re-invent the wheel". The experience gained in other countries needs to be taken into consideration. Appropriate international co-operation, either directly between countries or under the auspices of organizations such as the IAEA, should be encouraged to ensure the sharing of experiences and the resolution of the spent fuel management issue.

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### **Abstract**

About 50 % of the electricity in Sweden is generated by means of nuclear power from 12 LWR reactors located at four sites and with a total capacity of 10 000 MW. The four utilities have jointly created SKB, the Swedish Nuclear Fuel and Waste Management Company, which has been given the mandate to manage the spent fuel and radioactive waste from its origin at the reactors to the final disposal.

SKB has developed a system for the safe handling of all kinds of radioactive waste from the Swedish nuclear power plants. The keystones now in operation of this system are a transport system, a central interim storage facility for spent nuclear fuel (CLAB), a final repository for short-lived, low and intermediate level waste (SFR). The remaining system components being planned are an encapsulation plant for spent nuclear fuel and a deep repository for encapsulated spent fuel and other long-lived radioactive wastes.

## **1. INTRODUCTION**

About 50 % of the electricity in Sweden is generated by means of nuclear power from 12 reactors located at four sites 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. According to a decision by the Swedish Parliament in 1980 all reactors were to be phased out by the year 2010 at the latest. Until then about 8000 t of fuel would have been used and would have to be taken care of as spent nuclear fuel. Early 1997, however, three political parties in the Parliament reached an agreement on shutdown of one reactor in 1998 and another one in 2001, the latter provided that there is replacement power available. The year 2010, the year beyond which no reactor should be allowed to operate however is not an issue any longer. The Parliament voted in favor of the proposal in spring. The two reactors in question are those at Barsebäck in southern Sweden.

According to the Swedish spent fuel management programme the fuel from the Swedish reactors shall be taken care of within the country and be disposed of at about 500 m depth in the bedrock.

After unloading from the reactor core and a cooling period at the reactors the spent fuel is transported to a central interim storage facility where the fuel will remain for 30 - 40 years. During this period the radioactivity and the residual heat of the fuel will decay by about a factor of ten, thus making further handling and the final disposal simpler. The storage period will also provide time and flexibility for the elaboration of the details of these steps.

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, which has been given the mandate to safely manage the spent fuel and radioactive waste from its origin at the reactors to the 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.

Today the total irradiated fuel quantity amounts to about 4500 t including the fuel in the reactor cores.

## 2. SYSTEMS AND FACILITIES

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 (see Fig. 1) are:

- 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 system components now being planned are:

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

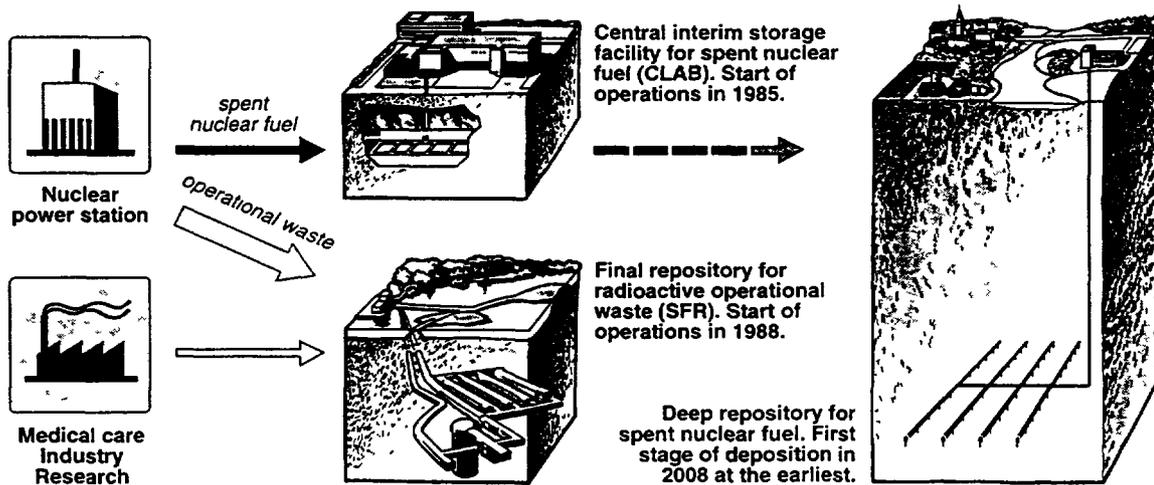
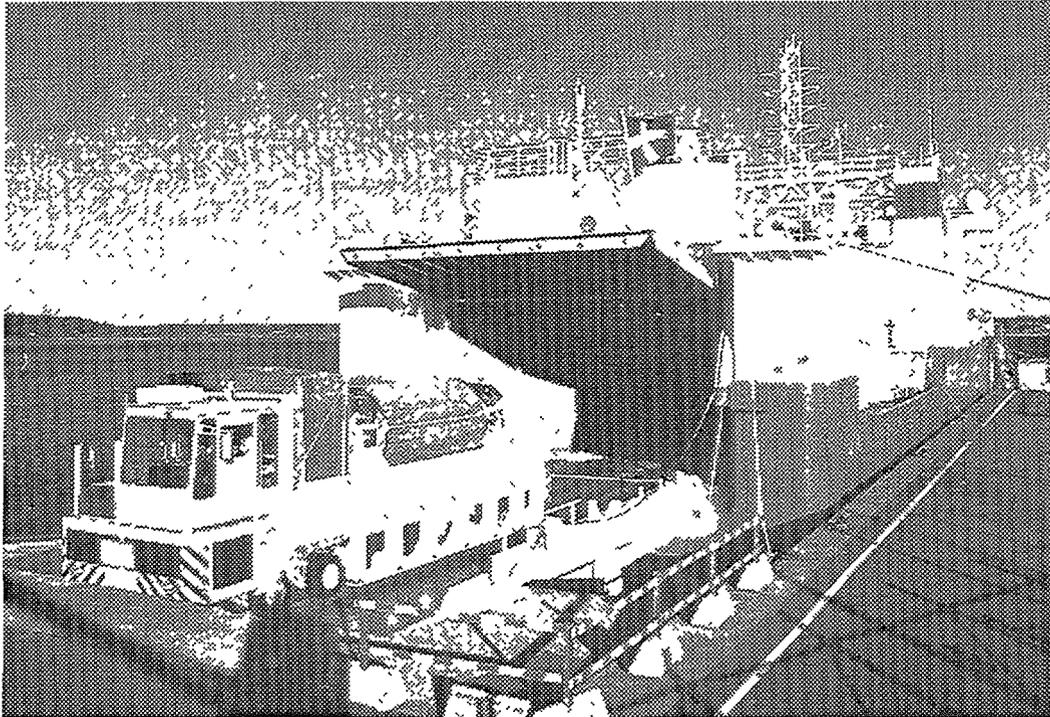


Figure 1. The Swedish system for radioactive waste management.

## 3. THE TRANSPORT SYSTEM

As all the nuclear power plants and CLAB are located on the coast and have their own harbors SKB has developed a sea transport system. This has many advantages, such as a high load capacity and low interference with other traffic. The system comprises a purpose built ship, the M/S Sigyn, 10 transport casks for spent fuel, 2 casks for spent core components and 5 terminal vehicles. The latter are used for the land transport from the reactor to the harbor and from the harbor to CLAB (Fig. 2).

M/S Sigyn is a roll on/roll off - lift on/lift off ship built for transports of radioactive waste. She has a dead-weight of 2000 t and can carry up to 10 transport casks for spent fuel. She has been designed with the most restrictive IMO rules concerning floatability after damage, similar to those for ships carrying chemicals in bulk. The ice breaking capability of the ship increases the availability in winter conditions. Another important safety factor is the most modern navigation equipment installed on board. However the ultimate safety of the transports depends on the 80 t heavy transport casks designed according to the IAEA regulations.



*Figure 2. A fuel transport cask is being loaded on board the M/S Sigyn*

The dry nitrogen filled cask cavity can accommodate 17 BWR or 7 PWR assemblies corresponding to 3 t of fuel. The cask is cooled by natural air convection around the 40 000 cooling fins on the cask outer surface. The cooling capacity allows fuel with a burnup of 55 000 MWd/tU and 9 months cooling time to be transported.

The dose rate criteria given by the transport regulations are, however, more limiting than the thermal ones, mainly due to the buildup of neutron emitters at high burnup. Transport of fuel assemblies with a burnup of 43 000 MWd/tU requires a cooling time of minimum 20 months. With higher burnup the necessary cooling time increases rapidly. Therefore a modification of two of the casks was performed in 1995 implying an increase of the thickness of the neutron absorbing resin on the cask surface at the cost of somewhat reduced thermal performance of the casks.

### **3.1. Operating experiences**

In mid 1997 more than 900 fuel transport casks have been shipped to CLAB, corresponding to about 2700 t of uranium. In parallel around 80 casks with highly active core components, e.g. control rods, have been received at the facility. Typically 75 fuel casks are transported annually, corresponding to about 225 t of fuel.

The performance and availability of the system has been excellent. The fast and efficient handling and lashing operations for the casks on the ship have resulted in crew doses close to the those coming from the background radiation.

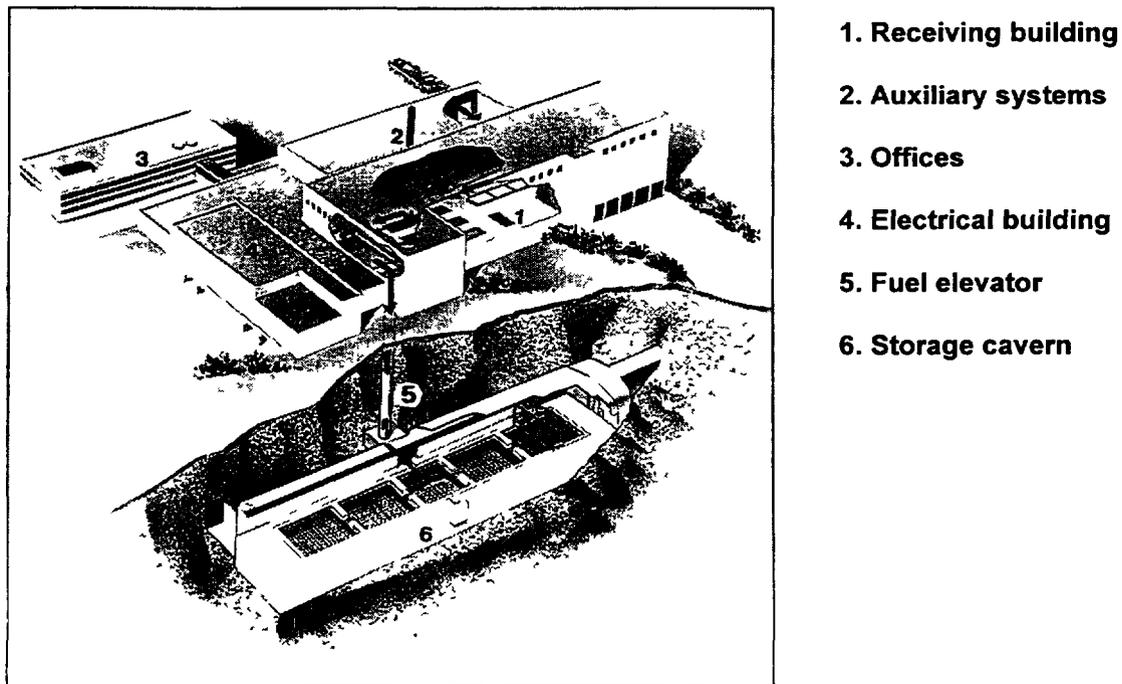
Another factor contributing to the good results is the detailed cask maintenance programme according to which a cask is brought in for maintenance after 15 transport cycles. Each transport cask has already gone through about 10 such overhauls. A more extensive maintenance is made after 60 cycles. The transport system is also used for transport of low and intermediate level waste to the SFR facility.

During the last 8 summers M/S Sigyn has been used as a floating exhibition for information to the public about the Swedish nuclear waste management system. In total almost 500 000 persons have visited the exhibition corresponding to about 6% of Sweden's population.

#### 4. CENTRAL INTERIM STORAGE FOR SPENT FUEL, 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. Operation started in 1985, and in June 1997 some 2700 t of fuel in about 900 casks had been received in addition to about 80 casks with activated core components, e.g. control rods.

CLAB comprises two principal parts, one above ground and one under ground. The main complex above ground is the receiving building, where the transport casks are received, prepared and unloaded. The unloading is performed under water. The storage section is located in a rock cavern, the roof of which is 25-30 m below the ground surface. Auxiliary systems, such as water cooling and purification, and electric power and control are located in buildings wall to wall to the receiving building (Fig. 3).



*Figure 3 CLAB, Central Interim Storage Facility for Spent Fuel*

The receiving building has three receiving pool lines, two of which comprise two pools specially equipped for the standard TN17/2 transport casks. The third line can receive non-standard casks. Prior to unloading, the cask is provided with a metal skirt for protection against outside contamination and damage. The skirt also makes it possible to apply very efficient water-cooling via the 40 000 cooling fins on the cask outside. After tests for fuel leakage and other preparatory measures including internal and external cooling the cask is transferred to one of the receiving pool lines, where the lid is removed so that the fuel assemblies can be lifted out one by one. Unloading - and all subsequent handling of the fuel assemblies - is performed under water with specialized

handling machines. The two pools of each of the lines for standard casks are arranged in such a way that during unloading the outer surface of the cask skirt is in contact with non-contaminated water, while the fuel assemblies and the internals of the cask are in contact with contaminated water. After unloading from the cask the fuel assemblies are transferred directly to a storage canister, which subsequently serves as the handling unit in the facility. This greatly reduces the number of necessary handling operations.

The storage underground section consists of 4 storage pools in a 120 m long rock cavern. Each storage pool contains about 3000 m<sup>3</sup> of water and can hold 300 storage canisters. In the storage pools the canisters serve as storage racks. A fifth pool stands as a reserve in case of problems with one of the storage pools. The canisters are brought down to the rock cavern by means of the fuel elevator. The 40 m high elevator shaft itself is not water-filled but the canister is placed in the water filled elevator cage during the transfer.

The storage canisters are designed to maintain an adequate margin against criticality under normal and accident conditions. The original storage canisters can hold 16 BWR or 5 PWR fuel assemblies and have an internal structure made of normal stainless steel. As of 1992 the canister capacity has been increased to 25 BWR and 9 PWR fuel assemblies respectively, thereby enlarging the total storage capacity of CLAB from 3000 t of uranium to 5000 t within the existing space. The new canisters are successively being introduced in the facility. Two alternatives were considered to achieve the denser packing without exceeding the required reactivity margin of 5 percent units:

- the use of canisters with internal neutron absorbers, i.e. boronated steel or
- credit for burnup.

After a thorough analysis the boron option was chosen. This has the advantage that the control of the fuel assemblies at reception can be made much easier as it allows fresh fuel elements to be stored.

In the case with credit for burn up rather complicated verification measurements on the fuel assemblies at reception would have been necessary. It was further found that unexpectedly great uncertainties were coupled to the BWR fuel assemblies due to their axial burnup profile and void history. In order to cover all possible cases it would therefore have been necessary to apply a great reactivity penalty in the licensing calculations. Under these conditions it would not have been possible to store many of the already existing assemblies in the new canisters.

Other methods to increase the storage capacity were studied, e.g. new fixed storage racks, two tier storage and rod consolidation. These were however not feasible due to higher costs and/or technical reasons.

The present capacity of CLAB of about 5000 t of uranium covers the needs until around year 2004. CLAB must therefore be expanded by adding storage pools in a new rock cavern close to and parallel to the existing one. This was anticipated already during the construction of the facility and certain preparations were made to facilitate the building and the connection of the new cavern to the existing fuel containing pools. In June 1997 SKB submitted the application for the expansion to the government. According to current planning the construction of the second cavern will start in the second half of 1998.

## 4.1. Operating experiences

The performance of the plant has been excellent and due to improvements the operating costs have successively been reduced considerably. Some factors contributing to the reduction are improved operating procedures that has made it possible to unload casks in one shift instead of two and increased sharing of staff with the co-located reactors. A great saving was made possible by the installation of a heat recovery system allowing the residual power from the fuel in the storage pools to be used to heat the entire plant.

The residual power from the stored fuel to the water is regularly measured and compared to what should be expected from calculations. It has been observed that the measured value (with corrections for different factors) is lower than the calculated value. In the beginning a difference of as much as 40-50% between the two methods was observed. After correction for the power history of the individual fuel assemblies the difference has shrunk to about 20% which still is not satisfactory. In parallel calorimetric measurements are made in CLAB on individual BWR and PWR assemblies with different burn-ups and cooling times up to 15 years. In these cases the measured and calculated values correspond much better. The measurements and calculations continue and the definitive results will be reported in due time.

The activity release from the fuel has been much lower than anticipated. During the design phase, based on foreign experience, a high crud release was expected when the fuel is exposed to a certain thermal shock when the dry cask is filled with water. The experience at CLAB is, however, much better, which may to a great extent be attributed the small crud amount on the fuel which in turn is due to the good water chemistry and the materials used in the Swedish reactors. The activity release from the fuel in the pools is also low and dominated by ionic Co-60. This release is sensitive to temperature but up to now not to the amount of fuel in the pools. The low activity release in combination with an optimized management of the used filter resins has reduced the number of waste packages emanating from CLAB. The actual volumes are at least ten times lower than expected. Also the releases to the environment via water or air have been very low, less than a factor 1000 below the permissible limits.

The annual collective radiation dose to the staff and contractors has been between 50 and 135 mmanSv over the years, which is about 20 - 50 % of the dose originally calculated in the final safety report. In 1996 the collective dose was 50 mmanSv and the projection for 1997 is well below 100 mmanSv. The main part of the dose comes from plant refurbishment work and the handling, preparation and maintenance of the transport casks.

Normally the fuel from the Swedish reactors is transported to CLAB in the standard TN17/2 cask as mentioned above. Occasionally some non-standard casks have been received, e.g. with fuel from the old Swedish PHWR reactor at Ågesta and old MOX fuel from some German nuclear power plants. The latter as a result of an exchange of a small amount of fuel between Sweden and Germany. Also encapsulated residues from Post Irradiation Examination of LWR fuel at the Studsvik research centre are being shipped to CLAB. All these non-standard casks have been received in the third unloading pool without any problems.

## 5. THE DISPOSAL CANISTER AND THE ENCAPSULATION PLANT

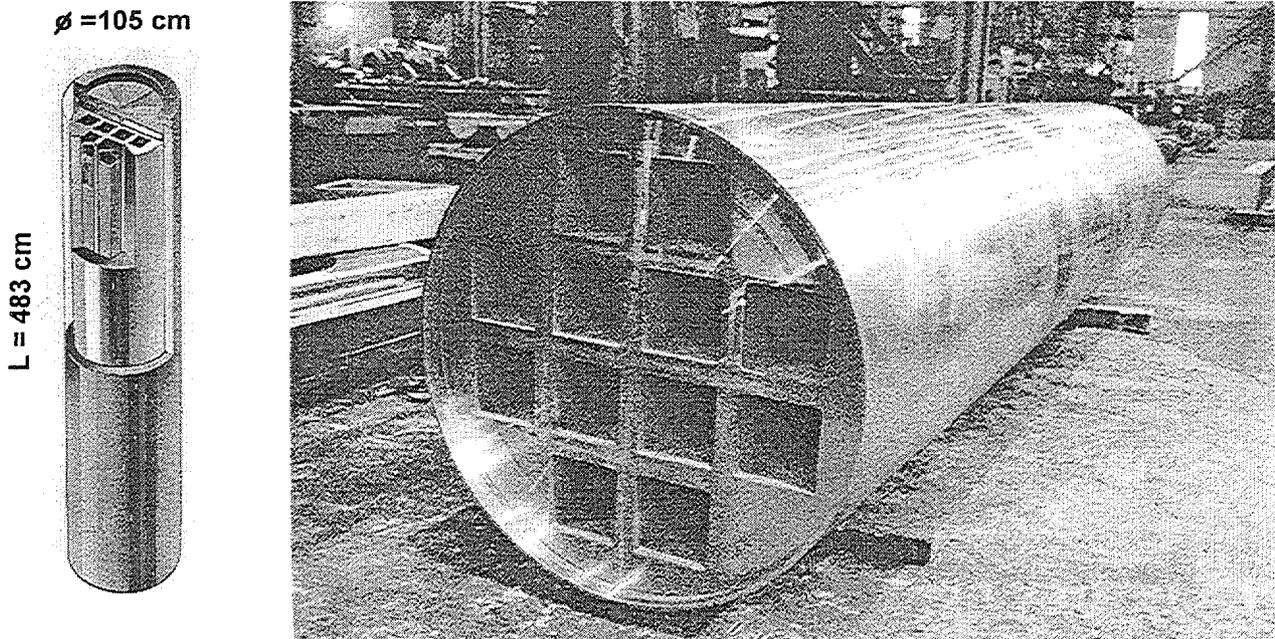
The spent fuel will remain in CLAB for 30-40 years prior to encapsulation in a corrosion resistant canister and final disposal.

### 5.1. The disposal canister

The function of the canister is to provide adequate enclosure and radiation shielding of the fuel during handling before disposal and to provide an absolute barrier against radioactivity release for

very long time thereafter. To remain tight the canister must sustain the mechanical and chemical environment in the repository (Figs. 4, 5).

The canister material most thoroughly studied in Sweden is copper which is almost unaffected in the reducing ground water found in the granite bedrock. The expected corrosion lifetime for a thick copper canister in such environment is millions of years and would thus isolate the fuel even beyond the lifetime of Pu-239. The canister consists of an inner steel body made of cast metal with channels for the fuel assemblies and a thick outer copper shell. The metal insert provides the necessary mechanical strength.



*Figure 4. Disposal canister and cast canister insert.*



*Figure 5. Full size test canister.*

The canister will contain 12 BWR or 4 PWR fuel assemblies. The amount of fuel that can be loaded into the canister is limited by the maximum permissible temperature of the canister surface after disposal. This means that there is no advantage in consolidating the fuel assemblies. The outer diameter of the canister is 105 cm, the height 483 cm and the total weight approximately 25 t. For handling and transport an extra radiation shielding will be needed.

The filled canister must have a sufficient margin to criticality. Under normal conditions when the canister is dry this is not a problem. For certain accident scenarios in the repository or the encapsulation plant, however, it can be postulated that the canister will be water filled. If all the fuel assemblies in the canister are assumed to be fresh, a critical configuration may exist according to preliminary calculations. It will probably be necessary to take credit for the burnup of the fuel, and/or to control the reactivity by other means. As part of the checking of the fuel assemblies before placement in the canister it is therefore expected that a gamma measurement will be made. This measurement may also be utilized for the determination of the residual power and safeguards verification that may be necessary before the closing of the canister.

## **5.2. The encapsulation plant**

The encapsulation plant is planned to be built adjacent to CLAB as a direct extension of the facility. The plant will incorporate the following main functions:

- Reception of storage canisters from CLAB via the existing fuel elevator;
- Selection of assemblies for encapsulation. Measurement;
- Filling of the disposal canister with fuel;
- Closing of the inner metal canister;
- Welding of the lid to the copper canister and non-destructive testing of the weld;
- Decontamination of filled canisters;
- Loading of canisters in transport casks for transport to the repository;
- Buffer storage for filled canisters.

In addition to the handling and process equipment necessary for these steps, auxiliary-, service- and control systems are required as well as facilities for the staff. Great advantage can be achieved by utilizing the existing corresponding systems etc. in CLAB.

A flexibility will be built in into the plant as a preparation for a later treatment of activated core components that are going to be disposed of in a deep repository as well.

A feasibility study of the plant was performed in 1993/94 and in June 1994 BNFL Ltd., England, was selected by SKB as main contractor for the Basic Design of the encapsulation process. In parallel ABB Atom, Sweden, was contracted for the service and auxiliary systems and service areas. The Basic Design was completed in 1996 and will be the base for the Preliminary Safety Report. The licensing application is planned to be submitted to the authorities in 1999 at the earliest. The expected construction cost for the plant is around 2 billion SEK (approximately 250 million US\$).

A crucial function in the encapsulation plant is the welding of the lid of the copper canister. This must be done remotely with a high accuracy, and in such a way that the result can afterwards be checked by non destructive testing.

The welding method preferred at present is electron beam welding at reduced atmospheric pressure. The development work has been going on for many years and by mid 1997 four full size test canisters have been fabricated.

The encapsulation is planned to be performed at a rate of one canister per working day. In order to be able to manufacture canisters on an industrial scale to meet this requirement SKB is building a canister laboratory for further development of final seal welding and non-destructive testing. The

laboratory is located in the harbor of the town of Oskarshamn near the CLAB facility and the welding tests will start in spring 1998.

## 6. FINAL DISPOSAL IN A REPOSITORY

The safety of the repository is based on a "defense in depth"- concept in three levels. The first level is isolation of the radionuclides which is accomplished by the long-lived corrosion resistant canister. The second level is delay of the transfer of the radionuclides to the biosphere in case the isolation is broken. This delay is achieved by a very slow dissolution of the fuel and sorption and slow transport in the near and far field around the repository. The dispersion in the biosphere can be seen as the third level of the system. Based on this approach, the Swedish system for final disposal, according to current plans, has the following features: encapsulation of the spent fuel in the long-lived, corrosion resistant canister described in section 5 above, and disposal in the Swedish bedrock.

The disposal will be made at about 500 m depth. The canisters are deposited in holes drilled from the floors of drifts at a center to center distance of about 6 m. In the holes the canisters are surrounded by highly compacted bentonite which acts as a mechanical and chemical buffer material and prevents direct contact between flowing water and the canister. The tunnels and shafts are backfilled with a mixture of sand and bentonite (Fig. 6).

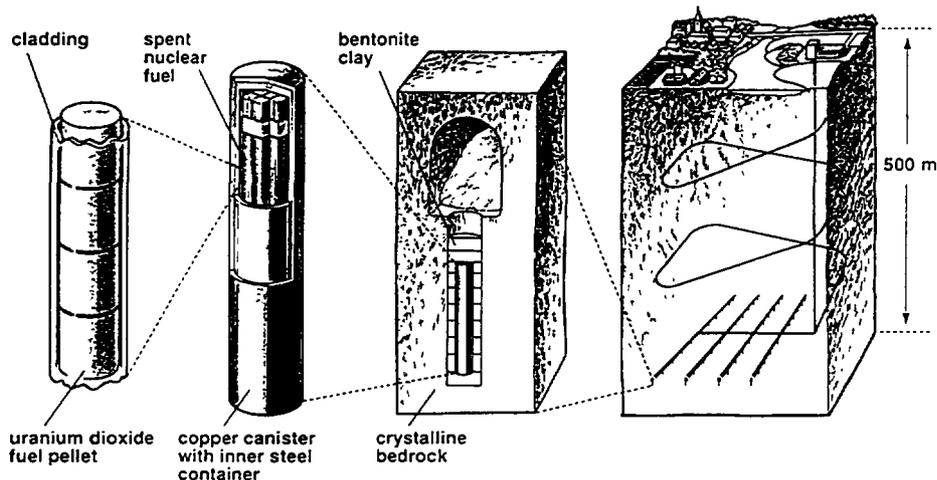


Figure 6. The canister in the deep repository.

The siting of a deep repository is politically sensitive in Sweden as in many other countries. From a technical point of view investigations have shown that there are many areas in Sweden where suitable geological conditions exist for a repository. Other factors such as transports, infrastructure, employment situation and political aspects will also be important for the siting.

To facilitate the siting and the public acceptance the disposal will be made stepwise. The first step will be the building of the deep repository for a limited amount of spent fuel as a demonstration. When the demonstration deposition has been completed, the results will be evaluated before a decision is made whether or not to expand the facility to accommodate the total waste amount. This plan also makes it possible to consider whether the deposited fuel should be retrieved for alternative treatment. This procedure will make it possible to demonstrate the siting, licensing, design and construction, handling of the canisters and operation of the facility. For obvious reasons the long-term safety of the repository cannot be demonstrated. This must always be based on a technical-scientific assessment.

The start of the demonstration deposition could at the earliest be in the latter part of the next decade. Studies of the local conditions are in progress in some areas in co-operation with the local authorities. If the conditions are found suitable detailed site investigations will then be made at one site.



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**Abstract**

Nuclear generating capacity in the UK is static with no units currently under construction. All the Magnox reactors previously belonging to Nuclear Electric plc and Scottish Nuclear Limited have been retained in a new publicly-owned company, Magnox Electric plc, which is currently planned to be merged with British Nuclear Fuels (BNFL). The AGRs and the UK's only PWR, Sizewell B, are operated by Nuclear Electric Limited (NE) and Scottish Nuclear Limited (SN) who are subsidiaries of British Energy plc (BE) which was privatised in July 1996. Prompt reprocessing of all Magnox fuel will continue. NE has recently signed a contract covering the lifetime arisings of AGR fuel which allows for both reprocessing and long term storage as required. Taken with SN's contracts signed in 1995 this means that all AGR spent fuel will now be sent to Sellafield for reprocessing or storage. Spent PWR fuel will continue to be stored at the reactor site.

**1. BACKGROUND AND GENERAL ISSUES**

The UK's nuclear generating capacity comprises some 8400 MW AGR and one 1200 MW PWR operated by British Energy's (BE) subsidiaries Nuclear Electric Limited (NE) and Scottish Nuclear Limited (SN), and 2950 MW Magnox operated by Magnox Electric plc (ME), together with 400 MW Magnox operated by British Nuclear Fuels plc (BNFL). No new nuclear capacity is currently under construction.

The main events that have taken place since the 1995 status report to the Advisory Group are as follows:

- The Thorp plant at Sellafield has finished active commissioning and a licence for full commercial operation was granted in August 1997 following a period of public consultation. Currently, over 800 tU of spent fuel has been reprocessed in the plant.
- Ownership of the Magnox reactors which previously belonged to Nuclear Electric plc and SN has been retained within a new publicly-owned company ME, which will eventually be merged with BNFL. NE and SN, who own the AGR and PWR stations, are subsidiaries of a holding company, BE, which was privatised in July 1996.
- In June 1997, BNFL and NE signed fixed-price contracts covering the provision of fuel cycle services. The main features of the agreement are:
  - ⇒ BNFL will take all of NE's remaining uncommitted AGR spent fuel (which could be between 2000 and 3000 tU depending on reactor lifetimes).
  - ⇒ The fuel will be stored at Sellafield and BNFL will decide whether or not to reprocess it. The choice of when to reprocess this fuel provides an important operating benefit for Thorp. It gives BNFL the flexibility to manage the throughput of the plant by choosing the time of reprocessing for this fuel, thus maximising the plant's efficiency.
  - ⇒ Revisions to SN's contracts signed in 1995 have also been agreed.
- A Public Inquiry into the Rock Characterisation Facility (RCF) in West Cumbria has resulted in the application for the construction of the RCF being rejected. As a result the shareholders and customers of Nirex, the organisation charged with developing the RCF and, ultimately, a deep underground LLW/ILW repository, are working with Nirex reviewing options for the future, in terms of organisation and programme, for discussion and agreement with the government.

- The Sellafield MOX Plant (SMP) is currently under construction, and is expected to be operational in 1998. The plant is designed to produce 120 tHM/y of LWR MOX (both PWR and BWR of any design) and will be able to handle plutonium from various sources including Thorp, Magnox reprocessing and third party sources.
- The Calder Hall and Chapelcross Magnox reactors have had their operating licences extended to allow them to operate up to a 50 year lifetime. It is anticipated that the other Magnox reactors will operate for an average lifetime of 37 years. The AGRs are expected to operate for 25 or 30 year lifetimes, although BE believes there are reasonable prospects that the lifetimes of some of the AGRs will be extended. The Sizewell B PWR is expected to have a 40 year lifetime.

## 2. SPENT FUEL MANAGEMENT

### 2.1. Magnox fuel

The strategy for Magnox fuel remains unchanged since the 1995 status report. All fuel will continue to be despatched to BNFL's reprocessing facility at Sellafield.

### 2.2. AGR fuel

Contracts were signed in June 1997 by BNFL and NEL for the provision of spent fuel management services for all remaining uncommitted AGR fuel. Taken with SN's contract signed in 1995 this means that all AGR spent fuel will now be sent to Sellafield where most will be reprocessed in Thorp during the baseload and post-baseload periods.

### 2.3. PWR fuel

The spent fuel storage pond at Sizewell B was designed to accommodate 18 years spent fuel arisings but has recently been reconfigured to accommodate 30 years spent fuel arisings. BE will consider in due course arrangements for further management of spent PWR fuel in the light of the prevailing commercial and regulatory environment.

### 2.4. SGHWR and WAGR fuel

Currently, some 160 tU of SGHWR and WAGR fuel are being stored at Sellafield. It is anticipated that all of this fuel will be reprocessed through Thorp.

### 2.5. Recycle of uranium

Over 15,000 tU of the uranium recovered by Magnox reprocessing has been recycled and about 1,650 tU of AGR fuel has been produced from this material. The recycle of reprocessed uranium from Magnox and AGR fuel currently has limited strategic benefit as assessed against alternative commercial options. The uranium market conditions are such that further Magnox Depleted Uranium (MDU) recycle is not economic at present. Higher residual enrichment product from Thorp is more economic. Construction of the Line 3 Hex plant at BNFL's Springfield's site, for the conversion of reprocessed uranium from Thorp into uranium hexafluoride, is underway with operation due for 2001 and this, along with the newly-opened Oxide Fuels Complex, will allow the closure of the fuel cycle as far as uranium is concerned.

### 2.6. Recycle of plutonium

The position remains unchanged from that reported previously. Briefly, recycle to fast reactors remains the preferred option but, given the withdrawal of Government support for the fast reactor

projects, in the shorter term Plutonium is continued to be stored at Sellafield. BE will consider in due course the feasibility of using MOX fuel at Sizewell B.

Recycle of Plutonium as MOX fuel in AGRs would require significant licensing and operator dose issues to be addressed. Modifications to the station fuel route would also be required which could have an adverse impact on station output.

## **2.7. Fast reactor**

Following the withdrawal of Government support for the project, the Prototype Fast Reactor (PFR) was shut down in March 1994 and is currently being decommissioned. Support for the European Fast Reactor project (EFR) was also withdrawn, although BNFL is continuing to fund EFR research into plutonium burning (the CAPRA project). Fuel from the PFR has been reprocessed in a mixed oxide reprocessing plant at Dounreay since 1979, with the plutonium arising transferred to Sellafield for storage. Completion of PFR reprocessing is due around the year 2000 by which time approximately 50 tHM of fuel will have been reprocessed.

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## SPENT FUEL MANAGEMENT IN THE UKRAINE

A. AFANASYEV

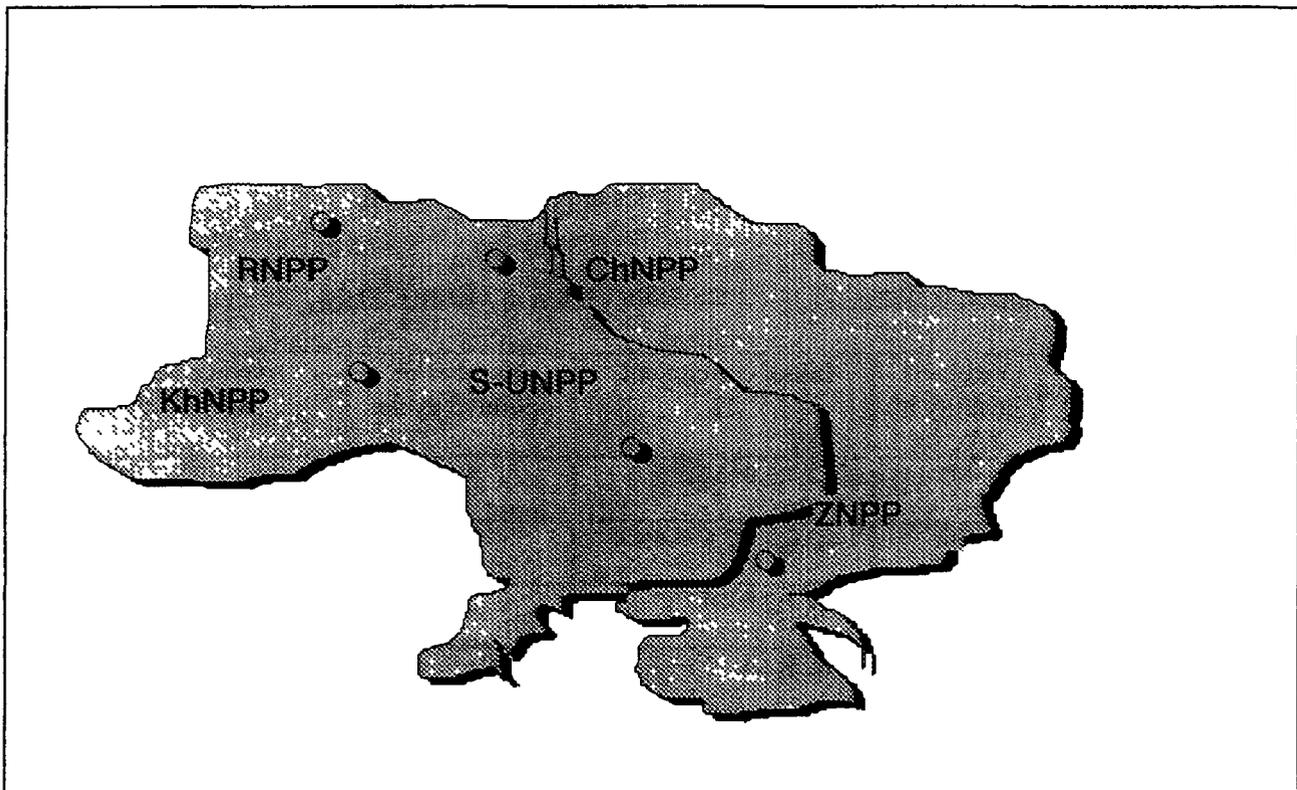
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### Abstract

There are fourteen nuclear power reactors at five NPP sites in operation, representing around 40-50% of the overall electricity production. Four power reactors are under construction. Spent fuel from VVERs-440 and VVERs-1000 is stored at the reactor water pools and partially is shipped to Russia for reprocessing. Dry interim storage systems at the VVER NPP sites are under design or in the process of licensing. Spent fuel from RBMKs-1000 is shipped to the intermediate storage facility at the NPP site. The development of a spent nuclear fuel management concept is underway.

### 1. INTRODUCTION

At present, five NPPs at five sites are in operation in Ukraine representing 23-25% of the total power generating capacity of the country (see Fig. 1 and Table I). Because of the limited national fuel resources and the reduction in supply from Russia (due to the economical difficulties), the Ukrainian NPPs produce now 40-50% of the total national electric power production in comparison to the 16-17% in the former USSR.



*Fig. 1. Location of the Ukrainian NPPs*

TABLE I. NUCLEAR POWER PLANT CAPACITY IN UKRAINE

Plant name	Reactor type	Capacity gross (MWe)	Construction start	Operation start	Status
Chernobyl 1	RBMK-1000	800*	Jun. 1972	Sep. 1977	Shut-down in 1996
Chernobyl 2	RBMK-1000	800*	Feb. 1973	Dec. 1978	Shut-down in 1991**
Chernobyl 3	RBMK-1000	1000	May 1997	Nov. 1981	In operation (on grid)
Rovno 1	VVER-440	402	Aug. 1976	Dec. 1980	In operation
Rovno 2	VVER-440	416	Oct. 1977	Dec. 1981	In operation
Rovno 3	VVER-1000	1000	Feb. 1981	Dec. 1986	In operation
Rovno 4	VVER-1000	1000	Aug. 1986	2001-2002	Under construction
South Ukraine 1	VVER-1000	1000	Mar. 1977	Dec. 1982	In operation
South Ukraine 2	VVER-1000	1000	Oct. 1979	Jan. 1985	In operation
South Ukraine 3	VVER-1000	1000	Feb. 1985	Sep. 1989	In operation
South Ukraine 4	VVER-1000	1000			Constr. canceled in 1990
Zaporozhe 1	VVER-1000	1000	Apr. 1980	Dec. 1984	In operation
Zaporozhe 2	VVER-1000	1000	Apr. 1981	Jul. 1985	In operation
Zaporozhe 3	VVER-1000	1000	Apr. 1982	Dec. 1986	In operation
Zaporozhe 4	VVER-1000	1000	Jan. 1984	Dec. 1987	In operation
Zaporozhe 5	VVER-1000	1000	Jul. 1985	Aug. 1989	In operation
Zaporozhe 6	VVER-1000	1000	Apr. 1986	Oct. 1995	In operation
Khmelnitskaya 1	VVER-1000	1000	Nov. 1981	Dec. 1987	In operation
Khmelnitskaya 2	VVER-1000	1000	Feb. 1985	2001-2002	Under construction
Khmelnitskaya 3	VVER-1000	1000	Mar. 1986	2003-2004	Under construction
Khmelnitskaya 4	VVER-1000	1000	Feb. 1987	2004-2005	Under construction
Chernobyl 4	RBMK-1000	1000	Apr. 1979	Dec. 1983	Accident 26 April 1986

\* Capacity of the units was limited to 800 MW(e) after the Chernobyl accident.

\*\* Chernobyl 2 was shut down in October 1991 after a fire in turbine building. The decision to resume operation is deferred.

## 2. SPENT FUEL MANAGEMENT

The main VVER and RBMK fuel characteristics are listed in Table II.

TABLE II. MAIN CHARACTERISTICS OF USED VVER AND RBMK FUEL

Type of Fuel Assembly (FA)	Uranium quantity in FAs, kg	Initial enrichment, %	Number of operated cycles	Average burn-up MWd/kgU
VVER-1000 for 2-years fuel cycle	429	2.0;3.0;3.0+3.3;3.3	1-4	25-35 for 3.0%-3.3%
VVER-1000 for 3-years fuel cycle	402	1.6;3.0;3.6;3.6+4.4;4.4	1-4	40-41 for 3.6%-4.4%
VVER-440	120	1.6;2.4;3.6;4.4	1-4	31-36 for 3.6% 40-42 for 4.4%
RBMK-1000	115	1.8;2.0;2.4	1000 eff. days	19,4 for 2.4%

## 2.1. Spent Fuel Management Scheme in the Former USSR

The spent fuel management concept as part of the Nuclear Energy Programme is stated in [1,2,3].

After discharge from the reactor, spent fuel assemblies (SFAs) were put into at-reactor cooling ponds where they were stored for at least three years (for RBMK spent fuel at least 1.5 year). After storage in at-reactor ponds, VVER-440 SFAs were transferred to the RT-1 plant ("Mayak" enterprise) for reprocessing. The reprocessed uranium was used for RBMK fuel manufacturing.

VVER-1000 SFAs were transferred to Krasnoyarsk-26 (Zheleznogorsk) for storage in the cooling ponds of the RT-2 plant, which is under construction, and for subsequent reprocessing as soon as the plant would be operating.

RBMK SFAs were shipped to the wet away-from-reactor storage located on the NPP site. Reprocessing of RBMK spent fuel was considered as inexpedient because of the very low content of fissile nuclides (0.4% U-235, 0.25% Pu239+Pu241).

## 2.2. Current Status

After the dissolution of the Soviet Union in 1991, spent fuel from Ukrainian VVERs-1000 was not transported to RT-2 plant until 1995, as a result of the ban that was in force in Russia. From 1995, spent fuel from VVERs-1000 and VVERs-440 has been transferred to the RT-1 and RT-2 plants, according to the contracts concluded between the NPPs and Techsnabexport (for VVER-1000) and "Mayak" enterprise (for VVER-440). One of the conditions of the contracts is the return of vitrified high level wastes to Ukraine after spent fuel reprocessing.

The interruption in spent fuel transfers to RT-2 in the period of 1991-1994, as well as the delay in the construction of RT-2 and in the implementation of the MOX fuel programme in the existing VVERs, have caused the following actions from the Ukrainian utilities:

- re-racking of spent fuel ponds;
- launching of dry interim storage projects.

The total quantity of SFAs unloaded from Ukrainian reactors during the whole period of operation (as of 01 September 1997) is:

<b>VVER-1000:</b>	Fuel Assemblies for 2- years cycle:	- 2956 (1268 tU)
	Fuel Assemblies for 3- years cycle:	- 2118 (851 tU)
	Total Fuel Assemblies:	- 5074 (2119 tU)
<b>VVER-440:</b>	Fuel Assemblies:	- 3277 (393.2 tU)
<b>RBMK-1000:</b>	Fuel Assemblies:	- 18042 (2074.8 tU)
<b>Total Fuel Assemblies</b> for all reactors:		- 26393 (4587 tU).

The total quantity of VVER-1000 spent fuel assemblies that have been shipped to Russian plant RT-2 (for storage and subsequent reprocessing after of the plant's construction) and to the institute of Atomic Reactors (NIAR) (for test and examination) is 2186. Among them, 1356 fuel assemblies were shipped under the contract between the Ukrainian NPPs and Techsnabexport from 1995 until September 1997.

The total quantity of VVER-440 spent fuel assemblies shipped to RT-1 for reprocessing is 2552. Among them, 1047 fuel assemblies were shipped under the contract between Rovno NPPs and PO "Mayak" from 1993 until September 1997. Till the end of 1997 the following shipments are planned:

- ◆ to RT-1 - 240 VVER -440 fuel assemblies;
- ◆ to RT-2 - 204 VVER -1000 fuel assemblies.

Tables III and IV illustrate the storage capacities and spent fuel arisings for RBMK and VVER.

TABLE III. RBMK STORAGE CAPACITY AND SPENT FUEL ARISING

(as of 01 September 1997)

Units, Facility	Capacity for FAs in Cooling Pool	Number of FAs in Cooling Pool
1	1728	1352
2	1728	1058
3	1728	891
AFR wet storage	17520	14741
<b>Total</b>	<b>22704</b>	<b>18042</b>

TABLE IV. VVER COOLING POOL CAPACITY AND SPENT FUEL ARISING

(as of 01 September 1997)

Plant name	Capacity for FA in Cooling Pool	Number of FA in Reactor Core	Number of FA in Cooling Pool	Number of unoccupied cells for FA	Number of FA in Pool as of 31 Dec. 1997 (forecast)
Zaporozhe 1	447*	163	195	252	244
Zaporozhe 2	440*	163	315	125	315
Zaporozhe 3	378	163	310	68	286
Zaporozhe 4	381	163	349	32	349
Zaporozhe 5	378	163	327	51	327
Zaporozhe 6	579*	163	54	525	54
<b>Total</b>			<b>1550</b>	<b>1053</b>	<b>1575</b>
South Ukraine 1	463**	163	232	231	232
South Ukraine 2	463**	163	213	250	213
South Ukraine 3	605**	163	259	346	254
<b>Total</b>			<b>704</b>	<b>827</b>	<b>699</b>
Khmelnitskaya 1	610*	163	238	372	238
Rovno 3	685**	163	396	289	378
<b>Total for all VVER-1000</b>			<b>2888</b>	<b>2541</b>	<b>2890</b>
Rovno 1	729	313	321	408	221
Rovno 2	729	349	404	325	264
<b>Total for all VVER-440</b>			<b>725</b>	<b>733</b>	<b>485</b>

\* after uncompleted (partial) re-racking

\*\* after finished re-racking

### 2.3. Forecast of spent fuel generation from 1998 to 2010

The estimated spent fuel arisings shown in Tables V and VI have been calculated on the assumption that:

- ◆ from 1998 onwards, advanced VVER-1000 Uranium-Gadolinium fuel assemblies (for four cycles, with an average burnup of 46-47 MWd/kgU and with a mass of 435 kgU per FA) will be used;

- ◆ between 1998-2006, the average operation period of the VVERs-1000 will be decreased to 10%, because of modernization of reactor monitoring and control systems;
- ◆ Chernobyl 3 and Chernobyl 2 will be in operation between 1998-2000.

TABLE V. VVER SPENT FUEL ARISING (FORECAST)

Type of reactor	Number of SFAs/ mass of uranium (tU)													
	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	Total
VVER-1000	523	523	523	577	629	629	600	560	560	600	600	600	600	7524
	---	---	---	---	---	---	---	---	---	---	---	---	---	---
	210	210	210	231	262	262	249	243	243	261	261	261	261	3164
VVER-440	204	196	196	196	196	196	196	196	196	196	196	196	196	2556
	---	---	---	---	---	---	---	---	---	---	---	---	---	---
	24.4	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	306.4

TABLE VI. RBMK SPENT FUEL ARISING (FORECAST)

Year	1998	1999	2000	2001	Total
Number of SFAs	1900*	710	710	3100**	6420
Mass of uranium (tU)	218	81.6	81.6	356	737.2

\* including FA in Reactor Core of unit 1 after shutdown in 1996

\*\* including FA in Reactor Core of unit 2 and 3

#### 2.4. Present status of on-site VVER dry storage

For all VVER NPPs, cask storage has been chosen. In 1997, the South-Ukrainian NPP jointly with the Institute of Complex Power (VNIPIET-Russia) finished tests of the modernized cask TK-13 (TK-13M). The capacity of the cask TK-13M is 12 SFAs and the duration of SFAs storage is 5 years. The usage of such casks on NPPs sites (South Ukraine-8, Rovno-7, Zaporozhe-8, Khmel'nitskaya-6) allows to ship spent fuel from NPPs sites to RT-2 independently from of the reactor shut-down for fuel reloading.

The Zaporozhe NPP jointly with Duke Engineering and Services (DE&S) company (USA) is developing a spent fuel storage (SFS) project on the basis of a metal and concrete container VSC-24 by SIERRA NUCLEAR. Transportation devices and details for the manufacturing of the first container have already been delivered from USA. The Safety Analysis Report for construction and operation of SFS is under consideration in the Nuclear Regulatory Administration. It is expected that the license of the SFS construction will be issued in October-November 1997, taking into consideration the results of the US NRC analysis of weld-related cracks found in VSC-24s at US sites.

Besides VSC-24, the following containers are being considered for interim storage of VVER fuel:

- ◆ two-purpose metal containers designed by "EMK" enterprise (Russia) with a capacity of 24-30 FAs. The containers are planned for production at Russian and Ukrainian plants, however, the design of the container is not finalized yet.
- ◆ KhNPP and RNPP are considering a metal-concrete container of the GNB company.

## 2.5. Programme of future work

Goscomatom has worked out a Spent Fuel Management Programme in Ukraine up to the year of 2010. The programme considers the following issues:

- Continuation of spent fuel transfer from VVERs-1000 and VVERs-440 to Russia (RT-1, RT-2).
- Re-racking of spent fuel ponds at RNPP (VVER-440) and ZNPP during 1997-2002.
- Spent fuel storage in TK-13M containers during 1-5 years on NPP sites.
- Research on state of spent fuel pins under long-term storage (Spent fuel assemblies from ZNPP were transported to NIJAR, Russia, and a contract was signed for research till the year of 2010, but the work is suspended because of the financial problems).
- Choice and construction of container storage at SUNPP, KhNPP, RNPP sites. Two-purpose metal containers are more preferable and the initial planned capacity of the storage is 4-5 reloadings (1997-2002).
- Re-racking of the spent fuel storage at the ChNPP site (1998-2001). Organization of a tender on dry storage design for RBMK spent fuel, within the framework of ChNPP decommissioning (1997-1998). Construction of dry SFS-2 (1998-2002).
- Site and design choice and construction of the regional spent fuel storage for VVER fuel (2002-2005). In the framework of the TACIS programme, experts of the European Fuel Cycle Consortium ( EFCC ) have carried out a comparative analysis of regional SFS on the basis of metal concrete casks, double-purpose metal casks and vaults since 1996. Ukrainian experts participated in this work.
- Changes to the legislative basis.
- Creation of research laboratories for selection of geological formations for vitrified high level waste and spent fuel disposal.

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# STATUS OF SPENT NUCLEAR FUEL MANAGEMENT IN THE UNITED STATES OF AMERICA



XA9846861

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## Abstract

The United States produces approximately 20% of its electricity in nuclear power reactors, currently generating approximately 2,000 metric tons of uranium (tU) of spent nuclear fuel annually. Over the past half century, the country has amassed 33,000 tU of commercial spent nuclear fuel that is being stored at 119 operating and shutdown reactors located on 73 sites around the nation. The cumulative discharge of the spent fuel from reactors is estimated to total approximately 87,000 tU by 2035. Many sites have reracked the spent fuel in their storage pool to maximize pool capacity, and a number of reactor sites have been forced to add dry storage to accommodate the growing inventory of fuel in storage. In addition, research and defense programme reactors have produced spent fuel that is being stored in pools at Federal sites. Much of this fuel will be transferred to dry storage in the coming years. Under current plans, the commercial and federally owned fuel will remain in storage at the existing sites until the United States Department of Energy (DOE) begins receipt at a federal receiving facility.

## 1. INTRODUCTION

The Nuclear Waste Policy Act, as amended, sets forth the nation's policy for the management of spent nuclear fuel and high-level radioactive waste. This policy calls for the development of one or more geologic repositories for the direct disposal of spent fuel and the disposal of high-level radioactive waste that resulted from the reprocessing of spent fuel for defense purposes. A small quantity of high-level waste also exists from the reprocessing of commercial spent nuclear fuel that will be disposed of in a geologic repository. Presently, DOE is investigating Yucca Mountain, Nevada for development as the nation's first geologic repository. If the site is found suitable, the Department would submit an application to the Nuclear Regulatory Commission (NRC) for authorization to construct a repository. With timely regulatory approval, operations at the repository would commence in 2010.

Early reactor spent fuel storage pools were small because it was presumed that spent fuel would be shipped off-site for reprocessing. Approximately two decades ago, the US abandoned efforts to reprocess commercial spent nuclear fuel. Consequently, many older reactors found themselves with spent fuel pools that could not accommodate the accumulation of fuel from a lifetime of reactor service. Barring authorization from Congress for DOE to establish a centralized interim storage facility in the country, acceptance of commercial reactor fuel by DOE would not be expected to occur until 2010 when the repository is expected to commence operations. Therefore, many sites have needed to examine alternate storage possibilities, including dry storage.

The Nuclear Waste Policy Act recognized the impact of cessation of reprocessing and anticipated this need for dry storage. The Act authorized demonstration and cooperative programmes to help develop and certify dry storage technologies. As such, demonstrations were conducted leading to the development of several of the dry storage technologies that are now commercially available.

## 2. SPENT FUEL STORAGE TECHNOLOGIES

Several dry storage technologies are currently in use at reactor sites around the country. Table I lists those certified storage-only technologies.

TABLE I. CERTIFIED STORAGE-ONLY TECHNOLOGIES

<b>Vendor</b>	<b>Technology</b>
General Nuclear Systems, Inc.	Castor V/21
General Nuclear Systems, Inc.	Castor X/33
General Nuclear Systems, Inc.	Castor X/74
Foster Wheeler	MVDS
Westinghouse	MC-10
NAC International	NAC 128 S/T
Sierra Nuclear	VSC-24
Transnuclear	TN-24
Transnuclear	TN-32
Transnuclear	TN-40
VECTRA	NUHOMS-7P
VECTRA	NUHOMS-24P
VECTRA	NUHOMS-52B

Several commercial vendors are pursuing development of “dual-purpose” technologies, which can be used in both storage and transportation. These designs will help minimize handling of the spent fuel which will reduce exposure to workers. Table II shows a list of those technologies that have obtained or are being developed for pursuit of dual-purpose certifications.

TABLE II. DUAL PURPOSE CERTIFICATION TECHNOLOGIES

<b>Vendor</b>	<b>Technology</b>
NAC International	MPC
NAC International	UMS
NAC International	NAC Storable Transportation Cask
Holtec	HI-STAR 100
Sierra Nuclear	TranStor
VECTRA	MP-187
Westinghouse	WESTFLEX

DOE had been pursuing development of a multi-purpose canister (MPC) that could be used during storage, transportation, and disposal of spent fuel. This system would have enabled spent fuel to be placed in disposable canisters at reactor sites, transported to the repository and placed in waste packages without the need to handle the bare fuel assemblies. However, in response to budgetary constraints on the programme, the DOE-led MPC programme was terminated by the Department. Private industry is expected to pursue technologies to pursue these functions.

Prior to termination of the MPC programme, DOE provided the specifications for the MPC to both vendors and utilities. Several utilities have used the specifications when procuring dual purpose dry storage systems. Should these canisters prove to meet the ultimate disposal package requirements, the programme could realize substantial savings versus having to repackage the spent fuel.

### 3. SPENT FUEL MANAGEMENT AT REACTORS

Several U.S. commercial reactors have already reached the maximum capacities of their spent fuel pools and have had to resort to dry storage systems to accommodate their storage needs. Currently, there are 10 reactor sites with spent fuel in dry storage. Presently, there is approximately 1,500 tU of fuel in dry storage at these sites (Table III). A number of other utilities have contracted to procure dry storage technologies (Table IV).

TABLE III. SPENT FUEL DRY STORAGE SYSTEMS AT REACTOR SITE

Site	State	Vendor/Technology
Robinson	South Carolina	Vectra Technologies/NUHOMS-7P
Oconee	South Carolina	Vectra Technologies/NUHOMS-24P
Calvert Cliffs	Maryland	Vectra Technologies/NUHOMS-24P
Fort St. Vrain*	Colorado	Foster Wheeler/MVDS
Palisades	Michigan	Sierra Nuclear Corp./VSC-24
Prairie Island	Minnesota	Transnuclear, Inc./TN-40
Surry Station	Virginia	Westinghouse/MC-10
		General Nuclear Systems/Castor V-21
		Nuclear Assurance Corp./NAC-128 S/T
Davis-Besse	Ohio	Vectra Technologies/NUHOMS-24P
Point Beach	Wisconsin	Sierra Nuclear Corp./VSC-24
Arkansas Nuclear One	Arkansas	Sierra Nuclear Corp./VSC-24

\* Fort St. Vrain was a gas-cooled graphite moderated reactor and is currently shut down. The Department of Energy is in the process of assuming the NRC license for its dry storage facility.

TABLE IV. PROCURED DRY STORAGE TECHNOLOGIES

Site	State	Vendor/Technology
Rancho Seco*	California	Vectra Technologies/MP-187
Oyster Creek	New Jersey	Vectra Technologies/NUHOMS-52B
Susquehanna	Pennsylvania	Vectra Technologies/NUHOMS-52B
Trojan*	Oregon	Sierra Nuclear Corp./TranStor-24
North Anna	Virginia	Transnuclear, Inc./TN-32
Cook	Michigan	Holtec International/HISTAR 100
Dresden	Illinois	Holtec International/HISTAR 100
Yankee Rowe*	Massachusetts	NAC/NAC UMS
Hatch	Georgia	(Technology not selected)

\* Shut down reactor sites

4. PROJECTED DRY STORAGE NEEDS

The United States continues to produce approximately 2,000 tU of spent nuclear fuel annually. At the current time, approximately 33,000 tU of commercial spent nuclear fuel is stored in a combination of spent fuel pools and dry storage at 73 operating and shutdown reactor sites. A number of sites have already turned to dry storage to handle their excess storage needs, and as the schedule for opening the repository is still more than a decade away and centralized interim storage is not authorized, the number of affected sites will continue to increase (Table V).

TABLE V. PROJECTED DRY STORAGE NEEDS

Year	Affected Sites	Affected States	tU
1997	10	9	1,500
2000	22	18	3,000
2010	53	30	13,000
2020	63	31	40,000

Several factors exist that could impact dry storage need estimates. Utility restructuring, which could close uneconomical nuclear power plants prior to license expiration, could significantly reduce dry storage need estimates. Also, the current trend of reactors using fuels to higher burn-up levels could reduce the estimates. Should certain reactors prove to be economical, utilities have the option to submit license extension applications to NRC, which could increase the life of the reactor and, thus,

the amount of spent fuel generated. No utility has yet applied for a license extension. Figure 1. illustrates the projected dry storage needs throughout the United States over the next several decades.

Additionally, a number of commercial nuclear power plants have already ceased operations. As decommissioning of these facilities commence, placing spent fuel in dry storage may help expedite the cleanup process. The shutdown facilities in the United States are listed in Table VI with the quantity of spent fuel still on site in fuel pools.

TABLE VI. SHUTDOWN NUCLEAR POWER PLANTS

Plant	Year Shutdown	Assemblies / tU in fuel pool
Dresden	1978	683 / 70
Fort St. Vrain	1989	All Spent Fuel in Dry Storage
Haddam Neck	1996	1000 / 410
Humboldt Bay	1976	390 / 29
Indian Point 1	1974	160 / 31
La Crosse	1987	333 / 38
Rancho Seco	1989	493 / 228
San Onofre 1	1992	207 / 76
Shoreham	1989	All Spent Fuel Transferred to Limerick
Three Mile Island 2	1979	Damaged Fuel was sent to INEEL for Scientific Study
Trojan	1992	780 / 359
Yankee Rowe	1991	533 / 127

## 5. STORAGE OF DOE SPENT NUCLEAR FUEL

DOE suspended reprocessing of its defense spent fuel in 1992, recognizing the reduction in requirements for weapons material. Consequently DOE has an inventory of over 2000 tons of spent fuel that is being stored at several sites across the country. Due to the nature of this fuel, some may require substantial treatment prior to disposal. In addition, the Department is accepting spent fuel from foreign research reactors. Over the past several decades, DOE has previously accepted 8 tU of foreign fuel, and the Department expects to accept an additional 20 tU of fuel from foreign countries. This fuel will be stored at DOE sites prior to shipment to a repository.

A programme was recently authorized to accept research and defense reactor spent nuclear fuel for disposal in the repository. Due to the degraded nature of much of the research and defense spent nuclear fuel and requirements in various legal agreements, the Department needs to transfer this fuel to dry storage prior to the time the repository will begin operations. Transfer of this fuel to dry storage presents several challenges. Since the fuel for these reactors may be unique with regard to dimensions, enrichment, burn-up, and physical condition, existing dry storage technologies have not been certified for use with this kind of fuel. Additional work is required to obtain certifications for the technologies in which these fuels will need to be stored and shipped.

Fuel from the shutdown Fort St. Vrain graphite reactor is currently in dry storage at the reactor site in Colorado. The Department has agreed to assume the NRC license for this facility. This effort is expected to be precedent-setting for the Department, since DOE has never been regulated by NRC in this manner before. Because NRC may ultimately regulate many more DOE nuclear facilities, a high level of attention is being paid to the process.

At the Idaho National Engineering and Environmental Laboratory (INEEL), fuel from the melted Three Mile Island Unit-2 core is currently stored. Although this fuel was generated at a commercial reactor, the Department has assumed responsibility for its care, and INEEL is pursuing a dry storage license from NRC, consistent with legal agreements.

Scenario: Announced shutdowns + 30 year life

(Assuming no acceptance of spent nuclear fuel by DOE, all shutdown reactors unloaded into dry storage)

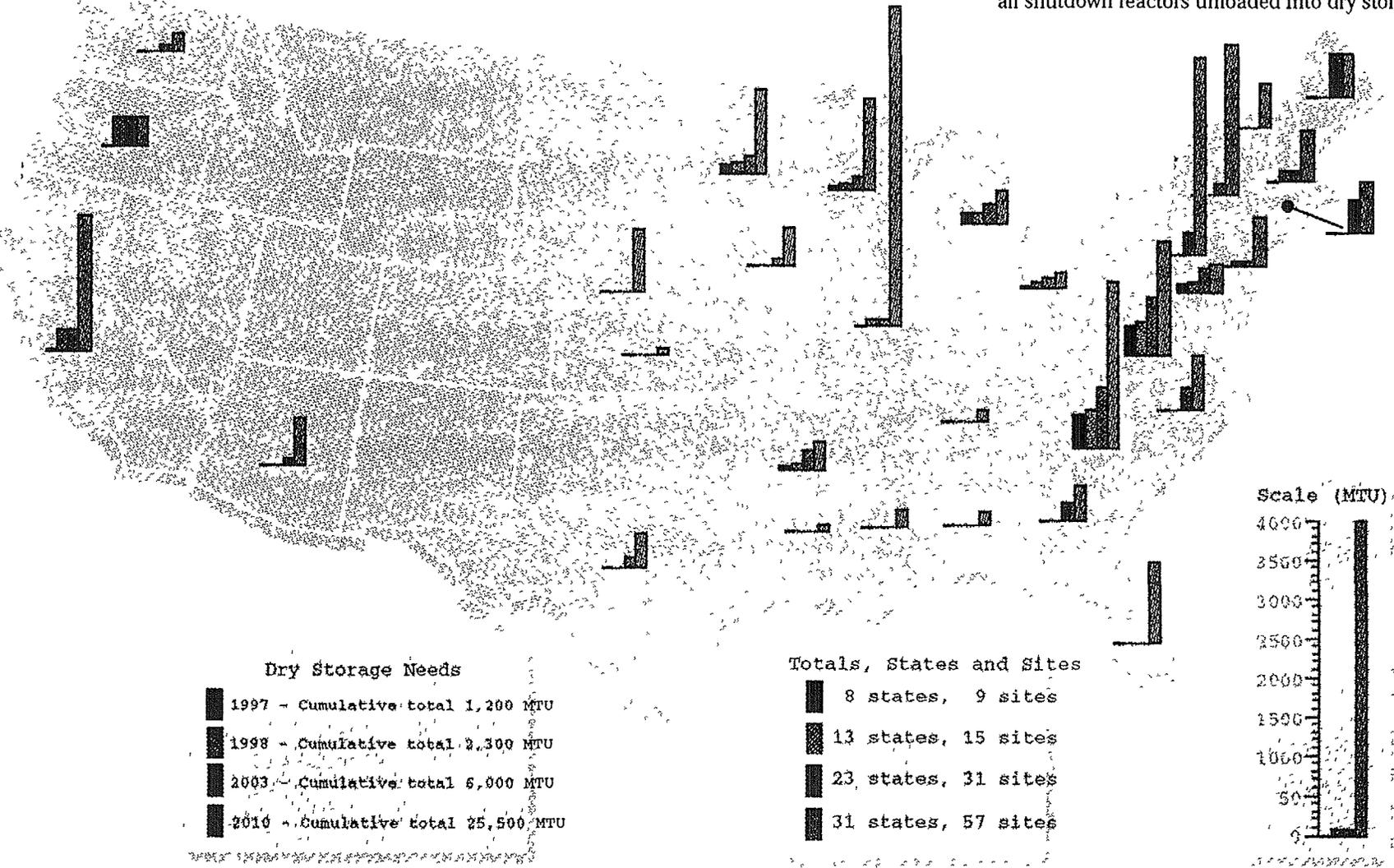


FIG. 1. Projected dry storage needs.

## 6. CENTRALIZED INTERIM STORAGE

Authorization for DOE to construct an interim storage facility has either been revoked or expired without success. Legislation currently under consideration in Congress would authorize the Department to construct and operate an interim storage facility prior to completion of the repository. Selection of a site remains a principal impediment to the passage of this legislation.

Since the final site selection of an interim storage facility is a major uncertainty, DOE has taken the approach to develop a non-site specific Topical Safety Analysis Report (TSAR). The TSAR assumes the bounding natural phenomena (e.g., earthquake, tornado, snow) conditions from most of the United States. Both NRC and DOE understand that a site must be selected before many issues can be resolved and a license issued. The expectation on both sides is that this TSAR will flag problem issues early, which will, in turn, help expedite the licensing process should DOE be granted the authority to construct such a facility.

The Department is also working with the Electric Power Research Institute (EPRI) to develop a dry transfer system (DTS) for spent nuclear fuel. A DTS TSAR was submitted to NRC in September of 1996. DOE hopes that NRC will complete their review in April 1998 and that this effort will help identify potential problem issues prior to submitting a license application for such a facility.

Although the Department is not authorized to perform centralized interim storage activities, private efforts to perform that role are permitted. The Private Fuel Storage Limited Liability Company (PFS LLC), founded by a consortium of utilities, has taken up the effort to pursue centralized interim storage. The group is working with the Goshute Indian Tribe to build a storage facility with a capacity of 40,000 tU on the Skull Valley Reservation in northwestern Utah. The PFS LLC has met with the NRC several times and recently submitted a license application for the facility. Although resistance to the facility is expected to be substantial, the group would like to initiate construction in the year 2000, with operations beginning in 2002.

## 7. REPOSITORY

DOE is characterizing the Yucca Mountain site in Nevada for development as the nation's first geologic repository. The site is located approximately 100 miles northwest of Las Vegas, Nevada, and is adjacent to the Nevada Test Site, where the United States conducted atomic weapons tests for years. The near-term milestones for work on the Yucca Mountain site are completion of the viability assessment in 1998, making a site suitability determination in 2000, and submitting a license application to the NRC in 2002, should the site be deemed suitable. Should everything go according to schedule, the license would be received and the repository ready for operation by the year 2010. Absent authorization for an interim storage facility, removal of spent fuel from reactor sites would be expected to commence in 2010 and be completed by 2044.

## 8. ENSURING SAFETY - MATERIALS ISSUES WITH STORAGE CASKS

The NRC has been reviewing the safety of storage casks following a hydrogen gas ignition event in a Sierra Nuclear Corporation VSC-24 cask at the Point Beach nuclear plant on May 28, 1996. The generic concern underlying the event was that the cask design criteria regulations and cask designs are more strongly focused on structural considerations rather than materials selection.

The event occurred during transfer of fuel from wet to dry storage. Two successful transfers had previously been completed without incident at the facility. On May 27, the VSC-24 cask was lowered into the spent fuel pool, loaded with fuel, and subsequently placed in the cask decontamination area. Borated water from the spent fuel pool remained in the cask, with a limited number of gallons drained to create an air space to permit seal welding. Eleven hours later, when welding commenced, hydrogen gas ignited. The ignition displaced the 6,400 pound shield lid, but did not damage any assemblies and no injuries or radioactive releases resulted from the event.

The root cause of the event was the use of Carbo Zinc 11 primer to coat the inside of the VSC-24 cask to enhance corrosion resistance. When left to interact with the borated water in the cask for an extended period of time, zinc reacted with the water to form zinc oxide, zinc hydroxide, and hydrogen, and other compounds. By the time welding commenced, sufficient hydrogen had built up in the cask under the shield lid to ignite and displace the lid.

As a result of this event, much more attention will likely be paid to materials compatibility considerations in the cask certification process. Cask designs currently in progress will have the benefit of this experience before construction; however, for casks currently in use, their safety basis may receive much more scrutiny in the materials area during periodic NRC inspections and when they are up for license renewal after 20 years in operation. The American Society for Testing and Materials is currently developing a standard to address certification concerns for storage casks.

One additional concern has been raised recently concerning storage casks. Seal welds on certain casks have demonstrated cracking. While there is no immediate safety threat from this, the Nuclear Regulatory Commission is concerned that cracking could lead to loss of inert atmospheres in these casks. If the inert atmosphere is compromised, fuel degradation may occur, increasing the difficulty of rehandling the fuel should it become necessary. No definitive conclusions have been reached on the cause of the seal weld cracking.

## 9. TRANSPORTATION

The DOE transportation programme has been working on an approach to procure commercial vendor services to transport spent nuclear fuel and high-level waste from the numerous storage sites to the eventual repository location. The Department has also ceased development of transportation equipment, opting to rely on industry to develop the necessary equipment by the time spent fuel is ready for shipment. Some industry equipment already exists, while other designs are currently under development.

## 10. CONCLUSION

The accumulation of spent fuel at reactor sites is continuing at a rate of approximately 2,000 tU annually. As reactors reach the capacity of their spent fuel pools, spent nuclear fuel is being stored on site in a variety of dry storage technologies. As reactors are beginning to undergo decommissioning, there is a trend to develop multi-use technologies for transport and storage. With the desire to reduce spent fuel handling, it is expected that disposal considerations will begin to be incorporated into these designs. Barring new legislation authorizing centralized interim storage, the need to develop dry storage at reactor sites will continue until the repository opens. By 2010, it is expected that over 13,000 tU of fuel will be in dry storage.

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