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Extended storage of spent fuel

Final report of a Co-ordinated Research Programme on the Behaviour of Spent Fuel and Storage Facility Components During Long Term Storage (BEFAST-II) 1986–1991

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FOREWORD

This document is the final report on the IAEA Co-ordinated Research Programme on the Behaviour of Spent Fuel and Storage Facility Components during Long Term Storage (BEFAST-II, 1986-1991). It contains the results on wet and dry spent fuel storage technologies obtained from 16 organizations representing 13 countries (Argentina, Canada, Germany (previously the German Democratic Republic and the Federal Republic of Germany), Finland, Hungary, Italy, the Republic of Korea, Japan, Sweden, the United Kingdom, the USA and the USSR) who participated in the co-ordinated research programme.

Considerable quantities of spent fuel continue to arise and accumulate. Although some new reprocessing facilities are being constructed, many countries are investigating the option of extended spent fuel storage prior to reprocessing or fuel disposal. Wet storage continues to predominate as an established technology with the construction of additional away-from-reactor storage pools. However, dry storage is increasingly used with most participants considering dry storage concepts for the longer term. These topics were evaluated by all the participants as very important and helpful for the nuclear community.

The IAEA wishes to thank the BEFAST-II Chairman, E. Vitikainen (Technical Research Centre of Finland) and the Group Leaders, C.W.E. Addison (United Kingdom), V. Kritskij (USSR) and R. Lambert (USA), who were responsible for the preparation of the report, as well as all the programme participants, observers and consultants. The Scientific Secretaries of the CRP were V. Onufriev and G. Sukhanov, and the staff member responsible for this report was F. Takáts, all of the Division of Nuclear Fuel Cycle and Waste Management.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

Extended spent fuel storage is an important activity for all countries with nuclear power programmes because fuel, after its discharge from the reactor, is required to be stored before reprocessing or final disposal. The storage period is highly dependent upon the individual national strategies to complete the nuclear fuel cycle.

This is why 12 organizations from 11 countries (Austria, Canada, Czechoslovakia, Finland, Germany (previously the German Democratic Republic and the Federal Republic of Germany), Hungary, Japan, Sweden, USA, and USSR) participated in the Co-ordinated Research Programme (CRP) "BEFAST-I" (Behaviour of Spent Fuel Assemblies During Extended Storage) under the auspices of the IAEA to exchange information on spent fuel storage technology. The CRP was performed from 1981 to 1986 and the information collected covered the following aspects:

- potential fuel degradation mechanisms during wet and dry storage;
- spent fuel examination and surveillance programmes;
- impact on storage facilities and equipment.

The final report of the CRP [1] was published after the final Research Co-ordination Meeting (RCM)/ Technical Committee Meeting (TCM) held in Leningrad, USSR, in May 1986 [2]. The unanimous opinion of all BEFAST-I members was that the co-operation and the results obtained in the programme were important for all participating countries and the participants recommended continuation of this effort during the next five years (1986-1991) [3].

The new follow-up programme "BEFAST-II, Behaviour of Spent Fuel and Storage Facility Components during Long Term Storage", was initiated by the IAEA in September 1986 and 16 organizations from 13 countries (Argentina, Canada, Germany (previously German Democratic Republic and Federal Republic of Germany), Finland, Hungary, Italy, Republic of Korea, Japan, Sweden, the United Kingdom, the USA and the USSR) participated in the CRP.

Altogether eight subjects were initially proposed to be covered by the BEFAST-II:

(i) Wet and dry storage experience;
(ii) Spent fuel monitoring,
(iii) Effects of decontamination on materials,
(iv) Handling and transport of spent fuel after storage,
(v) Storage of defected fuel,
(vi) Storage of fuel exceeding 50 years, extrapolation of present experience,
(vii) Predictive models: Failure mechanisms including material aspects, and
(viii) Crud impact on spent fuel integrity during extended storage.

These subjects were later grouped under three major topics for both wet and dry storage:

(a) Long term behaviour,
(b) Surveillance, and
(c) Facilities & operation,

which also form the basis for this final report.

TABLE I. BEFAST-II RESEARCH SUBJECTS

Long-term Behavior	Surveillance	Facilities & Operation
● Materials aspect (claddings & components) ● Degradation mechanisms and Models ● Validation - experimental - experience	● Monitoring, Wet/Dry - environment - components - fuel assemblies - dose rate of workers ● Fuel Conditions - operational - fabrication - technology - defect fuel rods and assemblies ● Different Reactor Types	● Dose Rate Reduction ● System Performance ● Changing Modes wet - dry ● Capacity Enhancement - high density racks - reracking - double tiering - doped coolant - rod consolidation ● Handling of Heavily Damaged Fuel

Distribution of BEFAST-II research subjects within the agreed major topics is shown in Table I, and the research matrix is shown in Table II.

Three research co-ordination meetings were held in the course of the programme: the first in April, 1988 in Budapest, Hungary; the second in August, 1989 in Espoo, Finland, and the third in March, 1991 in Vienna, Austria.

In addition three BEFAST-II consultants meetings were held (Erlangen, Saluggia, Vienna), which prepared programmes and recommendations for the RCMs. The preparation and processing of this report were executed by the working groups formed by the programme participants during the RCMs.

This final report was one of the objectives of the CRP. Another objective was to provide a forum for the exchange of information on spent fuel storage between specialists from Member States. Table III gives an overview of spent fuel management technologies in the participating countries.

A list of BEFAST-II contributions with the latest research results are given in Annex I. Updated reports on experience in spent fuel storage and spent fuel management strategies in various countries are presented in Annex II.

On completion of the BEFAST-I and II programmes, it was appropriate to consider future activities in the field of extended storage of spent fuel.

A note was prepared during the final research co-ordination meeting of BEFAST-II, which suggested the formation of a new five year CRP - BEFAST-III (1992-1996). The major objective for BEFAST-III is to collect and exchange experience on spent fuel storage to form an international database. In addition, any research results relevant to fuel storage could also be reported to BEFAST-III.

TABLE II. BEFAST-II RESEARCH MATRIX

SUBJECTS / COUNTRY	A LONG TERM BEHAVIOUR			B SURVEILLANCE		C FACILITIES & OPERATIONS	D OTHER
	MATERIALS	MECHANISMS & MODELS	VALIDATION	WET	DRY		
ARGENTINA				- wet storage experiments			
CANADA	- UO_2-Zr-cladding, wet + dry storage	- mechanisms of UO_2 oxidation. wet + dry storage	- experimental, experience, wet + dry storage	- Zr-clad UO_2 stored since 1962	- Zr-clad UO_2 since 1974 in comcrete canisters	- wet storage of spent fuel 26a (1988) - dry, ERB, CEX-1, -2 (concrete canisters)	
FINLAND	-------→		- wet storage experience	- periodical examinations			
GERMAN DEMOCRATIC REPUBLIC		-------→				- verification of temperature calculation models in dry casks	
FEDERAL REPUBLIC OF GERMANY	-------→	- life-time prediction methods for dry-stored fuel - computation of spent fuel rod behaviour	- review of available dry storage data			←-------	
HUNGARY	- investigation of local corrosion processes in Al and ZRYs			- corrosion monitoring methods		- compact storage in water pools (collection of basic data)	
ITALY				-------→		- handling and transport of LATINA spent Magnox fuel	
REPUBLIC OF KOREA			- water pool storage experiments	- periodical examination - controlled storage of defected fuel in water pools			

TABLE II. (cont.)

SUBJECTS / COUNTRY	A LONG TERM BEHAVIOUR			B SURVEILLANCE		C FACILITIES & OPERATIONS	D OTHER
	MATERIALS	MECHANISMS & MODELS	VALIDATION	WET	DRY		
SWEDEN	-------→			-------→		- examination of damaged spent fuel - discolution and leaching processes	
UNITED KINGDOM	- oxidation of UO_2 in dry storage	←------→	- wet storage experience	- monitoring of wet storage		←-------	
USA	-------→	-------→	- wet storage experience	←-------	- dry storage program	←-------	
USSR	- UO_2-Zr+1%Nb cladding wet + dry storage	- life time prediction models for dry and wet stored fuel	- wet storage experience	- monitoring of wet storage	- dry storage program	- verification of temperature calculation models in dry casks	

TABLE III. SPENT FUEL MANAGEMENT TECHNOLOGIES (INTERNATIONAL OVERVIEW FROM BEFAST-II MEMBERS)

		Arge ntina	Cana- da	Ger- many	Fin- land	Italy	Japan	Korea R. of	UK	USA	Swe- den	USSR	Hun- gary
Spent Fuel Pool	Standard	X	X	X	X	X	X	X	X	X	X	X	X
	Densified Storage		X	X	X	X	X	X	-	X	X	X	X
Spent Fuel Storage (AR)	Wet	X	X	X	X	X	X	X	X	X		X	
	Dry cask	+	X	X		Δ		Δ		X		+	
	vault				Δ			Δ	X	O			
	drywell												
Rod Con-soli-dation	Wet			O			Δ			X	Δ	O	
	Dry								X	+			
Spent Fuel Storage (AFR)	Wet			X		X	X	+	X	X	X	X	
	Dry cask		+	X			O			X		+	
	vault			+				Δ					
	drywell						X						

TABLE III. (cont.)

		Argentina	Canada	Germany	Finland	Italy	Japan	Korea R. of	UK	USA	Sweden	USSR	Hungary
Spent Fuel conditioning plant				O					O		Δ		
Interim HLW-Storage		O		O			X	Δ	X			Δ	
Reprocessing		O	Δ	O			X	Δ	X O				
Final Repository		+	+	+	+		+	Δ	Δ	+	Δ	Δ	
Transport	Wet	+	X	X	X	X	X	X	X		X	X	X
	Dry		X	X		X	X	+		X	X	X	X

X = in practice + = under development, in exploration
O = under construction Δ = envisaged

2. WET STORAGE

This chapter discusses:

(a) Methods of monitoring and assessing ongoing spent fuel integrity,
(b) Factors that affect intact and defective spent fuel integrity,
(c) Factors important to maintaining water pool liner and structure longevity,
(d) Operational experience with wet storage and methods of enhancing water pool capacity, and
(e) Experience with water pool spent fuel handling and transfer of spent fuel from wet to dry storage.

2.1. SPENT FUEL STORAGE PERFORMANCE

2.1.1. Zirconium alloy and stainless steel clad fuel

Zircaloy clad spent fuel[1] has shown reliable behaviour during storage. It is licensed in BEFAST-II member countries. Positive experience is available for up to 27 years of storage (see Tables IV and V). Thus it can be predicted that at least 50 years of storage is feasible and would result in no fuel integrity problems [4]. However, this positive storage performance is closely related to maintaining specified poolwater chemistry. If the fuel assembly (FA) contains operational cladding defects, it can also be stored in pools for extended periods. Handling of defective fuel assemblies has also been accomplished without problems. If defective fuel is stored without being first placed in canisters, a poolwater purification system with effective caesium removal capability is needed to maintain acceptable low pool water activity levels. Otherwise placement in canisters is preferable. Cladding defects caused by storage have not been observed so far, and in-reactor cladding defects have not increased [4]. It has been recently reported that initial signs of UO_2 oxidation/hydration have been observed after 21 years wet storage of intentionally defected CANDU spent fuel; however no change to fuel rod integrity was identified [4].

Although only limited quantities of stainless steel (SS) clad fuel[2] exist, this has also behaved satisfactorily to date in pool storage [5].

To date nuclear power in the USSR is based on WWER-440, WWER-1000, RBMK-1000 and RBMK-1500 reactor types with Zr-1%Nb claddings. 46 power units are working with spent fuel arisings of 1300 t/year. The concept of spent fuel management adopted nowadays is to store fuel in at-reactor (AR) or away-from-reactor (AFR) facilities for 3-10 years before shipment for reprocessing or long term storage. Wet storage is the preferred mode and will evidently remain as such during the next 20-30 years due to its high reliability and efficiency. There are two basic types of pools currently in use: AR pools receiving fuel directly from the reactor for cooling during a three year period, and pools, usually of AFR type, for interim storage during 10 to 30 years. Defective fuel is stored in sealed individual cans or in baskets placed at the pool bottom (WWER fuel) or in cans hanged from the ceiling (RBMK fuel). Long term operation experience during 18 years shows high corrosion resistance of intact fuel. Moreover, no further deterioration of defective fuel has been observed after several years of pool storage.

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[1] Includes CANDU fuel.

[2] Stainless steel clad fuel is used in pressurized water reactors.

TABLE IV. SUMMARY OF THE MOST SIGNIFICANT RESIDENCE TIMES OF SPENT FUEL AND FUEL POOL COMPONENTS
(Based on Technical Reports Series No. 290, partially updated in March 1991)

Fuel or component	Environment	Date of first pool exposure	Storage status February 1986	Burnup and remarks
		Longest spent fuel storage		
Zircaloy-2 clad fuel (Shippingport)[a]	Deionized water	1959	Continuing	4 000 MW·d/t U
Zircaloy-4 clad fuel (H.B. Robinson)	Boric acid	1973	Continuing	17 500 MW·d/t U
Stainless steel clad fuel (Sellafield)	Deionized water	1969	Continuing	32 600 MW·d/t U
Stainless steel clad fuel (Conn. Yankee)	Boric acid	1967	Continuing[b]	18 800 MW·d/t U
Aluminium clad fuel (JRR-3)	Deionized water	1970	[c]	1 200 MW·d/t U
		Highest burnup of spent fuel[d]		
Zircaloy clad fuel (Zion)	Boric acid	1982	Continuing	55 000 MW·d/t U
Zircaloy clad fuel (Shippingport)[e]	Deionized water	1974	Continuing	41 000 MW·d/t U
Zircaloy clad fuel (Obrigheim)	Boric acid	1975	[f]	39 000 MW·d/t U
Stainless steel clad fuel (BR-3)	Deionized water	1969	Continuing	32 000 MW·d/t U
Stainless steel clad fuel (Conn. Yankee)	Boric acid	1978	Continuing	37 000 MW·d/t U
Aluminium clad fuel (HFR)	Deionized water	1978	Continuing	550 GW·d/t U
Zr+1%Nb clad fuel (Leningrad NPP)	Deionized water	1973	Continuing	14–27 GW·d/t U
		Longest residence time of pool components[g]		
Aluminium alloys (JEN)	Deionized water	1959	Continuing	Fuel storage rack
Aluminium alloys (Yankee Rowe)	Boric acid[h]	1962	Removed 1979	Fuel storage rack
Stainless steel (NRX)	Deionized water	1956	Continuing	Fuel handling equipment
Stainless steel (Yankee Rowe)	Boric acid[h]	1962	Continuing	Piping, fuel machine
Carbon steel (painted) (Halden)	Deionized water	1959	Continuing	Pool liner
Titanium (Studsvik)	Deionized water	1973	Continuing	Heat exchanger

[a] This fuel is now stored (March 1991) at Savannah River or INEL, Idaho.
[b] Dry storage after 1986.
[c] Removed to dry storage in 1982.
[d] Fuel with burnups up to 55 000 MW·d/t has been stored for ~3 years with no indication of storage or handling problems.
[e] In-reactor exposure from 1957 to 1974; 12.3 years were at reactor operating conditions.
[f] Removed for reprocessing in 1982.
[g] Some operating spent fuel pools have components that are older than those mentioned; however, material compositions are not available for components in some older pools.
[h] The maximum of boron in spent fuel pools during component residence was 800 ppm.

2.1.2. Magnox fuel

Magnox fuel is reprocessed and therefore is not, usually, stored for long periods. In general, satisfactory spent Magnox fuel performance has been reported. The storage performance is closely related to the pH value of the pool water and the chloride and sulphate impurities. Fuel has been stored satisfactorily for up to about 5 years with a pH value of ~11.5 and (Cl^-+SO_4^{2-}) < 1 mg/kg. Raising the pH value to 13, with low anion levels, has provided even better storage performance. Fuel handling problems do not occur from storage of operationally defective Magnox fuel, although fission product and actinide removal from pool water may be required. Sometimes, however, Magnox fuel develops

TABLE V. SUMMARY OF SPENT FUEL EXAMINATIONS TO DEFINE STATUS OF POOL STORED FUEL
(Based on Technical Reports Series No. 290, partially updated in March 1991)

Fuel type/reactor	Fuel: where inspected/ by whom	Fuel characteristics		When examined	How examined	Remarks
		Burnup (MW·d/t U)	First water storage			
Zircaloy/UO_2/KWO[a]/PWR	KWO pool/KWU	up to 39 000	1974	1975/77/ 80/82[f]	NDT/eddy current/ profilometry/ visual	No evidence that reactor induced defects are changing; no evidence that intact cladding is degrading
Zircaloy/UO_2[b]/PWR	Sellafield/UKAEA/BNFL	33 000	1972	1977	NDT/hot cell	No evidence of pool induced degradation
Zircaloy/UO_2[b]/BWR	Sellafield/UKAEA/BNFL	20 000	1971	1977	NDT/hot cell	No evidence of pool induced degradation
Zircaloy/UO_2/SGHWR	Sellafield/UKAEA/BNFL	1 900	1968	1977	Hot cell	No significant degradation at reactor induced defects
SS/UO_2[b]/PWR	Sellafield/UKAEA/BNFL	28 700	1973	1977/78	NDT/hot cell	No evidence of pool induced degradation
Zircaloy/UO_2[b]/PHWR	Sellafield/UKAEA/BNFL	6 500	1966	1977	NDT/hot cell	No evidence of pool induced degradation
Zr+1%Nb/UO_2/RBMK[c]	Leningrad/NPP	14 000	1974	1975/79	NDT/hot cell	Contact electrochemical corrosion Zr/SS
Zircaloy/UO_2[b]/water reactor[c]	Sellafield	Various	1969	Several times 1982/89	NDT/hot cell	Generally good condition for defected and intact fuel
Stainless steel/UO_2/AGR[c]	AR/Nuclear Electric (CEGB)	up to 18 000 channel average	1978	Several times 1980/90	NDT/hot cell	Evidence of slight degradation of sensitized cladding

TABLE V. (cont.)

Fuel type/reactor	Fuel: where inspected/ by whom	Fuel characteristics		When examined	How examined	Remarks
		Burnup (MW·d/t U)	First water storage			
Stainless steel/UO_2/AGR[c]	AFR/Sellafield (BNFL)	up to 18 000	1980	Several times 1982/90	NDT/hot cell	pH7 satisfactory, limited storage with low Cl^- level. pH11, good corrosion protection
Zircaloy/UO_2/NRU NRX/Douglas Pt./NPD[d]	Chalk River/AECL	up to approx. 8 000	1962	1978	NDT/hot cell	No evidence of pool induced degradation
Zircaloy/UO_2/ Shippingport[e]/PWR	Battelle/PNL/BCL	4 800	1959	1980[g]	NDT/hot cell	No evidence of pool induced degradation
SS/UO_2/PWR Conn. Yankee[e]	Battelle/PNL/BCL	32 000	1974	1980[h]	NDT/hot cell	No evidence of pool induced degradation
Zircaloy/UO_2/PWR	Oconee/B&W/PNL	12 100 and 23 100–26 300	1974	1975, 1978 and 1982	Visual/photo & scan	No evidence that test reactor induced defects are changing; crud appears to be loosening

[a] 28 rods are periodically examined: 10 with reactor induced defects, 18 intact; KWO: Obrigheim reactor, Federal Republic of Germany.
[b] Proprietary.
[c] Updated March 1991.
[d] Approximately 140 rods are selected for the Canadian surveillance programme, to be examined periodically up to the year 2000.
[e] Fuel examined under USDOE programme; to be placed in water storage for extended surveillance.
[f] Final inspection.
[g] Fuel stored at Savannah River Plant; visual surveillance.
[h] Fuel stored at Battelle Columbus Laboratory; visual surveillance; shiped to INEL in 1986.

cladding defects by pitting corrosion during storage. Such defects require fission product and actinide removal from storage water, but otherwise do not cause handling problems, provided the fuel is kept wet [6].

2.1.3. Aluminium clad fuel

Good performance experience of spent aluminium clad fuel has been reported with storage periods now approaching 24 years in 1991 (Argentina, Hungary, Italy). Near neutral pH and low ionic concentrations are beneficial for cladding integrity. By extending storage time to 30 years, there is some risk of cladding defects by pitting corrosion. Electrochemical conditions, e.g. the concentration and the concentration gradient of corrosion products, are parameters of influence. Such factors are also valid for in-reactor defective fuel, when stored in a pool.

2.1.4. AGR fuel

A proportion of higher burnup AGR fuel cladding can undergo radiation induced sensitization in the reactor, which makes the cladding susceptible to corrosion attack during wet storage. AGR fuel discharged to date will be reprocessed after several years of wet storage. Spent AGR fuel performance is satisfactory if the pH value of the pool water is about neutral and ($Cl^{-}+SO_4^{2-}$) < 1 mg/kg. Storage performance has been improved by raising the pH to a value of 11.4 or higher. Most of the storage experience has been for neutral pH water, but AFR storage is now exclusively at elevated pH. Defective spent AGR fuel shows little impact on wet storage operations, in that there have been no handling problems, and fission product release is low compared to Magnox fuel. While AGR fuel may develop defects during storage at neutral pH, rod consolidation via dismantling techniques has not been affected by the presence of defective fuel. Pool water purification by filtration and ion exchange is carried out [6].

2.2. SPENT FUEL SURVEILLANCE

2.2.1. Methods of surveillance generally available

The major methods of surveillance in practice worldwide during wet storage of spent fuel include the following:

<u>Pool inspections</u>

* underwater visual and TV examination
* diameter measurements
* eddy current techniques (oxide thickness, crack detection, hydrides)
* periodical analysis of pool water
* electrochemical potential and noise measurements
* γ-scanning of fuel assemblies
* ultrasonic tests
* crack detection by neutron interrogation.

<u>Out of pool inspections</u>

* non-destructive and destructive examinations in hot cells.

2.2.2. Spent fuel surveillance and research programmes within the BEFAST-II CRP

The programmes have involved both undefected and defected fuel. In some cases, the defects in the cladding have been intentionally produced. The most common method of identifying the presence of fuel defects has been by periodic analysis of the pool water to identify fission products. This has the advantage that all the fuel in the pool is monitored whereas examination in hot cells can only involve a small proportion of the fuel.

Neutron radiography has been used by Canada for the determination of water inside fuel pins [4].

Visual underwater examinations are carried out by many countries including the UK, Argentina, Canada, Finland, the Rep. of Korea, the USSR, Hungary and the USA. Usually, cameras or modified telescopes are used, and permanent records are obtained via still photography or videotapes.

Radiation-resistant equipment has been developed and colour recordings have been made (e.g. in the UK).

In addition, hot cell visual examinations and other tests have been performed by the UK, the Rep. of Korea, the USSR, the USA and Canada and are planned in Argentina. By these means, more detailed examination is possible using a wider range of techniques.

Destructive testing has been used for detailed examination of FAs in Canada, the UK and the Rep. of Korea. Element dismantling can be carried out, followed by more intensive investigation of the fuel, the fuel pins and fuel element structural components. Mechanical tests, such as drop tests, may be performed. It is also possible to leak-test pins for cladding integrity using specially developed equipment. Metallography of cladding and ceramography of fuel pellets has been undertaken.

Periodic analysis of pool water has been carried out by all participating countries. It is generally recognized that satisfactory long term spent fuel integrity during wet storage depends on good control of the poolwater conditions, especially pH, temperature, conductivity and anion content, e.g. Cl^-. The presence of fission products can be a very sensitive guide to fuel integrity in the pool.

Electrochemical measurements have been carried out in Hungary, both in the research reactor spent fuel pool and under laboratory conditions. Beyond conventional techniques, like electrode impedance, electrolyte conductivity and cyclic voltametry, the fluctuations of electrochemical potential at zero current have been determined and their frequency spectra analysed by numerical methods of fast Fourier transformation, maximum entropy and range-per-scatter. Characteristic spectral behaviour for general and pitting corrosion has been recorded. Pitting has been seen to be the main mode of corrosion of aluminium cladding in the spent fuel pool of the research reactor.

The γ-scanning technique has been used by Canada for fuel degradation studies [4]. Argentina has also utilized this method for burnup assessments of research reactor fuel. Traces from γ-scans could indicate the migration of fission products in spent fuel rods, although migration has never been observed to date.

Ultrasonic measurements have been used by the Republic of Korea to measure the degree of wetting of fuel pellets around deliberately defected positions on Zircaloy fuel rods.

Ultrasonic signals are reflected from fuel rod surfaces and are plotted against the distance from the defect. Changes in the plots indicate the degree of water penetration in the fuel pellet.

Cladding crack detection has been measured by neutron interrogation in Argentina. In this test a fuel rod was irradiated by neutrons in a closed, pumped-water loop. Fission products escaping from the fuel rod were concentrated in an ion exchange medium, which was then analyzed by γ-spectroscopy. Thus, any contamination levels above normal surface activity give information on fuel pin integrity.

2.2.3. Results of the BEFAST-II CRP

Underwater visual examinations have shown insignificant degradation in PWR, BWR and CANDU Zircaloy clad fuels during pool storage. Regions of PWR top-nozzle/grid tubes in particular have been inspected periodically for crevice corrosion and shown to be satisfactory. Some instances of water-level marks on LWR fuel pins were considered to relate to possible temporary exposure of bottled fuel to air above the water level. Some instances of minor damage have been recorded; these were probably caused during handling. Magnox fuel stored out of containers in non-optimum environments has been shown to incur some localized corrosion attack with the presence of sludge in the skip (basket). Examination of non-containerized AGR fuel stored at neutral pH have revealed small areas of corrosion deposit on structural components and on top fuel-pin end caps.

Visual examinations in hot cells have revealed no degradation of intact and deliberately defected PWR pins after 30 months of wet storage (Republic of Korea). In the UK, AGR fuel has been inspected in hot cells; some graphite sleeve chipping and small amounts of corrosion deposit were observed at crevice positions on fuel stored in neutral conditions. AGR fuel, stored at elevated pH, has been shown to be in very good condition.

Extensive destructive examinations of spent fuel after extended pool storage have been carried out in Canada, the UK and the Republic of Korea. CANDU fuel elements were tested by neutron radiography, fission gas analyses, diameter of intact and defected elements, cladding ring tensile tests, cladding metallography, ceramography, cladding H_2/D_2 analyses; and, in the defected fuel elements by X ray photoelectron spectroscopy, scanning electron microscopy and X ray diffraction and ceramography of the UO_2 . Evidence of UO_2 oxidation/hydriding was observed in an intentionally defected CANDU fuel element stored under water for 21 years; however, the resulting oxidation/hydriding had no apparent effect on the integrity of the fuel element. Intact CANDU spent fuel showed insignificant changes after 27 years of wet storage [4].

AGR fuel in the UK, stored at elevated pH, was shown to be in very good condition after two years of storage. Magnox fuel, after prolonged storage in non-optimum environments, has suffered from localized cladding attack.

In the Republic of Korea, PWR fuel was shown by metallography to be in good condition; white deposits at limited regions around defect positions were found and their structures have not been identified but may possibly be due to caesium hydroxide.

In general, periodic analysis of pool water has confirmed the need for tight chemical control to avoid cladding corrosion. Aggressive anions (e.g. Cl^-) should be kept as low as possible: typically < 0.3 mg/kg Cl^-. Elevated pH above 11.3 with (Cl^- + SO_4^{2-}) < 0.2 mg/kg has been demonstrated to be beneficial for Magnox fuel storage. Similarly, high pH is desirable for AGR fuel storage. LWR and MTR fuel (Japan) has been satisfactorily stored in the averaged pH of 5.5, Zircaloy and aluminium clad elements have shown very good performance at neutral conditions.

In Hungary, visual observation of aluminium cladding showed serious attack, having the appearance of spread out local corrosion regions, particulary with storage times longer than some 20 years. Electrochemical noise spectra indicated pitting corrosion to be the main mode of attack. The reason for this process was considered to be the inhomogeneous build up of corrosion deposits in the pool water. Although the poolwater activity has been maintained at a very low level, the damaged bundles are now being bottled.

Ultrasonic tests by the Republic of Korea suggested that, in deliberately defected PWR fuel, the distance of water ingress in the fuel pellet was from 1.3 cm to 8 cm after 1 hour and 19 days, respectively. Caesium release increased rapidly over the first 30 days but then increased more slowly up to 90 days, then remained nearly constant. The dissolution rate of Co-60 was constant with about 0.093 $Bq\cdot sec^{-1}\cdot rod^{-1}$ which was consistent with the reported value of 0.1 $Bq\cdot sec^{-1}\cdot rod^{-1}$ [7].

By neutron interrogation methods it was found that, after 15 years of storage in demineralized water, the cladding of Al clad MTR fuel was sound. No change was observed in the cladding behaviour by measurements taken over the last seven years.

Discussion

The results from the surveillance programmes in the CRP support the existing data from earlier investigations reported in the literature. Major investigation programmes had been performed earlier in Canada [8], the Federal Republic of Germany, Switzerland, the USA and France. Nondestructive and destructive investigation in pools and hot cells had shown that intact and in-reactor defected zirconium alloy clad fuel did not experience any significant detectable changes. The investigation methods used in the CRP activities are in part identical to those used in earlier investigations (e.g. diameter measurements of fuel rods, fission gas analysis, neutron radiography, cladding ring tensile tests, cladding H_2/D_2 analysis, cladding metallography and ceramography), while others are complementary (e.g. gamma scanning, X ray photoelectron spectroscopy, X ray diffraction, electrochemical methods and scanning electron microscopy).

The role of water chemistry was recognized very soon resulting in clear specifications for the control of poolwater conditions. The beneficial influence of increased pH values for pool storage of Magnox and AGR fuel has recently been confirmed.

Fission product release results obtained in the Republic of Korea correspond to earlier results reported from Canada, UK and the Federal Republic of Germany [9-18]. It was confirmed that the rapid release of caesium from intentionally defected PWR spent fuel rods occurs over the first few months of water pool storage and is then followed by very slight release.

It has been found that specified poolwater temperature control is essential for long term spent fuel integrity and to avoid structural damage to the facility. In addition, poolwater radioactivity levels have been reduced by temperature cycling (USSR).

Generally speaking the CRP results strengthen existing data. In part the reported results identify possibilities for better approaches in pool storage modes or improved reliability in long term storage performance prediction.

2.3. WET STORAGE FACILITIES AND OPERATION

2.3.1. Work reported on facilities within the BEFAST-II CRP

Water chemistry

Evaluation of purification (i.e. filtration and ion exchange) equipment and its optimization have been carried out in Argentina, Canada [19], the UK, the USA and the USSR.

Optimization of water chemistry parameters (i.e. pH, conductivity, chloride and fluoride contents) has been carried out in Argentina, Canada [19], Finland, the UK, the USA and the USSR.

Design

At-reactor water pools have been designed by most participating countries.

Design of AFR water pools has been carried out by the German Democratic Republic, Finland, the UK, the USA and the USSR.

AFR storage facilities are used in the German Democratic Republic, Italy, the UK, and are under construction in Argentina. In Finland, Japan and the USA the AFR storage does not represent buffer storage. Some countries plan to construct AFR storage pools (Rep. of Korea, UK, USSR).

Fuel handling equipment design and optimization has been carried out in Canada [20], Finland, Italy, Japan, the Rep. of Korea, the UK and the USA.

Design and installation of higher storage density containers/racks have been completed in Canada, the Federal Republic of Germany, Italy, Hungary, the UK and the USA.

Rod consolidation is carried out by the UK and remains a future option for the USA.

Materials

Alternative liner materials have been studied in Argentina, Canada, the UK, the USA, and the USSR. Stainless steel and epoxy are the materials most commonly used for storage pools. Effect of radiation dose and water quality on pool liner degradation has been studied in Argentina, Canada, Hungary, the USA and the USSR. An epoxy liner suffers radiation damage after 10^{10} to 10^{11} rad[3] radiation exposure (Canada). Underwater curing epoxy repair techniques have been developed and used in Canada.

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[3] 1 rad = 1.00×10^{-2} Gy.

For consolidated spent fuel elements, neutron absorbing construction materials (e.g. boron steel) may be used as the structural material of the pool racks (UK, USSR, USA).

2.3.2. Operational experience

Experience to date suggest that Zircaloy clad water reactor spent fuel could be stored safely for at least 50 years [4] and then transported to a reprocessing plant or disposal site (Argentina, Canada, Italy and USA). The effect of poolwater quality has been studied. Standards for storage pool water quality have been set up by all countries. With effective monitoring and control procedures, poolwater chemistry specifications have been generally maintained. External radiation doses to pool operators are within the prescribed limits, thus no special dose reduction activities are needed. With specified water quality, the effect of radiation on SS storage pool liners was found to be negligible (Argentina, Canada, Finland, Hungary, UK, USA and USSR).

High density fuel storage racks are used in many countries (Canada, Finland, Hungary, the Federal Republic of Germany, Italy, Republic of Korea, Sweden, USA and USSR) as well as reracking.

Rod consolidation is under way in the UK (AGR fuel), and has been evaluated for LWR fuel in the USA. The UK AGR fuel dismantler offers a 3:1 compaction factor for the fuel rods, whereas in the USA a 2:1 consolidation of LWR rods was achieved. Non-fuel bearing component handling and management require further development before rod consolidation technology can be considered complete (USA). This waste is stored in a separate building (UK).

For evaluation of a change of the storage mode from wet to dry, the spent fuel storage criteria that have to be considered are type of fuel, cost, initial fuel enrichment, fuel burn up, duration of wet storage and fuel integrity.

Wet-dry mode change has been demonstrated in some countries (Canada [19], the Federal Republic of Germany and the USA). In the USA and the the Federal Republic of Germany, this mode - as a result of successful demonstrations - will be the normal operating mode for spent fuel transportation. In Italy and the UK it has been demonstrated to be feasible. The wet-dry-wet mode has been demonstrated in the the Federal Republic of Germany and Canada.

Experience in the management of damaged fuel has been accumulated in most BEFAST-II participating countries.

2.3.3. Handling and transfer facilities

On-site spent fuel handling and transfer facilities are designed so that the fuel is not damaged and fuel integrity is maintained. Positive experience with such equipment is available in all countries with nuclear programmes. Within the CRP, good performance is reported by most participating countries.

Experience with the handling of on-site transfer casks and/or off-site transportation casks has been accumulated in most countries. Experience with such casks includes receipt at the spent fuel pool, in-pool fuel loading and/or unloading, draining and closure, and decontamination of the casks and transfer on-site, or transshipment off-site.

2.4. SUMMARY AND CONCLUSIONS

- Extended wet storage of zirconium alloy clad spent fuel with no loss of cladding or fuel pellet integrity is feasible even if the cladding has through-wall operational defects.

- Magnox and AGR spent fuel storage times are relatively short because currently the fuel is or will be reprocessed after a few years of wet storage. To maintain Magnox and AGR spent fuel cladding integrity, close chemical control of pool water is necessary.

- Monitoring and surveillance have confirmed that adherence to the specified pool water chemistry is essential to prevent fuel degradation during pool storage for all types of spent fuel. Whereas near-neutral pH conditions with low ion content are satisfactory for zirconium alloy clad fuels, elevated pH is beneficial for Magnox and AGR fuels. For aluminium clad fuels, a pH of ~5.5 is in use.

- Adherence to specified poolwater temperatures is required to maintain fuel assembly integrity, minimize water radioactivity levels and preserve the pool structural integrity.

- For extended wet storage (over 50 years), certain water pool liner materials, such as non-stabilized stainless steel and epoxies, may require further investigation.

- In addition to the standard pool inspection techniques and hot cell inspections, some advanced monitoring techniques such as γ-scanning, neutron interrogation, ultrasonic inspection and electrochemical noise analysis are now being employed.

- Wet storage will continue to be the main storage option in all participating countries. For the long term, most participating countries are investigating dry storage options.

- Enhancement of wet storage capacity will remain an important activity. Rod consolidation to increase wet storage capacity will continue in the UK and is being evaluated for LWR fuel in the USA, and may start in some other countries. High density storage racks have been successfully introduced in many existing pools and are planned for future facilities.

- For extremely long wet storage (≥50 years), there is a need to continue work on fuel integrity investigations and LWR fuel performance modelling. It might be that pool component performance in some cases could be more limiting than the FA storage performance. It is desirable to make concerted efforts in the field of corrosion monitoring and prediction of fuel cladding and pool component behaviour in order to maintain good experience of wet storage.

3. DRY STORAGE

This chapter discusses:

(a) Dry storage performance,
(b) Dry storage surveillance,
(c) Dry storage facilities and operation, and
(d) Dry storage summary and conclusions.

3.1. SPENT FUEL DRY STORAGE PERFORMANCE

General storage experience is given in Table VI [5].

3.1.1. Zirconium alloy and stainless steel clad LWR fuel

Storage of Zircaloy clad spent LWR fuel in an inert gas atmosphere - preferably helium - is a proven technology up to FA hot-spot temperatures of about 450°C. Dry storage of that fuel is licensed in the Federal Republic of Germany for temperatures $\leq$410°C and in the USA for $\leq$380°C. To optimise compliance with these temperature limits, the decay heat when the fuel is loaded must be considered, and well verified heat transfer codes for precise cladding temperature prediction are beneficial (German Democratic Republic). Positive experience with dry storage is available for storage periods of more than a decade. Licensed storage periods of at least 20 years or longer are available. The storage of operationally defective Zircaloy clad spent fuel in an inert atmosphere is without consequences. The effects of low levels of oxygen have been studied (Japan, UK).

There is continuing research in Japan and the USSR into mechanisms of defect production during storage. In Japan, creep tests are being performed on unirradiated and irradiated Zircaloy cladding for assessment of the effect of internal gas pressure on cladding integrity. In the USSR, stress corrosion cracking experiments on unirradiated Zr-1%Nb cladding are being used to verify code calculations; from these calculations a maximum temperature of 350°C has been derived for inert gas storage of spent fuel with this cladding (USSR).

Pessimistic evaluation of the storage of Zircaloy clad spent fuel in oxidizing atmospheres has also being done. Defect-free fuel should have adequate storage performance at temperatures estimated to be less than 330°C [21]. However, the possibility of propagation of defects in operationally defective fuel limits the maximum storage temperature. There are research programmes in Canada, Japan, the UK and the USA on UO_2 oxidation behaviour. In January 1991 the USA reported that an intentionally defected rod segment failed by splitting after 23 000 hours in air at 150 °C [22]. Definition of the maximum storage temperature in oxidising atmospheres must consider the fuel irradiation history, cooling time and the proposed storage period.

For SS clad spent LWR fuel, there is no dry storage experience available. Due to the small amount of such fuel, no related R&D work is intended.

3.1.2. CANDU fuel

The dry storage of spent CANDU fuel in air is a proven technology. To date the dry storage of 455 MTU spent CANDU fuel is licensed at four reactor sites. A fifth reactor site,

TABLE VI. SPENT FUEL BEHAVIOUR PRIOR TO AND DURING DRY STORAGE (Based on Technical Reports Series No. 290, partially updated in March 1991)

Country	Storage facility	Organiza-tion/site	Fuel history			Inspection prior to storage				Inspection during storage			
			Type	Burnup (MW·d/t)	Wet storage period	Date	No. of assemblies	Type of test	Observa-tions/results	Date	No. of assemblies	Type of test	Observa-tions/results
Canada[a]	Concrete canister expts.	AECL/OH Whiteshell											
	ERB	"	CANDU/ Pickering	7 270	~3 a	1978	2[b]	Visual/photos, profilometry, fission gas, gamma scans,	Type CANDU No crud	1982	2[b]	Same as pre-storage	No changes from pre-storage exam
	ERB	"	WR-1 (OCR) UC	10 000	1.2 a	1978	3[b]	mechanical tests, metallo-graphy,	Fixed crud	—			
	ERB	"	WR-1 (OCR) UO_2	6 800	~2 a	1978	10[b]	ceramography	Fixed crud	—			
	CEX-1	"	CANDU/ Pickering	8 000	5–8 a	1980	4[b]	As above plus XPS, XRD, SEM on fuel. Intentional defects in 2 Pickering and	Type CANDU No crud	1994 1989	4[b] 1[b]	"	Undefected fuel, no change. Defected fuel-bulk UO_2 oxi-dation. No fuel swelling.
	CEX-1	"	CANDU/ Bruce	7 500	~3 a	1980	4[b]	2 Bruce bundels	Type CANDU No crud	1984	4[b]	"	As above.
	CEX-2	"	CANDU/ Pickering	8 700	5–9 a	1982	4[b]	As above	Type CANDU No crud	1984 1988	4[b] 1[b]	"	Undefected fuel, no change. Defected fuel-grain boundary oxidation. No fuel swelling.
	CEX-2	"	CANDU/ Bruce	8 000	2–4 a	1982	4[b]	As above	As above	1984 1988	4[b] 1[b]	" "	As above. As above.

TABLE VI. (cont.)

Country	Storage facility	Organiza-tion/site	Fuel history			Inspection prior to storage				Inspection during storage			
			Type	Burnup (MW·d/t)	Wet storage period	Date	No. of assemblies	Type of test	Observa-tions/results	Date	No. of assemblies	Type of test	Observa-tions/results
Canada[a] (cont.)	Concrete canister demonstrations	AECL/ Whiteshell	WR-1 (OCR) UO_2 AECL/ Douglas Pt.	5 000 <8 000	<5 a <5 a	1975 1976	138[b] 360[b]	Visual	Fixed crud Type CANDU	Periodical	—	Monitor air + radiation fields	No problems
	Concrete canisters production	AECL/ Whiteshell	WR-1 (OCR) UO_2	5 500	~0.5–25 a	1978–	565[b]	Visual	Fixed crud	"	—	"	"
			WR-1 UC	8 600	1.5–25 a	1990	721[b]	Visual	Fixed crud	"	—	"	"
			WR-1 UC, UO_2	~7 000	~0.5–25 a	1978–1990	237[c]	Visual	Fixed crud	"	—	"	"
	Concrete canisters commercial	Hydro Quebec/ Gentilly	CANDU/ (BLWR) Gentilly-1	2 255	~7a	1985	67 t U	Visual	Loose deposit	"	—	"	"
	Concrete canisters commercial	AECL/ Douglas Pt.	CANDU/ Douglas Pt.	7 700	3–20 a	1987	300 t U	Visual	Type CANDU	"	—	"	"
	Concrete canisters commercial	AECL/ Chalk River	CANDU/ NPD	~6 000	2–25 a	1989–1990	65 t U	Visual	Loose deposit	"		"	"
	Concrete canisters commercial	New Brunswick Power/ Pt. Lepreau	CANDU-6/ Pt. Lepreau	7 800	6–10 a	1991–ongoing	150 t U	Visual	Type CANDU	"	—	"	Start loading June 1991
	Dry storage container	Ontario Hydro Pickering-A	CANDU/ Pickering	~8 500	6–10 a	1988, 1989	15 t U	Visual	"	"	—	"	No problems

TABLE VI. (cont.)

Country	Storage facility	Organization/site	Fuel history: Type	Fuel history: Burnup (MW·d/t)	Fuel history: Wet storage period	Inspection prior to storage: Date	Inspection prior to storage: No. of assemblies	Inspection prior to storage: Type of test	Inspection prior to storage: Observations/results	Inspection during storage: Date	Inspection during storage: No. of assemblies	Inspection during storage: Type of test	Inspection during storage: Observations/results
Germany, Fed. Rep. of	Metal cask Gorleben	DWK	PWR, BWR		1 a	—	1500 t U	Sipping	—	—	1500 t U	Ventilation monitoring, seal monitoring	—
	Metal cask Castor Ic	DWK, Würgassen	BWR Würgassen	28 000	1.3 a	1982	16	Sipping, visual of all assemblies	Typical spent LWR fuel	Continuous Periodical	16	Temperature, pressure recording, cover gas sampling	3 a storage, behaviour as predicted
	Metal cask Castor Ia	DWK, Biblis, Jülich	PWR Biblis	35 000	1.3 a	1983	4	Profilometry, oxide layer, eddy current at individual rods	No leakers	End of storage	4	Profilometry, oxide layer	3 a storage ongoing, behaviour as predicted
	Metal cask Castor Ib	DWK, Stade, Karlsruhe	PWR Stade	34 000	0.8 a	1982	4				4	Eddy current at individual rods	2 a storage, behaviour as predicted
	Metal cask TN 1300	DWK, Biblis	PWR Biblis	2 000 –39 000	0.3 a –2.7 a	1984	12	Sipping, visual	No leakers	Continuous	12	Temperature, pressure recording	
Switzerland	Metal cask Castor/Diorit	EIR, Würenlingen	HWR Diorit	8700 (average)	6–8 a	1983	350	Visual, dose measurement	Cladding intact	Continuous	350	Leak detection of cask. Temperature of cask and gamma dose	No leaks found

TABLE VI. (cont.)

Country	Storage facility	Organization/site	Fuel history: Type	Fuel history: Burnup (MW·d/t)	Fuel history: Wet storage period	Inspection prior to storage: Date	Inspection prior to storage: No. of assemblies	Inspection prior to storage: Type of test	Inspection prior to storage: Observations/results	Inspection during storage: Date	Inspection during storage: No. of assemblies	Inspection during storage: Type of test	Inspection during storage: Observations/results
United Kingdom[a]	Vault (CO_2)	Nuclear Electric/ Wylfa	Magnox	~5 000	0	Burst can detection facility on reactor. TV facility				(Short term storage only)			
	Vault (air)	Nuclear Electric/ Wylfa	Magnox	~5 000	0	Decay heat check before loading (from CO_2 vault)				Activity monitoring in storage, visual inspection on removal			
USA[a]	Concrete silo EMAD	Nevada test site		27 000	3–4 a	1978	1	1 assembly taken to BCL for destr. examination 5 assemblies NDE at BCL	1 leaker. Dark crud. Some spallation during hot test			Cover gas analysis, visual, photos	1 failed in storage 1 initial leaker. Assemblies retrieved. No significant problems. Transferred to INEL in 1986.
	Surface dry well	Nevada test site	Turkey Point fuel PWR	27 000	3–4 a	1979	4						
	Vault	Nevada test site		27 000	3–4 a	1979	1						
	Deep dry well Climax	Nevada test side			3–4 a	1980	11			1983	11	Cover gas analysis, visual photos	Included in in EMAD fuel
	Metal cask GNSI CASTOR V/21, TN-24P, Westinghouse MC10		PWR (Surry)	29 000	2–4 a 2–4 a 2–4 a	1986 1985 1986	Total 168	Sipping, visual, gamma scan, FFDS (Ultrasonic)	No leakers. Hard dark crud heavier toward top of rods	2 month test	21 24	Cover gas analysis, visual, photos, clad temperature, gas pressure	Cracks in basket. No effect on fuel. Good heat transfer and shielding performance. Further results in progress

TABLE VI. (cont.)

Country	Storage facility	Organiza-tion/site	Fuel history: Type	Fuel history: Burnup (MW·d/t)	Fuel history: Wet storage period	Inspection prior to storage: Date	Inspection prior to storage: No. of assemblies	Inspection prior to storage: Type of test	Inspection prior to storage: Observa-tions/results	Inspection during storage: Date	Inspection during storage: No. of assemblies	Inspection during storage: Type of test	Inspection during storage: Observa-tions/results
USA[a] (cont.)	Dry well field. SS canister, metal seal	INEL	LMBR Shipping-port UO_2–THO_2		3 a	1985	2 (48 total)	Sipping	No leakers found. Very little crud	— Continuous	2 (48)	Canister gas not monitored. Well and soil monitoring	Results in progress
	Horizontal concrete silo[d] (NUHOMS)	CP&L/ USDOE/ EPRI	PWR H.B. Robinson	35 000 (max)	5 a (min)	1989							
	Horizontal concrete silo[c] (NUHOMS)	Duke Power	Oconee	35 000 (max)	5 a (min)	1990	144						
	Metal cask REA-2023	USDOE/ GE Morris	BWR Cooper		2–3 a	1985	52	Visual, sipping, calorimetry, gamma scan	One leaker. Red/brown crud. Evidence of spalling	4 1/2 months	52	Cover gas analysis, clad temperature, gas pressure	Leaker detected. No effects on fuel handling. Returned to wet storage
USSR[a]	Steel cask TK 13	South Ukrainian NPP	VVER 1000	27 000	800 d	1988	12	Visual, sipping	No leakers	1989	12	Clad temperature	No leakers

TABLE VI. (cont.)

Country	Storage facility	Organization/site	Fuel history: Type	Fuel history: Burnup (MW·d/t)	Fuel history: Wet storage period	Inspection prior to storage: Date	Inspection prior to storage: No. of assemblies	Inspection prior to storage: Type of test	Inspection prior to storage: Observations/results	Inspection during storage: Date	Inspection during storage: No. of assemblies	Inspection during storage: Type of test	Inspection during storage: Observations/results
France	Vault TOR 5 cans/well		Phenix fast breeder	40 000	2 a	1986	385 Cans (9.3 U+Pu)	Visual after dismantling, leak of cans		—	385 Cans (9.3U+Pu)	Filtering and monitoring of cooling air	—
	Vault Cadarache arrangement	Cadarache	EL-4 (Brennilis) and Osiris	15 000 20 000 25 000–32 000	3 a 2.5 a	1988	5400 902	Monitoring transport cask to detect failed fuel		Intermittent	5400 902	Well gas sampling	—

[a] Updated March 1991.

[b] Fuel bundles.

[c] Loose elements stored in element storage cans.

[d] 8 NUHOMS containers, each holding 7 elements.

[e] 6 NUHOMS containers, each holding 24 elements.

Pt. Lepreau, is presently in the process of obtaining a license to load 150 MTU by the end of 1991. It is predicted that intact and defected spent CANDU fuel, if ≥10 years out-reactor, would experience no additional loss of cladding integrity for at least 100 years storage in dry 100°C air, based on laboratory data and experimental results. The Canadian dry storage experience is described in Refs [23 - 25]. The results from the latest dry storage examinations (intentionally defected CANDU fuel stored for about 6 years in moisture saturated air at 150 °C and for 8 years in dry air) have been presented at the BEFAST-II meetings in Espoo and Vienna.

3.1.3. Magnox fuel

Storage of spent Magnox fuel is licensed in CO_2 and air (UK) or in N_2 (Italy). Fuel can be stored in CO_2 at temperatures of around 300 °C, whereas the temperature in air is currently limited in normal operation to ≤150°C.

In the UK, positive storage experience is available for time periods of up to about 8 years as the fuel is sent on for reprocessing. Taking the positive experience of air storage into account, several decades of storage is considered to be possible. The storage of operationally defected fuel needs no special precautions. Storage related defects with dry air or CO_2 stored spent Magnox fuel have never been observed.

Italy has accumulated 20 years of experience of storage in N_2, at pool storage temperatures.

3.1.4. Aluminium clad fuel

Only limited experience of inert gas storage of spent Al clad fuel at ambient temperature is available [26, 27].

3.1.5. AGR fuel

No operational dry storage experience is available. In support of a possible AGR fuel dry buffer store, research programmes are investigating long term behaviour of irradiated cladding in radiolysed air atmospheres, irradiated UO_2 oxidation in air [28], and air-inert gas mixtures (UK).

3.2. SPENT FUEL SURVEILLANCE IN DRY STORAGE

3.2.1. Programmes and methods

In the dry storage of spent nuclear fuel, the spent fuel is normally placed in a sealed container, hidden from direct inspection. As a result, the surveillance approach that is employed for dry spent fuel storage is based on first determining an envelope of environmental parameters within which the spent fuel may be safely stored, and then monitoring to ensure that the environment remains within the acceptable envelope. The primary considerations are to ensure that the spent fuel is not damaged, either in the short term, or due to long term cumulative effects, and that any effects on the surrounding environment remain within acceptable limits. For some experimental purposes, samples of fuel are also periodically removed and directly inspected.

For experimental purposes, the parameters that are monitored are the radiation and thermal fields, cover gas composition, and gaseous radioactive material releases to ensure the maintenance of the required gas environment surrounding the fuel, to detect fuel cladding failure, and to confirm the continued structural integrity of the facility. For a typical first-of-a-kind demonstration programme, thermocouples are generally inserted into the fuel assemblies. Thus, the temperature of the fuel can be directly measured as a function of external temperature, and the analytical models used to design the facility may be validated. For subsequent operational installations of the same type of dry storage facility, only external temperature measurements are made, as a point check to ensure that the measured parameter continues to be within the bounds predicted by the analytical model.

Similarly, for first-of-a-kind demonstration installations, adequate radiation measurements are performed to verify that the system performs as predicted. Then, for subsequent operational facilities and routine storage, point checks are made at selected locations near the facility and in its vicinity to ensure that the radiation levels are in compliance with licensed requirements.

For cask storage, measurements are also taken to ensure the continued presence of the required inert cover gas. This is typically accomplished by installing instrumentation to monitor continually the pressure in the annulus between the inner and outer seals on the storage enclosure. Visual examinations are also conducted to ensure that the physical integrity of the facility is not degrading. Other facilities, such as vaults, may also provide cover gas monitoring.

The measures discussed above generally reflect the practice of inert gas storage in the USA and the Federal Republic of Germany. In addition, the BEFAST participants from Canada reported the following supplementary measures. Specifically, surface and subsurface groundwater at the site of their concrete storage canisters is monitored for radioactive material releases. Also, fuel bundles from experimental facilities are periodically removed from storage and subjected to visual and selective destructive examination to verify the continued integrity of the spent fuel. Finally, Canada is installing tubes in the concrete walls in some current and future canisters to allow the insertion of radiation monitors capable of verifying the contents of the canisters (for IAEA safeguards purposes).

3.2.2. Dry storage experience

The experience with all dry spent fuel storage installations operated to date has generally been satisfactory. Canada has 15 years of experience with the storage of spent fuel in concrete canisters. They have experienced no release of radioactive materials to the environment, including verification of no radioactivity in surface and subsurface groundwater during the ten years that it has been monitored. In the last ten years, the temperature of the stored fuel has dropped from 110 °C to 66 °C. In the USA, the first dry storage was initiated in 1984 at the Idaho National Engineering Laboratory, with the first licensed, commercial installation being placed into operation in 1986. There have been no releases of radioactive material from any of the dry storage installations in the USA. Similarly, there have been no operational or maintenance problems. Operational experience has been well within the design limits and regulatory requirements. In the the Federal Republic of Germany licensed storage started in 1982 at-reactor in Wuergassen for demonstration purposes and today two dry storage facilities are licensed.

None of the other BEFAST participating countries who are conducting dry storage operations reported any systematic problems or unexpected performance.

3.3. DRY STORAGE FACILITIES AND OPERATION

Dry storage facilities are operated at-reactor (AR) sites in Canada, the Federal Republic of Germany, the UK (Magnox) and the USA. Dry storage is also being considered in Argentina, Italy, Japan and the UK (AGR fuel). The types of dry storage facilities are concrete canisters in Canada, vaults in the UK, storage/transportation casks in the the Federal Republic of Germany, the USA and the USSR, a horizontal concrete module system (NUHOMS) and metal casks in the USA and drywells (AFR) in Japan.

A description of each type of fuel storage facility, operational experience and fuel transfers is provided in the following sections.

3.3.1. Dry storage facilities and operational experience

3.3.1.1. Concrete canisters

The concrete canister is a passive cooling storage system. It is a hollow, reinforced concrete cylinder constructed in a vertical position on a concrete pad. The concept was developed by Atomic Energy of Canada Limited (AECL). Testing and demonstration of the concept was performed in 1974-1976 at the Whiteshell Laboratories (WL) in Manitoba. In this concept, CANDU fuel bundles are stored in cylindrical steel containers (called baskets) in either a helium or air atmosphere. Neutron and gamma shielding is provided by the concrete structure. Heat is removed by convection and conduction, through the shielding structural material, to the environment. Two containment barriers are provided in this system.

Since the concept was conceived, five concrete canister dry storage facilities have been constructed in Canada. The facilities are located at the WL, the Gentilly-1 Nuclear Reactor Site (G-1), the Douglas Point Nuclear Generating Station (DPNGS), Chalk River Laboratories (CRL) and Point Lepreau Nuclear Generating Station. Fuel loading at Point Lepreau is to begin in June 1991. The location of these facilities and their storage capacities are shown in Table VII. Figure 1 shows the Douglas Point concrete canister dry storage facility.

Hydro Quebec plans to use concrete canisters for the storage of fuel from its 600 MW(e) CANDU Gentilly-2 NPP in the province of Quebec, Canada, starting in 1994 at the rate of 100 MTU per year. AECL has signed an agreement with the Korean Electric Power Corporation (KEPCO) to provide KEPCO with AECL concrete canister dry storage technology. The Republic of Korea is planning to load spent CANDU fuel from its Wolsung-1 NPP into concrete canisters starting in 1992/1993 at the rate of 100 MTU per year.

3.3.1.2. Vault storage

A vault is a reinforced concrete structure containing an array of storage cavities. Each storage cavity can contain one or more spent fuel assemblies. Shielding is provided by the structural material. Primary heat removal is by forced or natural air movement (convection). This heated air is rejected to the atmosphere either directly or via appropriate filtration, dependent on the individual vault design. In the last case, forced-air circulation is necessary. This storage concept permits modular extension.

TABLE VII. CONCRETE CANISTER DRY STORAGE SPENT FUEL FACILITIES IN CANADA

Site	Location	No. Canisters	No. Bundles	Cover gas	MTU
Whiteshell	Manitoba	19	1 900	air or He	24
Gentilly	Quebec	11	3 123	air	67
Douglas Pt.	Ontario	47	22 236	air	300
Chalk River	Ontario	11	4 855	air	65
Pt. Lepreau	New Brunswick	20*	8 100	air	150
				Total:	606

* Only 15 of the 20 canisters are to be loaded with spent fuel in 1991

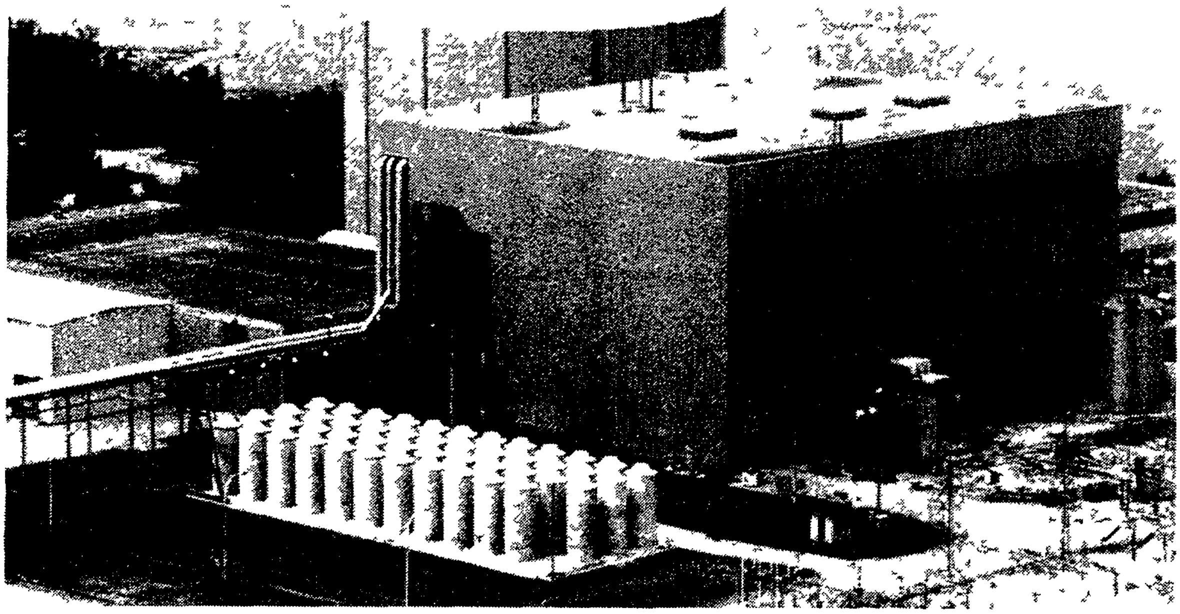


FIG. 1. The Douglas Point Nuclear Generating Station and spent fuel concrete canister site Spent CANDU fuel containing 300 MTU generated during the 18-year lifespan of the station are stored in the canisters.

Modular concrete vault storage in the United Kingdom

Two types of dry storage facility are in use at the Wylfa Magnox power station in North Wales to provide temporary storage for spent fuel before reprocessing. The first type has three vaults, with a capacity of 83 MTU each, which have been operating for approximately 20 years. Fuel elements are placed in tubes filled with carbon dioxide, and the heat generated by the fuel is removed from the exterior surfaces of the tubes by the natural convection of air (Figure 2). No degradation of the fuel has been detected. The maintenance and operation of the facility require very limited personnel support. The second type involves two modular concrete vaults (350 MTU each module) with forced-air cooling systems [29]. The design of this modular concrete vault eliminates the need for the carbon dioxide cover gas. The atmosphere inside the vaults is maintained at a slightly negative pressure by means of an exhaust fan discharging to the atmosphere. A filter system, with redundancy, is provided to ensure that atmospheric contamination is maintained below specified limits. The maximum temperature of the fuel in air is less than 150°C under normal operating conditions. Spent fuel, newly discharged from the reactor, necessarily undergoes a period of storage in a carbon dioxide filled cell for a minimum of 150 days before being transferred to the air-cooled system when the decay heat has decreased to an acceptable value. The first modular concrete vault with the forced-air cooled system has been effectively operated since 1979. The activity levels of the discharged air have been at background levels. Maintenance requirements are minimal, and the radiation exposure of the workers has been very low.

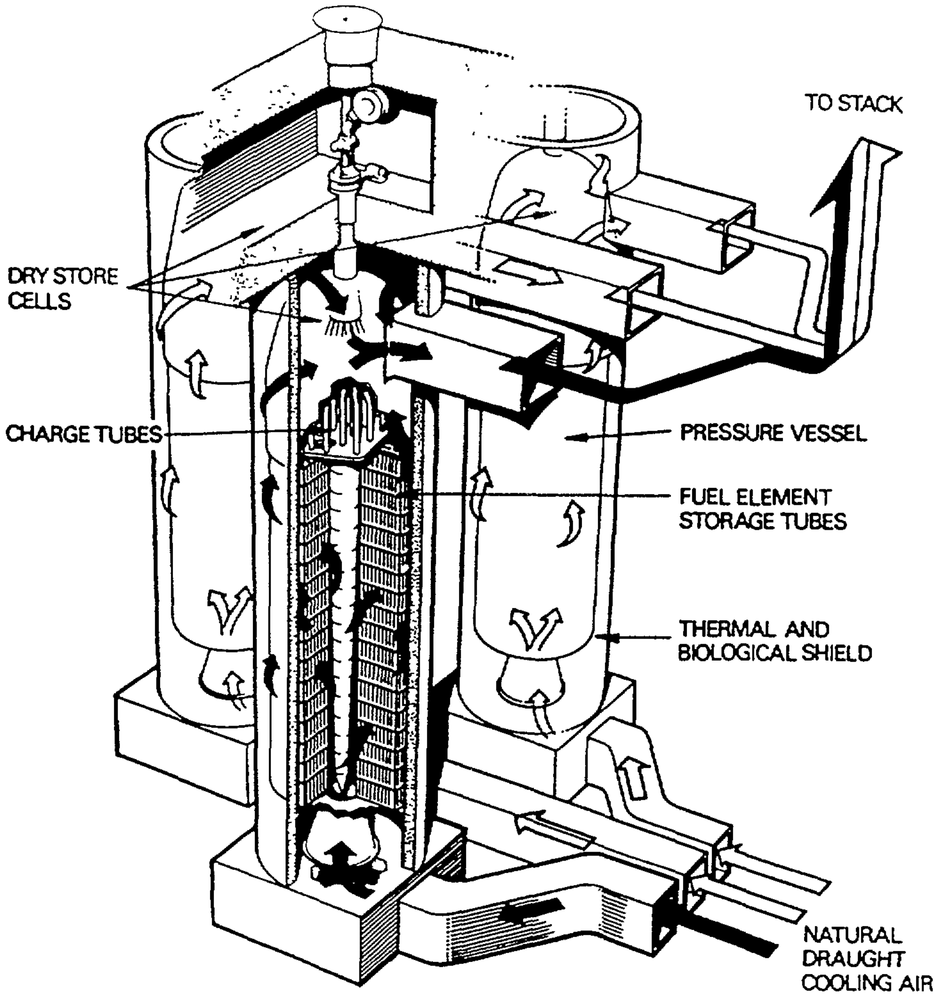


FIG. 2. The Wylfa CO_2 dry storage system.

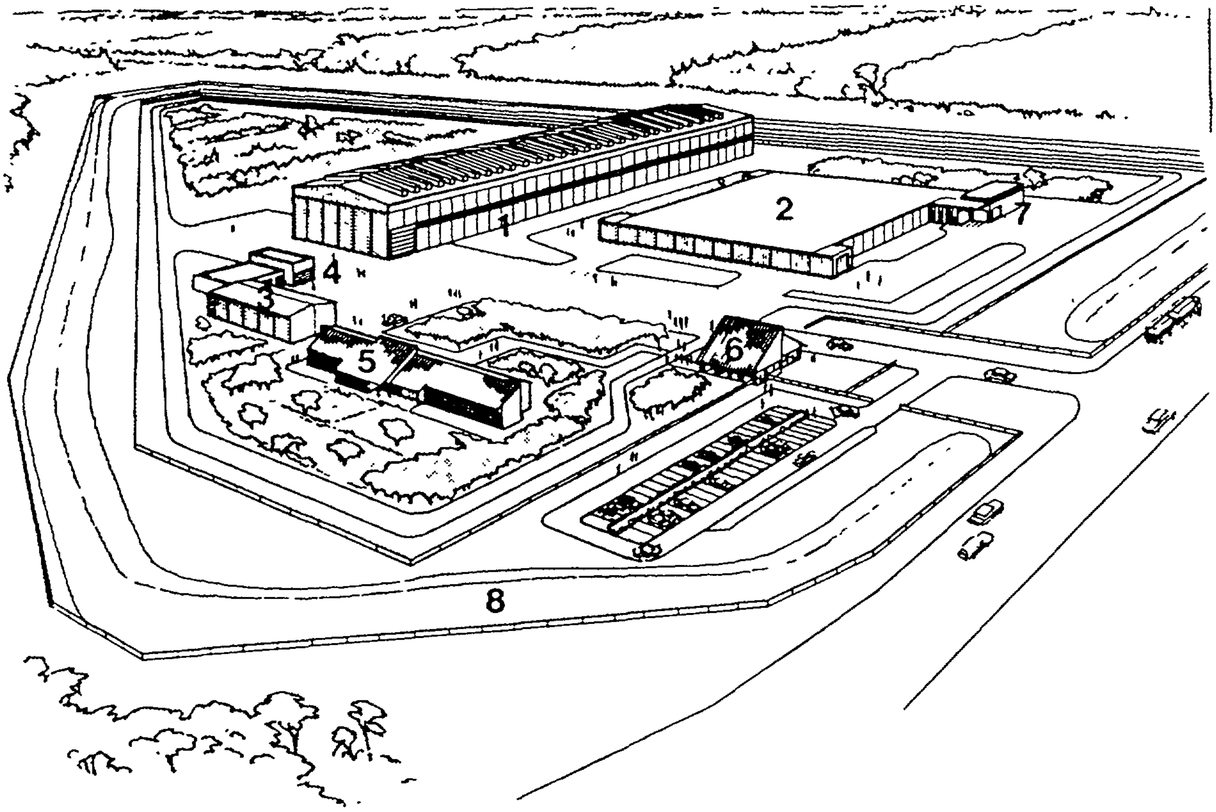
Nuclear Electric and Scottish Nuclear are considering new centralized vault systems, which would allow storage of spent fuel from the British advanced gas reactors for periods up to 100 years.

Other vault designs

Siemens (Federal Republic of Germany) has designed a vault to store quantities of uranium greater than 300 MTU [5]. Foster-Wheeler Energy Applications Inc. has designed a modular concrete vault for the USA, whose topical report [30] has been approved by the NRC. This facility was under construction in 1991. The fuel in the Siemens vault is blanketed with helium, while fuel in the US vault is blanketed with nitrogen; the Republic of Korea is also conducting a vault feasibility study.

3.3.1.3. Spent fuel storage and storage/transportation casks

A cask is a massive container for transportation and/or storage of radioactive material. It contains one or more storage cavities with a controlled environment. Each storage cavity can be designed to contain several spent fuel assemblies. Shielding and radioactive particulate confinement is provided primarily by the cask structural material such as steel, cast iron or concrete. Heat removal is by conduction through the structural material to the atmosphere.



1. Storage building for containers
2. Storage building for low radioactive waste
3. Workshop building
4. Plant building
5. Administration and recreation building
6 Porter's lodge
7. Reception building for low radioactive waste
8. Safety fencing

FIG. 3. Perspective view of the Gorleben storage site.

Cask tests and demonstrations in the Federal Republic of Germany

the Federal Republic of Germany has conducted tests and demonstrations of several Castor casks (Castor IA, IB, and IC; Castor AVR; Castor KRB-MOX; and Castor WWER 1000) and a TN-1300 cask. The objective was to verify the performance of dual purpose storage and transportation casks, optimize cask handling procedures, and expand the database on the long term integrity of stored spent fuel. The results confirmed both the performance of the casks and the maturity of the technology. the Federal Republic of Germany is now concentrating on optimizing the casks.

As a result of the successful tests and demonstrations, the away-from-reactor storage facility at Gorleben received a licence to store fuel in the Castor I casks, which will first be used to transport the fuel to Gorleben.

the Federal Republic of Germany is licensed to store spent fuel at central facilities for approximately 40 years. Of these central facilities, the store at Gorleben has been licensed. Approximately 1500 MTU of spent fuel will be stored in 420 casks. These will be positioned in a building about 182 m long, 38 m wide and 20 m high. The storage building protects the installation from the effects of weather. Openings in the walls and roof allow a flow of air by natural convection, which dissipates the fuel decay heat to the environment (see Figure 3).

Storage casks in the USA

Since 1984, four types of metal storage casks have been tested in the USA. One of the four tested was the REA-2023, manufactured by Ridihalgh, Eggers and Associates for the storage of BWR fuel. The results of the tests conducted at Barnwell indicated that the casks could be handled at many reactor sites and could, with minor design refinements for shielding, be used for the safe storage of spent fuel at a licensed storage facility.

In 1984, the Virginia Power Company entered into a cooperative agreement with the DOE and the Electric Power Research Institute to test, at the Idaho National Engineering Laboratory, the performance of three different metal casks for storing PWR fuel: the Castor V/21 cask marketed by General Nuclear Systems, Inc., the TN-24P cask manufactured by Transnuclear, Inc., and the MC-10 cask manufactured by the Westinghouse Electric Corporation. In addition to the general objectives stated above, these tests were conducted to support the utility's efforts to obtain an NRC licence for a dry storage facility at the site of the two Surry reactors.

The tests, performed in 1985 and 1986, indicated that all three casks could be safely handled and loaded and, with very minor refinements, used at reactor sites for the safe storage of spent fuel. The shielding performance met design expectations, and the heat-transfer performance of the casks was exceptionally good with the measured temperatures lower than predicted in the topical reports.

In July 1986, after approximately 4 years, Virginia Power obtained from the NRC a licence for dry storage (i.e., an independent spent fuel storage installation) at the Surry site.Three Castor V/21 casks were subsequently loaded with fuel in the storage pool of the Surry reactor site, and their use was successfully demonstrated in 1986 and 1987. Virginia Power also plans to use casks from other vendors in these licensed demonstrations. An NAC cask is now scheduled for loading in 1991 and a Westinghouse cask in 1992.

As of the end of 1990, Virginia Power had twelve loaded Castor V/21 casks on the storage pad of the Surry dry storage installation. The utility plans to place two or three more loaded casks a year on the storage pad.

Storage/transportation casks in the USSR

The USSR has designed and constructed a storage/transportation cask (TK-13) for the investigation of spent WWER-1000 fuel behaviour in a gaseous (inert gases, air) medium. The flask, designed to handle >3-year cooled fuel, is primarily intended for transportation but will also be used for short term (1 year) storage. One successful demonstration lasting one year has already been provided at the South Ukrainian NPP.

Concrete casks

In Canada, Ontario Hydro has undertaken a programme to assess the feasibility of using concrete casks for storing, transporting, and possibly disposing of spent fuel from the CANDU heavy water reactors. A concrete dry storage container, formally known as a CIC, has been designed. The container is approximately 2.6 m in diameter and 3.6 m in height and, when loaded, weighs 68 t. The walls of the cask will be made from a special high-density concrete mix with rebar reinforcement. The reinforced-concrete body is lined on both the inside and the outside with epoxy coated carbon steel. Trunnions are attached to the body for lifting. A lid, also made from the carbon-steel lined reinforced concrete, is bolted to the body of the cask and sealed using conventional elastomer and metal seals.

Ontario Hydro believes that the dry storage container could become an important alternative method for CANDU spent fuel storage, transport, and possibly disposal. A two-phase demonstration programme has been started. Two such containers were built for storage demonstrations in 1988/1989. Ontario Hydro has upgraded its current demonstration container for transportation testing. The upgraded design will have a rectangular cross-section (rather than the current circular configuration), and a welded lid closure system. Drop and fire testing on 1/4 and 1/2 scale models are in progress and started in 1989. The tests are required for licensing and will be conducted to meet AECB/IAEA requirements.

In the USA, a Concrete Cask Cooperative Agreement was signed between DOE, Wisconsin Electric Power, EPRI and Pacific Sierra Nuclear Associates (PSNA) in 1989 to demonstrate spent fuel storage in a concrete cask at Idaho National Engineering Laboratory (INEL). Loading of this cask took place in October of 1990. Also, all technical and documentation work has been submitted to the NRC in support of PSNA's application for an approved Topical Report. The NRC is expected to issue this approval in 1991.

Concrete casks are planned to be used in NRC licensed spent fuel storage installations, starting with Consumers Power Co. of Michigan in 1992. A cutaway schematic of the Consumers Power cask is shown in Figure 4.

3.3.1.4. Horizontal concrete modules

In horizontal concrete storage modules, the spent fuel is kept inside a sealed stainless steel canister that is filled with helium or nitrogen, and the canister is protected and shielded

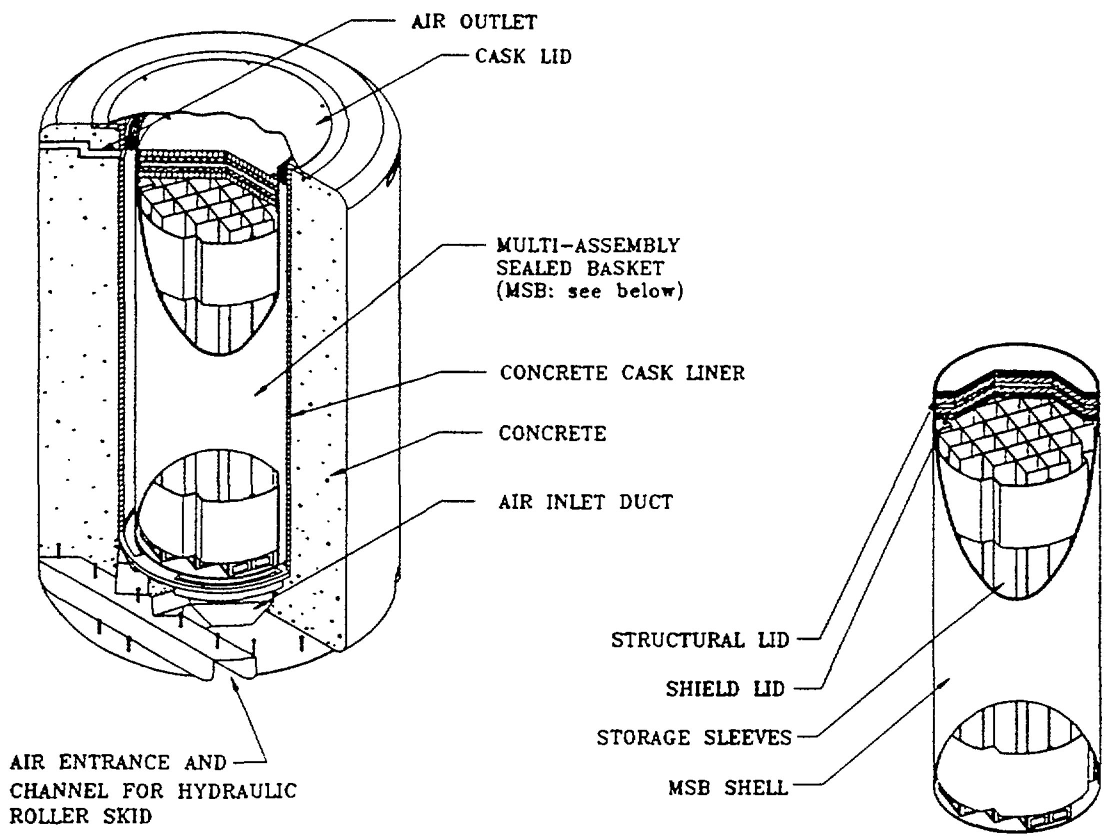


FIG. 4. Cutaway of the commercial 24 assembly concrete ventilated storage cask.

by a concrete module. The heat generated by the spent fuel is removed by radiation, conduction, and natural convection through air channels in the concrete module.

Two independent spent fuel storage installations which use horizontal concrete modules have been licensed and constructed, and at least two additional installations are being designed. All of the systems use the Nutech horizontal-modular-storage system (NUHOMS) manufactured by Nutech and are designated either NUHOMS-07 or NUHOMS-24, depending on the number of assemblies stored in a module.

The NUHOMS-07 system was selected by Carolina Power and Light Company (CP&L) for its H.B. Robinson nuclear power plant. The installation at H.B. Robinson is licensed for eight modules each holding seven elements, all of which have been loaded with spent fuel. Additionally, Duke Power at its Oconee station, has constructed a NUHOMS system using a 24 element sealed canister. The initial loadings at this facility were made in 1990 with six canisters in place by January 1991. Baltimore Gas and Electric also is constructing a NUHOMS storage facility. A schematic of the NUHOMS-24P system is shown in Figure 5.

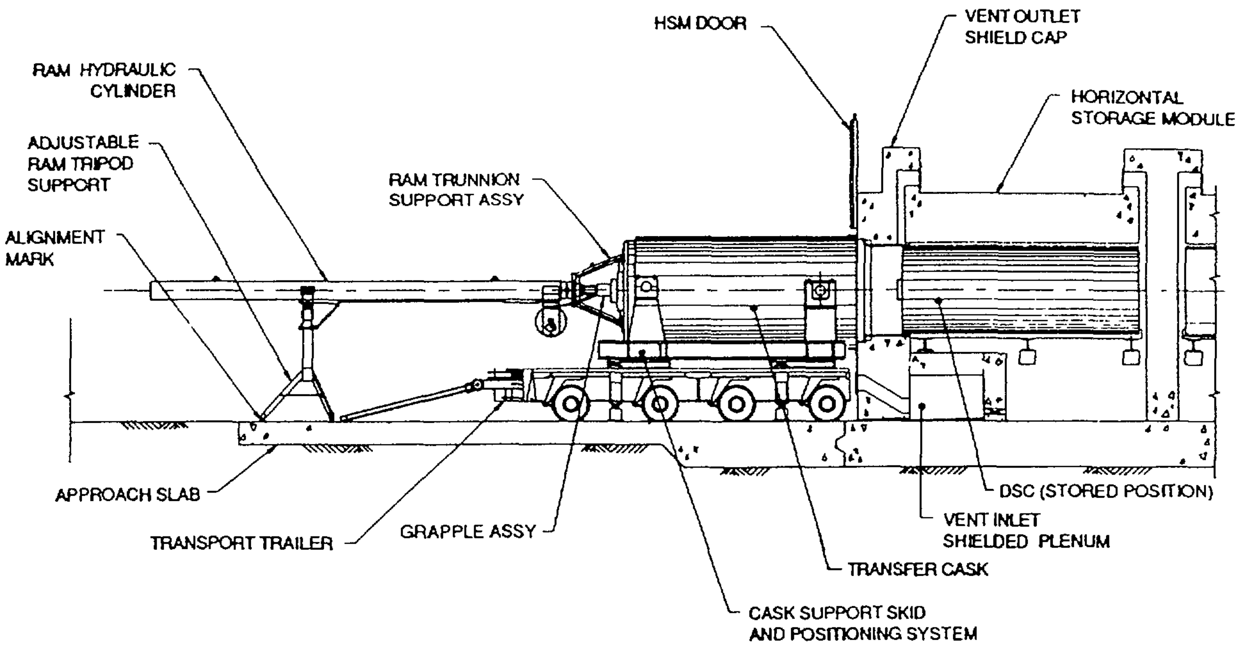


FIG. 5. NUHOMS® -24P system configuration.

3.3.1.5. Drywells

A drywell is a stationary, below ground, lined, individual cavity for the storage of one or more fuel assemblies. The actual number of fuel assemblies is determined by the fuel and storage media. However, it is normally necessary to cool the assemblies for two or three years before storage in a drywell. Shielding is provided by the surrounding earth and closure shield plug. Primary heat removal is to the earth.

In the USA, demonstration programmes of this concept have now been discontinued. Previously, the demonstration programmes (EMAD), were conducted with two PWR and two BWR drywells at the Nevada test site.

In Japan, aluminum clad natural U-metal fuel from the JRR-3 research reactor is being stored in drywells [26, 27]. The JRR-3 research reactor contained 600 assemblies with a total weight of 15 MTU. The assemblies were 5.42 m long including the shielding plug and the pin length was 3.27 m. The weight per assembly was about 80 kg and included 25.8 kg of heavy metal. The assemblies were in the core for 5-9 years and burnups ranged from 200 to 800 MW·d/t U. The assemblies were in wet storage about 9 years and the change to dry storage was initiated in 1982 and completed in 1984. They continue to be stored to the present. The fuel assemblies were dried in a hot cell for one week. All of the assemblies were visually inspected, with some undergoing X ray inspections. The dry assemblies were placed in a canister and the seal welded with a helium atmosphere in the canister. The canisters were transported to the dry storage facility. The heat rating in each canister upon entry into drywell was 16.4 W. There were 36 assemblies in each canister with one canister per drywell (see Fig. 6). The ambient condition in the well is dry air at less than ambient pressure monitored for temperature, pressure and radioactive material release. To date, no problems have occurred.

3.3.2. Fuel transfers

Dry storage handling of spent fuel assemblies has been a common practice for decades in hot cell facilities. In addition, some UK gas cooled reactors have used special shielded remotely operated equipment for the dry handling of spent fuel during the transfer from the reactor core to the storage facility. The above provides the basis for the handling of spent fuel for the various dry storage concepts. These dry storage concepts need different approaches for relevant dry fuel handling.

Before the fuel is placed in dry storage, at some point in time, in most cases, the fuel has to be transferred from a wet to a dry storage environment. Adequate experience exists for Zircaloy clad fuel, whether or not the cladding has suffered degradation. The use of special handling procedures is dictated by the requirements of the storage facility, the safety analysis and minimizing the dose to the operators.

3.4. SUMMARY AND CONCLUSIONS

<u>Zircaloy clad fuel</u>

- Dry storage in an inert atmosphere is licensed in the the Federal Republic of Germany and the USA, and in air in Canada.

- Maximum fuel cladding temperature is licensed in the USA (380°C) and the Federal Republic of Germany (410°C).

- Various modes of design approaches are used. Higher storage temperatures require an inert gas atmosphere (e.g. Federal Republic of Germany, USA) while low storage temperatures allow storage in air (e.g. Canada).

- In all operational applications, no significant fuel degradation in storage has been observed. A few indications of krypton release have been observed indicating possible individual rod failures. These suspect fuels, however, have not been examined to confirm the failures.

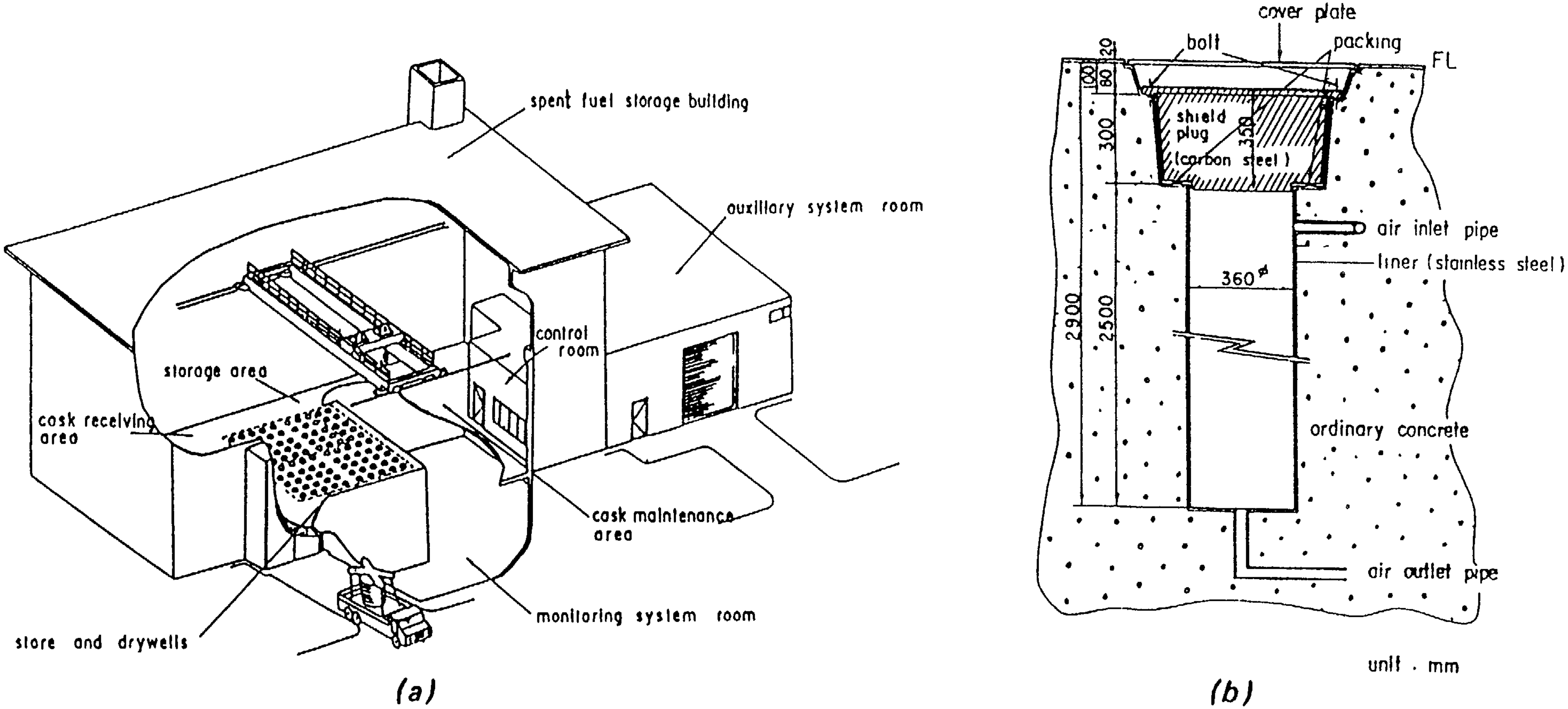


FIG. 6. Japanese dry storage facility for JRR-3 research reactor fuel. (a) bird's eye view of dry storage facility; (b) storage hole of DSF.

Zr-1%Nb clad fuel

- Inert gas storage is feasible.

- Maximum fuel cladding temperature of 350 °C is licensed for inert gas.

- Dual purpose casks constitute the preferred design approach in USSR.

- One successful demonstration using intact fuel with the cladding temperature of 340 °C has already been performed.

Stainless steel clad (water reactor) fuel

- No systematic storage experience is available.

Stainless steel/oxide (gas cooled reactor) fuel

- Dry storage is being investigated for spent AGR fuel.

Magnox fuel

- Dry storage of spent Magnox fuel has been licensed in the UK (CO_2 or air depending on fuel temperature) and in Italy (nitrogen).

Aluminium clad fuel

- Limited experience exists of inert gas storage for Al-clad fuel at ambient temperature.

Dry storage is becoming widely used as a supplement to wet storage for zirconium alloy clad oxide fuels. Storage periods as long as under wet conditions appear to be feasible. Dry storage will also continue to be used for Al clad and Magnox type fuel.

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LIST OF ABBREVIATIONS

AECB	Atomic Energy Control Board (Canada)
AECL	Atomic Energy of Canada Limited
AFR	Away-from-Reactor Storage
AR	At-Reactor Storage
AGR	Advanced Gas Cooled Reactor
BEFAST	Behaviour of Spent Fuel Assemblies in Extended Storage
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium-Uranium Reactor
CEX	Controlled Environment Experiment (Canada)
CIC	Concrete Integrated Canister (Concrete Dry Storage Container)
CRL	Chalk River Laboratories (Canada)
CRP	Co-ordinated Research Programme
DOE	Department of Energy (USA)
DPNGS	Douglas Point Nuclear Generating Station
DSF	Dry Storage Facility
EMAD	The Engine Maintenance, Assembly and Disassembly Facility (DOE, Nevada Test Site) (USA)
EPRI	Electric Power Research Institute (USA)
ERB	Easily Retrievable Basket (Canada)
FA	Fuel assembly
FRG	Federal Republic of Germany
GDR	German Democratic Republic
HLW	High level waste
INEL	Idaho National Engineering Laboratory (USA)
JAERI	Japan Atomic Energy Research Institute
KAERI	Korea Atomic Energy Research Institute
KEPCO	Korea Electric Power Corporation
LWR	Light Water Reactor
MAGNOX	Magnesium no Oxidation (Magnesium Alloy Cladding, UK)
MTR	Materials Testing Reactor
MTU	Metric Tons of Uranium
NAC	Nuclear Assurance Corporation (USA)
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (USA)
NUHOMS	Nutech Horizontal Modular Storage System (USA)
PSNA	Pacific Sierra Nuclear Associates (USA)
PNC	Power Reactor and Nuclear Fuel Development Corporation (Japan)
PWR	Pressurized Water Reactor
RBMK	USSR Type of Channel Reactor, Water Cooled, Graphite-Moderated (Reaktor Bolshoi Moschnosti Kipyaschii)
RCM	Research Co-ordination Meeting (IAEA)
RR	Research reactor
SF(A)	Spent fuel (assembly)
SS	Stainless steel
TRS	Technical Reports Series (IAEA)
UK	United Kingdom
USA	United States of America
USSR	Union of Soviet Socialist Republics

WWER	USSR Type of PWR (Wodo-Wodyanoi Energeticheckii Reactor)
WL	Whiteshell Laboratories (Manitoba, Canada)
Zry	Zircaloy

Annex I

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Annex II

ASPECTS OF SPENT FUEL STORAGE IN COUNTRIES PARTICIPATING IN BEFAST-II

SPENT FUEL STORAGE IN ARGENTINA

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1.- SPENT FUEL STORAGE

In Argentina all the spent fuels are now stored in water pools. The fuels of nuclear power plants are stored at the reactor site. The fuels of research reactors are stored at the reactor site for several months, and are deposited in storage pools at the Ezeiza Atomic Center. Additional storage will be required at the two nuclear power plants and it will be provided by water pools or by dry storage.

There are two basic types of pool currently used, the primary pool which receives fuel directly from the reactor, and the secondary pools which provide additional storage capacity as required. The secondary pool is interconnected with the primary pool and the transfer between bays is done under water.

The pools are concrete structures which are lined with stainless steel or epoxy.

A developing and cooperation program was initiated to provide dry storage capacity at the Embalse Nuclear Power Plant.

In support of the spent fuel program, research was initiated by CNEA in the early 70's to investigate the long term performance of spent fuel in wet storage controlling the water parameters and using periodically non-destructive tests. Results of this program suggest that fuel performance will be good for at least fifty years in wet storage.

2.- DISPOSAL

There is currently international consensus on the view that the disposal of high-level radioactive wastes, conditioned in solid form an located in suitable deep geological formations, is a solution which, for present and future generations, will involve risks that are no greater than those normally accepted for daily life.

The radioactive wastes will be incorporated in a borosilicate-type vitreous matrix whithin a stainless steel recipient. This recipient will then be encased in a lead wall some 10 cm thick with an external metal protective sheet, in order to ensure that the wastes remain isolated for a period of approximately 1000 years. The design of the containers will also comply with the requirements of the IAEA's Regulations for the Safe Transport of Radioactive Materials.

The four nuclear power plants that are planned up to the end of this century represent an installed electrical capacity of 2.3 GW.

The wastes arising from reprocessing of the fuel consumed in the course of 30 years will require approximately 2000 containers of approximately 0.60 m diameter and 1.60 m in height. The storage facility for these 2000 containers has a surface area of less than 1 Km2.

Preliminary evaluations showed that the disposal of radioactive wastes in crystalline rocks at a depth of 500 m or more would sufficiently reduce the overall radiological impact. For this reason, and bearing in mind the geological characteristics of the country it was decided to dispose of the wastes in stable granitic formations located at a depth of 500 m, away from seismic zones and with low hydraulic conductivity.

The first stage in the siting studies was to examine all of the known outcrops of granitic rocks in Argentina. This led to the identification of 198 possible granitic outcrops.

The second stage was to draw up a shorlist from among the formations identified; this resulted in the choice of seven granitic outcrops located in the provinces of Chubut and Río Negro, in the south of the country.

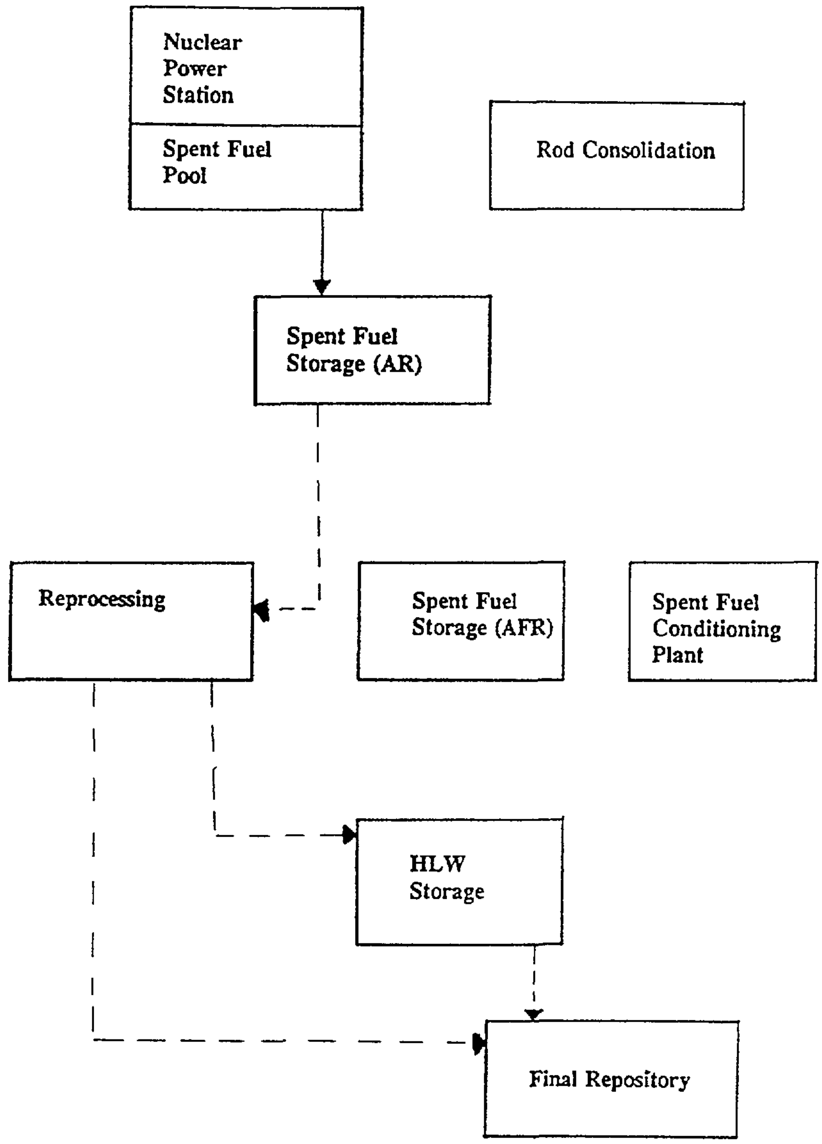
The third stage, involving the topographical survey of the granitic outcrop shortlisted led to identification of the la Esperanza and Chasicó massifs, in the province of Río Negro, and those of Sierra de Calcatapul and Sierra del Medio, in the province of Chubut, as the most appropriate in which to continue with detailed studies.

The studies commenced with the granitic outcrop of Sierra del Medio, in accordance with the following plan:

(a) Photographic interpretation;
(b) Statistical alignment analysis;
(c) Geological and geophysical inspection of the rock massif;
(d) Intermediate perforations to a depth of 200;
(e) Regional geomorphological and hydrogeological analysis; and
(f) Small-diameter deep perforations to a depth of 800 m

In the near future we plan to commence detailed geological and hydrogeological studies on the site selected in order to determine the hydraulic conductivity of the massif and to conduct various chemical and physico-chemical measurements in the deepest perforations. We are also performing geological studies of the volcanic masses located in the Gastre depression, which surrounds the Sierra del Medio. The aim of these studies is to determine the past and the future influence of volcanic eruptions on the stability of the Sierra. The studies carried out to date have enabled us to establish that the outflows have no resulted in deterioration of the Sierra and that the ages of the outflows, determined using the Ar-K technique, are greater than $0.8 + 0.1 \cdot 10^6$ years.

The results obtained so far confirm the suitability of this site. However, should subsequent results disqualify it, we will undertake a study on one of the other shortlisted granitic bodies.



Spent fuel management in Argentina

DRY STORAGE OF SPENT FUEL IN CANADA

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1 INTRODUCTION

The purpose of this report is to provide an update on the dry storage of used fuel in Canada described at the last BEFAST-II meeting in Espoo, Finland, in 1989 August [1]

2 CURRENT STATUS

2 1 AECL Designed Concrete Canisters

In Canada concrete canisters (CCs) are currently being used to store used fuel at four sites (1) AECL's Whiteshell Laboratories, (2) the Gentilly nuclear reactor site, (3) the Douglas Point Nuclear Generating Station, and (4) AECL's Chalk River Laboratories Details of the design of the canisters were presented at previous meetings [1,2]

Concrete canisters have been adopted as the storage method for future used fuel discharged from New Brunswick Electric Power Commission s 600-MW(e) Point Lepreau Nuclear Generating Station As of 1990 February 1, this station completed seven years of operation, at a lifetime capacity factor that was amongst the highest achieved in the world [3] The Point Lepreau CCs have a significantly greater storage capacity (10 3 Mg U/CC) than the Douglas Point or Chalk River CCs (6 5 Mg U/CC) and have been licensed to store 6-year-cooled CANDU fuel in air, although only older fuel will be loaded initially Of the 20 CCs constructed at Point Lepreau in 1990, 15 will be loaded in 1991 with used fuel containing 150 Mg U Loading is expected to start in June

Since the last BEFAST-II meeting used fuel containing 65 Mg U has been loaded into CCs at Chalk River Laboratories This fuel was discharged from NPD Canada s first demonstration CANDU power reactor, which was shut down in 1987 after 25 years of operation Also, another 1 Mg U of fuel from the WR-1 organic-cooled research reactor has been loaded into a CC at Whiteshell Laboratories WR-1 has also been shut down and all of its used fuel will be stored dry in canisters

By the end of 1991 fuel containing 606 Mg U will be stored in concrete canisters in Canada The quantities of fuel stored at each site are summarized in Table 1

TABLE 1

QUANTITIES OF FUEL STORED IN AECL DESIGNED CONCRETE CANISTER IN CANADA

Site	Location	No Canisters	Mg U
Whiteshell	Manitoba	19	24
Gentilly	Quebec	11	67
Douglas Point	Ontario	47	300
Chalk River	Ontario	12	65
Point Lepreau	New Brunswick	20*	150
		TOTAL	606

*Only 15 of the 20 CCs are scheduled for loading with 150 Mg U in 1991
Note Some of the CCs at all other sites, except Gentilly, are spares that contain no fuel

In 1990 February, AECL signed an agreement with the Korean Electric Power Corporation (KEPCO) to provide AECL s used-fuel dry storage technology for the storage in concrete canisters of CANDU fuel bundles from the Wolsung-1 reactor

2 1 Ontario Hydro's Dry Storage Container

In autumn 1990, Ontario Hydro loaded a second cylindrical demonstration concrete dry storage container, with 384 bundles (7 7 Mg U) of 6-year-cooled CANDU fuel from Pickering Nuclear Generating Station-A Details of Ontario Hydro's Dry Storage Container (previously referred to as a Concrete Integrated Container or CIC) have been provided at previous BEFAST II meetings [1 2] The demonstration containers were loaded in the used fuel interim storage pool transferred by the pool crane to a trailer then moved to an outside storage area An upgraded (rectangular) Dry Storage Container is currently in the design stage

Preliminary drop and fire tests have been carried out on a 1/4 scale rectangular Dry Storage Container for transportation licensing purposes The 9 and 1-m drop tests successfully demonstrated that the container design could meet the impact test requirements of a Type B transportation package [4] The fire test results indicated that neither the shielding nor containment integrity was compromised by the conditions required for Type B packages [4] The drop tests are to be repeated on a 1/2 scale model, whereas an analytical model will be used to model a fire test on a full scale container

3 FUTURE PLANS

Whiteshell Laboratories is in the process of obtaining criticality approval to load ~1 Mg U of enriched fast neutron fuel into CCs at Whiteshell

Since New Brunswick Electric Power Commission has decided to use concrete canisters to store future used-fuel arisings from its Point Lepreau Station, about 30 more years of used-fuel production will have to be stored, assuming a 40-year plant life This represents 147 000 bundles, or about 2 790 Mg U in 273 canisters At present, it is planned to construct and load 10 CCs per year (100 Mg U/a)

Hydro Quebec is evaluating the dry storage of used fuel in CCs for its 600-MW(e) Gentilly-2 CANDU Nuclear Generating Station Tentative plans include loading used fuel at a rate of 100 Mg U/a, starting in 1994

Ontario Hydro plans to use its concrete Dry Storage Container to store used CANDU fuel from Pickering Nuclear Generating Stations A and B Loading is expected to begin in 1993 at a rate of 500 Mg U/a

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WET STORAGE OF SPENT FUEL IN CANADA

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1.0 INTRODUCTION

The purpose of this report is to provide an update on the wet storage of spent fuel in Canada.

2.0 CURRENT STATUS

2.1 Background

All CANDU reactors discharge spent fuel directly to waterpools (see Table 1 for details of Canadian reactors, spent fuel arisings and stored inventory) for initial cooling and interim storage. After interim storage in the primary

TABLE 1

CANADIAN NUCLEAR GENERATING STATIONS [a]

Owner	Station	Capacity Mwe Net	Typical Annual Fuel Arisings (in thousand bundles)	Inventory on December 31/90 (in thousand bundles)
Ontario Hydro	Pickering A	515x4	6.8	211.1
	Pickering B	516x4	12.2	80.7
	Bruce A	740x4	18.5	246.2
	Bruce B	784x4	21.6	110.6
	Darlington[b]	881x4	23.6	0
Hydro Quebec	Gentilly 2	638	4.3	30.3
New Brunswick Electric Power	Point Lepreau	633	4.3	42.5
Atomic Energy of Canada[f]	NPD[c]	22		
	Douglas Pt[d]	206		
	Gentilly 1[e]	250		

(a) All units are CANDU, with a pressurized heavy water coolant (or CANDU-PHW)

(b) Two units are operational, two others are being commissioned or under construction.

(c) Unit shut down on August 1, 1987.

(d) Unit shut down on May 4, 1984.

(e) Unit shut down in 1976.

(f) Research reactors such as NRX, NRU, WR-1 excluded, because of small influence on fuel arisings.

TABLE 2

CANADIAN SPENT FUEL WATERPOOLS

Station	Type	Dimensions * (m)	Capacity 000's of Bundles	In-Service Date	Liner Material
Pickering A	Primary	16.3W x 29.3L x 8.1D	93	1972	All epoxy
	Auxiliary	17W x 34L x 8.1D	214	1978	All epoxy
Pickering B	Primary	16.3W x 29.3L x 8.1D	158	1983	Receiving bay, all S/S Storage bay, all epoxy
Bruce A	Primary	10W x 41L x 6D	21	1977	S/S-floor, epoxy-walls
	Auxiliary	18W x 46L x 9D	352	1979	S/S-floor, epoxy-walls
Bruce B	Primary	10W x 46L x 6D	36	1983	All S/S
		18W x 46L x 9D	330	1987	All S/S
Darlington**	Primary	(a) 9.65W x 20.6L x 5D (b) 17W x 32L x 9.2D (c) 17W x 4L x 9.2D	212	1989	All S/S
Gentilly - 2	Primary	11.5W x 22.1L x 7.6D	47.4	1983	All epoxy
Point LePreau	Primary	11.8W x 18.2L x 7.6D	49.4	1983	All epoxy

* W - width, L - length, D - depth.

** Darlington will have two identical primary pools, the second (east) one will be in-service in 1991, with the fill date about 2007. Each primary pool consists of a fuel receiving bay (a), storage bay (b) and a fuel cask handling bay (c). The second primary pool will also have a capacity of 212,000 bundles.

waterpool, for a period of from several months to about seven years, the spent fuel is transferred to an auxiliary wet or dry storage facility.

Table 2 provides details of the spent fuel pools in Canada. Commercial electricity generation by CANDU reactors began in 1962, and Canada has a significant quantity of spent fuel in interim wet storage. A companion paper(1) at this meeting updates the status of dry storage in Canada.

2.2 Operating Experience

Based on the excellent AECL experience(2) with spent fuel waterpool storage since 1947, all Canadian utilities have used waterpools for interim storage. This extensive operational experience, plus the development of high density storage systems, the successful on-site transfer of spent fuel from the primary waterpools to auxiliary wet and dry storage facilities, and the use of remote handling mechanisms, have all helped in meeting the logistics challenge of handling the large quantities of spent fuel bundles in Canada.

Operation has been relatively trouble-free. The only events of significance(3) have been (1) chemical cleaning in the early 1980s to remove fouling from heat exchanger tubes on the lake water side, at both Pickering A and Bruce A stations, (ii) some pool water leakage which is collected for analysis and appropriate treatment, and (iii) some radiation-induced surface degradation of the Pickering A primary pool epoxy liner. Using underwater-curing epoxy repair material, a leak at the interface between the Bruce A auxiliary waterpool stainless steel floor liner and the epoxy wall liner was successfully repaired in 1989. This type of repair material will be used to rehabilitate the Pickering NGS A primary waterpool epoxy liner where radiation - induced surface degradation has occurred.

2.3 Back-Up Research

If epoxy liner radiation-induced deterioration continues, there is a possibility that pool water may eventually contact the structural concrete. Thus, a program to investigate the long-term effect of water on concrete integrity is in progress. Static tests over more than a year showed that the rate of leaching of dissolved solids from the concrete by demineralized water decreased with time. Initial results after three months from dynamic tests, with demineralized water and slightly acidic water at 18°C and 40°C, indicate that the higher the water flow, the greater the calcium loss from the concrete. After a year's exposure, the porosity of the concrete had increased to a depth of 9 mm from the surface exposed to flowing water[(4)]. The monitoring of alkali metal cations, which are the most mobile of the concrete ions is a good indicator of the extent of concrete leaching. This work is on-going.

2.4 Long-Term Fuel Integrity In Wet Storage

A key element in the design of the waterpool is continued spent fuel integrity. Thus, all handling systems for spent fuel and spent fuel containers are designed to prevent any mechanical damage to the fuel bundles. By maintaining specified bay temperature limits and bay water chemical specifications, any possible adverse chemical effects on fuel integrity are minimized.

Ontario Hydro and AECL have a long-term program, initiated in 1977, to examine for possible deterioration spent fuel stored in water. The oldest bundles selected for the program have been in wet storage since 1962. Results from the first examination of this program in 1978/79[(5)], showed no apparent deterioration of either the uranium dioxide fuel matrix (for defected fuel) or zircalloy cladding, after wet storage for more than seventeen years. Based on these results, it was concluded[(5)] that spent fuel should maintain its integrity during at least fifty years of underwater storage. Results to date[(6)] from a re-examination after a further ten years storage indicate no significant change in (i) the fuel element integrity (ii) the condition of the cladding and (iii) for defected fuel, the UO_2 matrix, after up to 21 years wet storage.

These results indicate that the integrity of the CANDU spent fuel will be maintained in wet storage, and the fuel will be retrievable well into the next century for future downstream fuel management processes.

3.0 CONCLUSIONS

Canada has much experience with the design, construction, operation and maintenance of spent fuel wet storage facilities. Waterpools at the reactor sites have been designed with capacities ranging from about 700 Mg to 7000 Mg of spent fuel. Spent fuel is being successfully transferred from the primary storage pools to auxiliary wet and dry storage facilities.

The design of waterpools has effectively met the objectives to manage the fuel discharged from CANDU nuclear reactors in a safe, reliable and economic manner, and maintain spent fuel integrity, and has enabled relatively trouble-free operation over a period of forty or more years.

Tests on spent fuel after wet storage for periods of more than twenty one years indicate no deterioration, whether the fuel is defected (i.e., with a through-wall defect in the cladding) or not. All evidence to date suggests there will be no significant change in spent fuel bundle integrity over at least a fifty year wet storage period, whether or not there are any fuel cladding through-wall defects.

ACKNOWLEDGEMENTS

Thanks are due to A.L. Manzer, D. Barber and D. Koivisto of AECL CANDU, who provided background information on Pt LePreau and Gentilly-2 waterpools.

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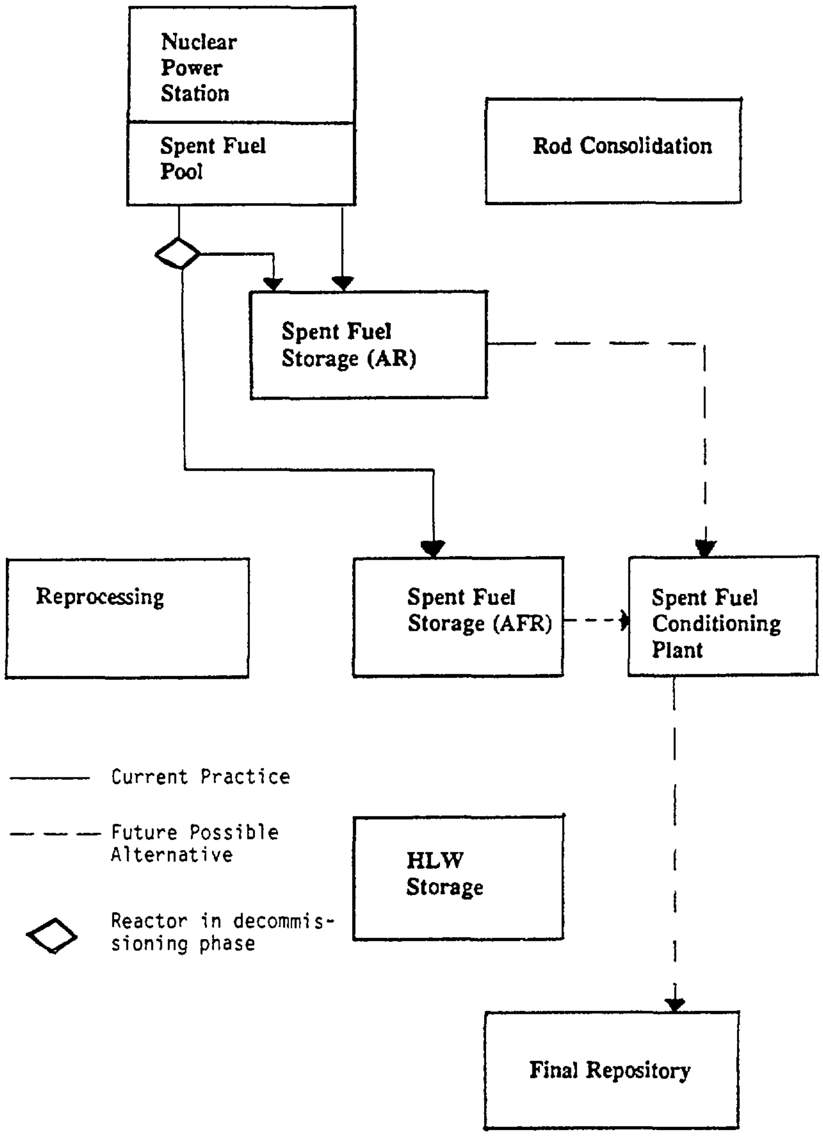
Spent PHWR Fuel



Spent fuel management in Canada

SPENT FUEL MANAGEMENT IN FINLAND
(Status March 1991)

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GENERAL

In Finland, about 30 % of electricity is produced by nuclear power. There are four reactors in operation, two PWRs at Loviisa (2 x 445 MWe, start of operation in 1977 and 1980) operated by Imatran Voima Oy (IVO) and two BWRs at Olkiluoto (2 x 710 MWe, start of operation in 1978 and 1980) operated by Teollisuuden Voima Oy (TVO).

The Nuclear Energy Act and Decree, passed in 1988, form the central regulatory basis for the management of spent fuel and other nuclear wastes.

Responsibilities, licencing procedures and financing principles are defined in them. The objectives and schedules of nuclear waste management have been defined in more detail in the Government's policy decision of 1983.

The utilities are responsible for the safe management of their spent fuel: research and development work, implementation and financing of spent fuel management. A number of research institutes, universities and consultants participate in the programme as contractors, such as the Technical Research Centre of Finland, the Geological Survey of Finland and the University of Helsinki.

The progress of the waste management programme is supervised by the Ministry of Trade and Industry. The Finnish Centre for Radiation and Nuclear Safety is responsible for the supervision of the safety of plans and activities. The major facilities are licenced by the Government. Part of the research work is funded by the Ministry in order to maintain independent expertise for the supervision of the activities of the waste producers.

SPENT FUEL ARISINGS

The Loviisa power plant produce annually 28 tU (tons Uranium) of spent fuel. IVO has an agreement with the Soviet fuel supplier on the return of the Loviisa spent fuel to the Soviet Union. Fuel assemblies are stored for five years at Loviisa before transportation. In March 1991 the amount of spent fuel stored at Loviisa was 124 tU.

The annual production at Olkiluoto is 45 tU. The amount stored at Olkiluoto was 450 tU at the end of 1990.

INTERIM STORAGE

IVO stores spent fuel assemblies in the water pools of the Loviisa power plant until they are transported to the Soviet Union.

Also TVO stores assemblies in the storage pools of power plant units. In addition, TVO has constructed an interim storage facility (KPA store) at the Olkiluoto power plant site. The three water pools of the store have a storage capacity of 1200 tU. The storage facility was commissioned in 1987. At the end of 1990, 250 tU had been transferred to the KPA store.

FINAL DISPOSAL

The alternatives for Olkiluoto spent fuel management after the interim storage phase are

- direct disposal of spent fuel in Finland
- foreign reprocessing and return of wastes to Finland
- foreign reprocessing including waste disposal, or foreign direct disposal services.

So far, no agreements have been signed on the foreign services. Preparations are made for final disposal of spent fuel in the Finnish bedrock.

An updated plan for domestic direct disposal of spent fuel was presented to the authorities in 1990. The repository concept comprises horizontal tunnels with vertical holes in the floors at a depth of several hundred meters in the crystalline bedrock. Fuel bundles are sealed in copper-steel-canisters prior to emplacement in the repository. The repository is planned to be constructed in the 2010's.

TRANSPORTATIONS

The spent fuel of Loviisa is transported to the Soviet Union by train in the Soviet wet flasks with a capacity of 30 assemblies. By March 1991, 10 transports have taken place. Also two leaking assemblies were included in one of the transports.

TVO has one flask for transfer of spent fuel assemblies from power plant units to the on-site interim storage facility. The wet flask has a capacity of 41 assemblies. In average 6 transfers take place annually.

RESEARCH AND DEVELOPMENT WORK

The site for final repository will be selected by the year 2000. Field investigations were started at five areas in 1987. The programme consists of airborne survey, deep and shallow drillings as well as measurements and sampling from the surface and in boreholes. Field work is followed by laboratory studies as well as modelling and evaluation activities.

Parallel to bedrock investigations, the technology of final disposal is being developed and optimized. Various long-term experiments are carried out for performance assessments of the repository system.

FINANCES

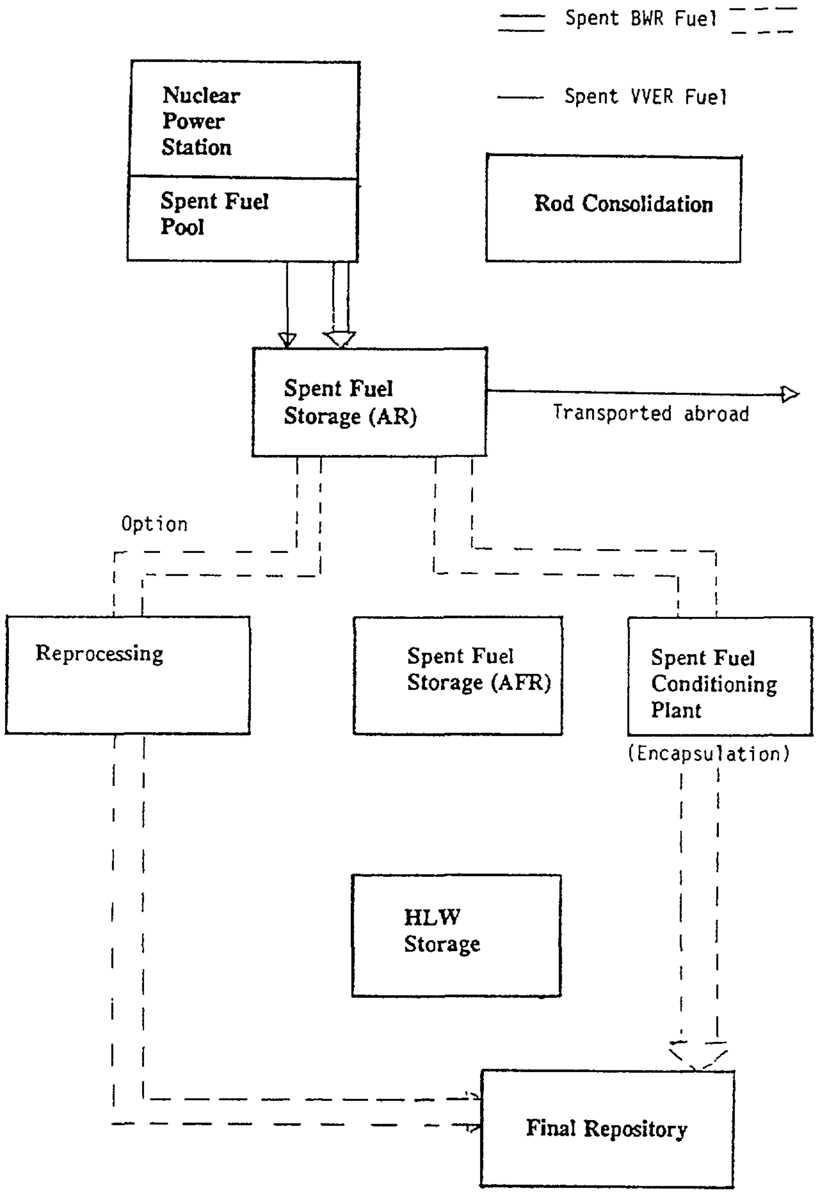
The utilities have to present annually updated cost estimates for nuclear waste management, including spent fuel, low- and intermediate-level wastes and decommissioning. Based on these estimates, the Ministry of Trade and Industry each year confirms the fee to be paid into a government-controlled fund.

The estimate includes the future costs for the management of the waste amount already produced. The costs are calculated at current prices without discounting. The funded money need not immediately cover the total costs. The difference has to be covered by securities.

Each annual fee has to increase the fund to the level corresponding to the ratio of the cumulative produced electricity and the total production during 25 operation years. By March 1991 IVO has paid MFIM 585 in the fund and TVO MFIM 1903. A utility is entitled in borrowing back 75 % of its own share of the fund.

The utilities TVO and IVO have signed information exchange agreements with the Swedish SKB, the Swiss NAGRA and the Canadian AECL and Ontario Hydro. Furthermore, the Finnish organisations have a number of contacts with the other waste management organisations especially in the countries studying crystalline rock.

The Finnish utilities, research institutes and authorities are participating in several international research projects, e.g. the Stripa project. There is lively exchange of information also within the expert groups of international organisations, e.g. the IAEA, the OECD/NEA and the Nordic NKS.



Spent fuel management in Finland

LWR SPENT FUEL MANAGEMENT IN THE FEDERAL REPUBLIC OF GERMANY

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1 Introduction

In its paper on management of waste from nuclear power plants and other nuclear installations put before the Deutsche Bundestag on the 13 January 1988 the Federal German Government reported on the present status and state of progress made in nuclear waste management. The present report is based upon the above paper and describes the present status of nuclear waste management in the FRG. The report reflects recent changes in the realisation of the different steps in the waste management.

2 Waste Management Concept

The safe management of waste ("Entsorgung") from nuclear power plants and particularly the orderly disposal of radioactive waste 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 waste is a precondition for the construction and 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. The resolution passed on 28 September 1979 by the heads of the State and Federal Governments confirms the integral waste management concept; it is to be based in general on onsite and offsite interim storage followed by reprocessing of spent fuel, recovery of radioactive materials, and conditioning for the disposal of radioactive waste. This waste management concept has continued to be put into practice in recent years in accordance with the projections contained in the 1983 Government report on waste management. The technical elements of the back end of fuel cycle strategy are unchanged in spite of the fact that the practical performance has been reorganized recently. More attention has been given also to the direct disposal approach.

Safety takes highest priority in waste management facilities. All those concerned with the waste management concept in the Federal Republic of Germany are confident that it can be implemented effectively and in due time.

The waste management concept embraces four significant steps:

1. Interim storage of spent fuel in the nuclear power plants and in offsite interim storage facilities.

2. Reprocessing of spent fuel and re-use of the nuclear fuel thus recovered in nuclear power plants (recycling).

3. Development of disposal for spent fuel for which, in accordance with Article 9 of the Atomic Energy Act, reprocessing is technically not feasible or economically not viable.

4. Disposal of radioactive waste
   in the stages
   - Conditioning
   - Interim storage in nuclear installations, in offsite stores or in regional collection centers
   - Intermediate storage of highly radioactive, heat generating waste (vitrified blocks) in interim storage facilities
   - Disposal

3 Interim Storage of Spent Fuel

3.1 Objective and Extent of Interim Storage

The resolution of the heads of the State and Federal Governments dated 28 September 1979 was based on the premise that interim storage facilities for irradiated fuel would have to be extended.

It specifically provides for the construction of offsite interim stores - at Ahaus and Gorleben - in addition to the onsite capacities. In their resolution the heads of government assumed that the desired reliability of waste management can only be assured in the individual stages - and therefore the offsite interim storage of irradiated fuel - are implemented continuously, in due time and effectively.

However, the interim storage of irradiated fuel from light water reactor is only an intermediate stage in waste management.

3.2 Forms of Interim Storage

Onsite storage is available at nuclear power plants for the interim storage of spent fuel generated in those plants. This takes the form of wet storage in the spent fuel pool. Offsite interim storage facilities consisting of shipping and storage casks are intended to receive spent fuel from all nuclear power plants and have been constructed or are under construction.

The total storage capacity available is composed of the following:

- Onsite storage at nuclear power plants approx. 5600 t are available in the old federal states and 250 t in the new federal states

- 3000 t in offsite interim stores in the old federal states and 560 t in the new federal states

  1500 t thereof in the Gorleben interim storage facility:
  This interim store received its license pursuant to the Atomic Energy Act on 5 September 1983 and 6 October 1988; the interim store is technically ready for operation.

  1500 t in the Ahaus interim storage facility:
  The procedure for licensing pursuant to the Atomic Energy Act has been in progress since 1979. The license according to Article 6 of the Atomic Energy Act for the storage of fuel from light water reactors was granted on April 1987. The

license to store THTR-fuel will be available in the first half of 1991. The construction of the storage facility performs in accordance with the time schedule.

560 t thereof in the pool storage at Greifswald:
That storage facility was commisioned in 1985. The license will be assessed in accordance with § 6 of the atomic energy act.

3.3 Basic Engineering Principles for Interim Storage

3.3.1 Wet Storage

On the grounds of experience a strong case can be made out that storage in water of spent LWR fuel in standard or high-density racks is based on a mature and viable technology free from technical difficulties. The available experience provides a substantial basis for concluding that Zirconium alloy clad fuel has not deteriorated to date.

3.3.2 Dry Storage

R+D work has been carried out to assess the integrity of spent fuel assemblies and their handling capability throughout dry storage periods of over a decade and more, during the discharge of the fuel from the stores and during shipment to fuel reprocessing facilities. The research programs covered theoretical predictions, laboratory word and performance testing on actual spent fuel.

The results of the R+D activities in the FRG can be summarized as follows:

- The experimental findings and the theoretical analyses are in agreement
- No indications of ISCC or crack growth in the Zircaloy cladding
- Tangential strain and cladding oxidation range around the detectability limit under cask storage conditions prescribed in the FRG

- Moisture can be removed during cask drying operations from rods which were degraded in operation
- Minimal contamination of the cask inner surface and basket
- No indications of any further propagation of cladding defects under inert conditions
- 420 °C is a safe and reliable upper temperature limit for dry storage in an inert atmosphere. Even somewhat higher temperatures seem to be acceptable.

4 Reprocessing

4.1 Objective

Reprocessing of spent fuel from light water reactors and recycling of the recovered nuclear fuel in nuclear power plants are essential components of the waste management concept of the FRG.

Reprocessing of spent fuel is of particular importance for ensuring secured power supplies in the medium and long-term. The nuclear fuel recovered during reprocessing provides raw material savings (up to 40 % of natural uranium) through recycling in present-day light water reactors.

Reprocessing of spent fuel and the use of the recovered radioactive materials also have a beneficial effect with respect to the disposal of radioactive waste from the use of nuclear energy. The quantity of high-level waste is reduced as is the proportion of long-lived radioactive substances.

4.2 Present situation

4.2.1 Fuel from reactors situated in the old federal states

The French/German Memorandum of Understanding between Cogéma and VEBA, and the subsequent joint Government Declaration on Nuclear Cooperation, particularly in the field of reprocessing, sealed the fate of the German 350 tonnes HM reprocessing plant project at Wackersdorf.

In November 1989, the German utilities finalized terms of an umbrella agreement with both Cogéma and BNFL which was approved by the German Minister for Environment, Nature Protection and Reactor Safety (BMU).

The fixed term of the contracts to be concluded with Cogéma is to be from 1999 to 2005, while th planned contracts with BNFL should run from 2002 to 2005. Under either contract there will be two consecutive options of 5 years each, so that the contracts could be extended until 2015.

4.2.2 Fuel from reactors situated in the new federal states

The spent fuel is for the time being in store at the reactor sites or in the central store. The "Energie Werk Nord" is negotiating with "Technaps Export" the possibility of closing the back end of the fuel cycle in cooperation with the USSR. In addition negotiations are underway with Cogéma and BNFL.

5 Conditioning, Interim Storage and Disposal for Radioactive Waste

Additional capacities for interim storage of radioactive waste were created with the opening of the store in Gorleben and the interim store in Mitterteich. In view of the available interim storage capacities for radioactive waste with negligible heat generation rates bottle necks may occur, if the Konrad final repository is not commisioned in time, since in 1996 - in some cases even earlier - the storage capacity of available or definitely planned facilities are exhausted.

The Gorleben repository is planned to receive all types of radioactive waste, particularly highly radioactive, heat-generating wastes; above-ground exploration of the salt deposit has been carried out.

Below ground exploration were initiated in September 1986 with the sinking of shaft No. 1. A mining accident in May 1987 did not cast doubts on the suitability of the salt deposit.

Exploration work at the planned German radwaste repository at Gorleben restarted early in 1989 after having been delayed since May 1987.

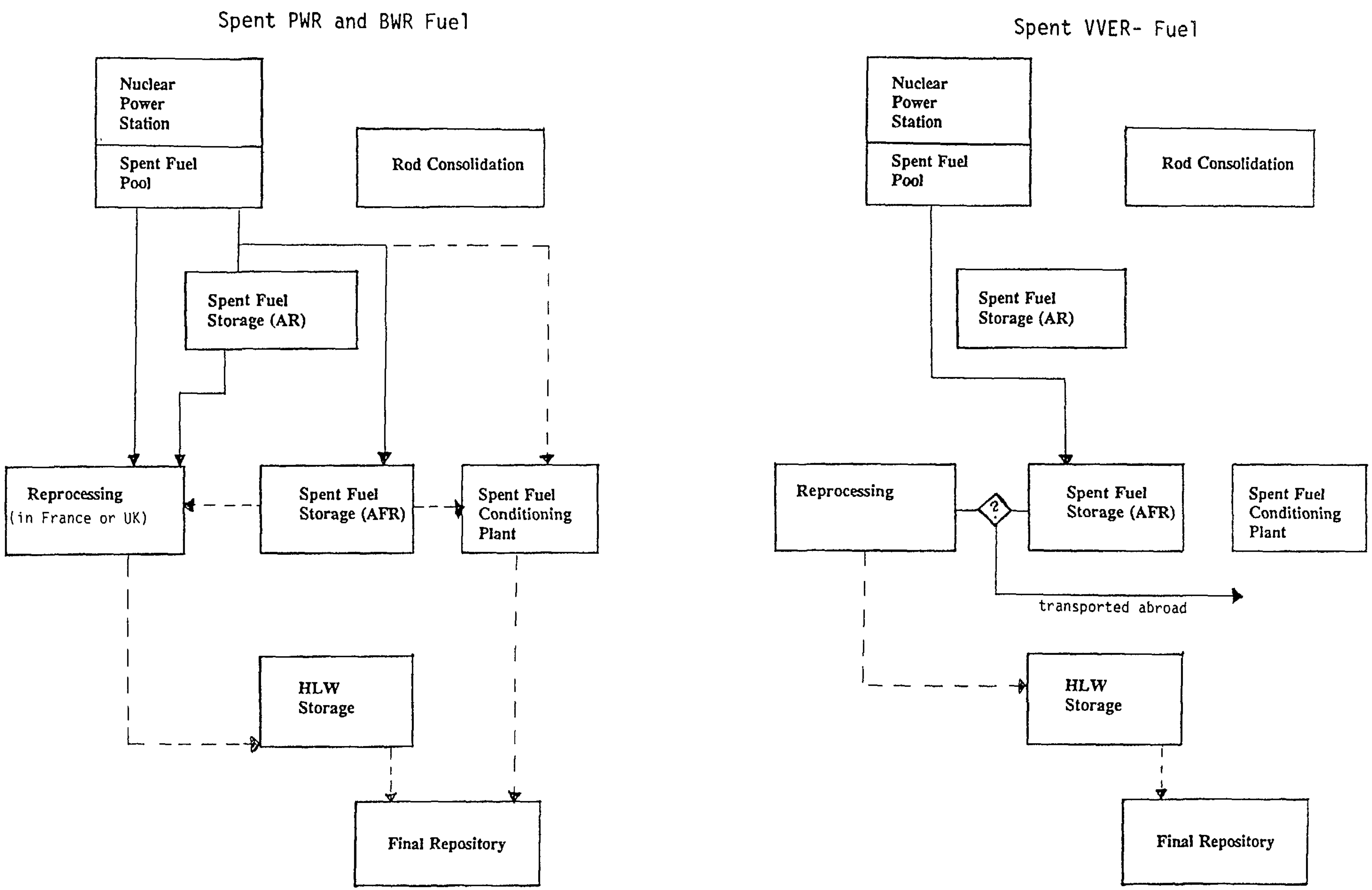
The Morsleben repository in the new federal states is in operation since 1981. The safety situation of this repository will be reassessed to answer questions recently raised. Deposition of wastes is interrupted due to court decissions.

6 Direct Disposal

The feasibility of direçt disposal of spent fuel was investigated as called for in the heads of government resolution of 28 September 1979; the safety evaluation was concluded on time.

On 6 May 1986 the German reprocessing company (DWK) submitted an application for a lincense pursuant to Article 7 of the Atomic Energy Act to construct and operate a pilot conditioning plant in Gorleben for the purposes of developing conditioning techniques for direct disposal.

In summer 1989, DWK's Board of Directors decided to construct the PKA, which is considered a necessary preliminary stage for the direct final disposal of spent fuel elements in Germany. Thus, the PKA will explore a supplementary back-end solution to reprocessing. In December 1989, the local authorities at Gorleben granted the ordinary construction license for the PKA project.



Spent fuel management in Germany

SPENT FUEL MANAGEMENT IN HUNGARY: IMPROVEMENT OF WATER STORAGE TECHNOLOGY OF SPENT FUEL

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ERÖTERV,
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1. Introduction

Nuclear electricity generation in Hungary is performed by the Paks Nuclear Power Plant. Four units of the NPP are now in operation, each with a WWER-440 type 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.

For the year 1989 47.5 % of total electricity generation of the country was supplied from this plant.

The fresh fuel is imported from the Soviet Union and according to the agreement between the Operator of the plant and the Supplier, the spent fuel can be shipped back to the Soviet Union 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, from unit 1 using the standard Soviet Railway Transport Unit, which includes 4 containers of the TK-6 type.

This means that the national strategy for the back-end of the NPP nuclear fuel cycle is motivated by the possibility to send the spent fuel back to the Soviet Union for a later reprocessing.

2. Fuel Systems and Manipulation

Main Fuel Data

There are 312 operating and 37 control assemblies in the reactor core. Fuel assemblies are of hexagonal cross section, one bundle encloses 126 fuel rods. The cladding of the fuel rods and the wall of the bundle are made of zirconium alloy. The maximum enrichment of bundles in steady state 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_2, specific thermal output is about 85 kW/l; this low value contributes to the problem-free behaviour of fuel bundles.

Fuel Systems

Two of the reactor units are situated in a common main building, so that certain elements of the fuel systems could be made to serve both. For example, the 2 units have a common fresh fuel storage room in the central part of the building, a single reloading machine is used, which can be moved from one unit to the other by a crane. If storage racks of the spent fuel pool become full and an extra discharge is required there is a special emergency rack which is again common.

The spent fuel is stored in at-reactor pools which are situated in the common main building and are connected through sluice gates to the reactor pits respectively.

Spent Fuel Manipulations

After refuelling, the spent fuel is located on the racks of the spent fuel pool, where its heat release and activity is reduced to the value required by the transport conditions. Each bundle is checked for leak-tightness when removed from the reactor. Assemblies with not perfectly tight elements are placed into a hermetic casing, situated on the rack, and remain in that also during transport. Removal from the reactor hall is performed in a railway container. A maximum of 30 bundles at a time can be placed in the transport container developed for WWER-440 reactors.

3. Spent Fuel Storage at the site of the NPP

A three years' storage of spent fuel elements removed from the reactor has been envisaged by the Technical 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 assemblies. This was achieved by a subcritical lattice structure without built-in absorbers.

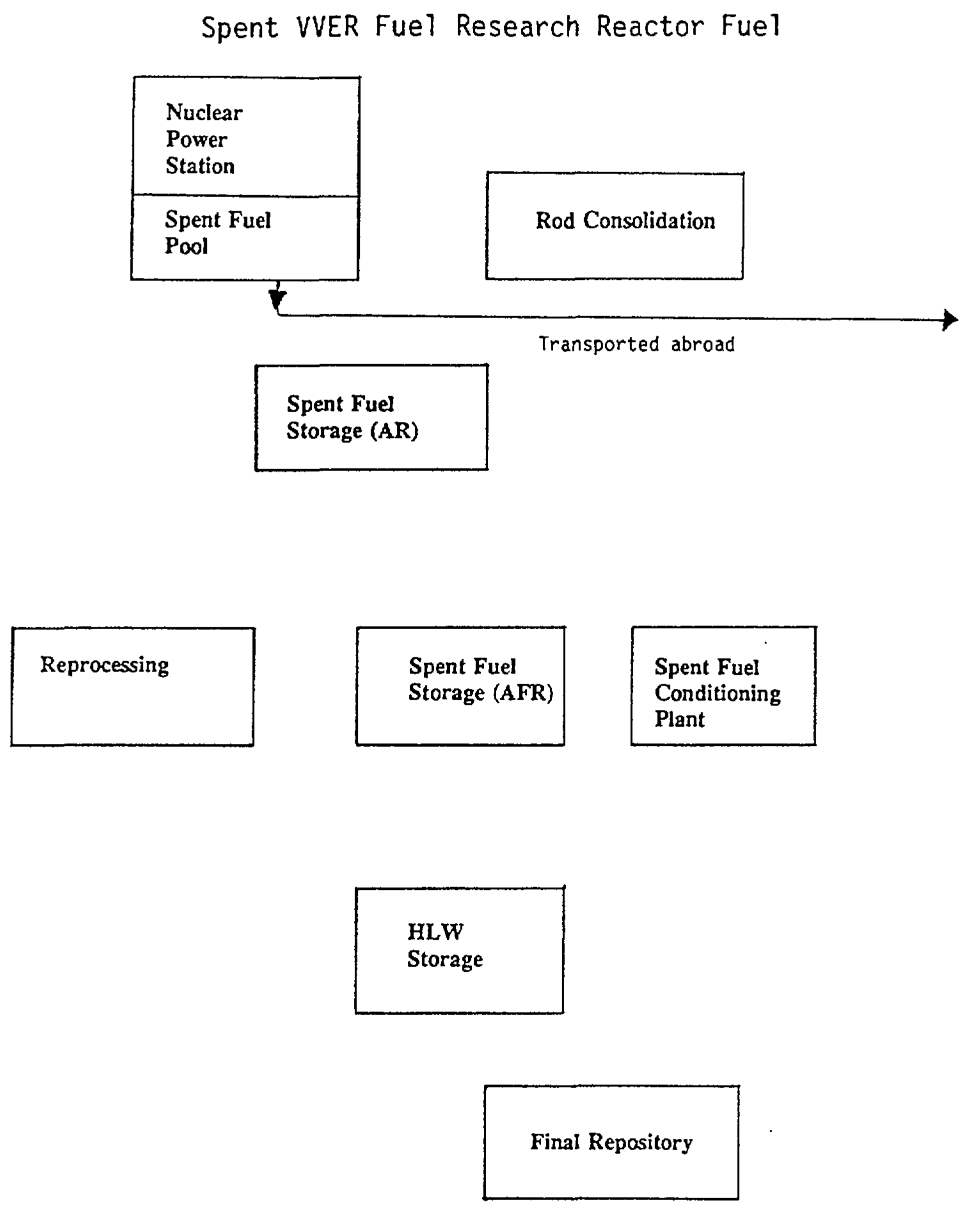
In 1979 the Soviets put in a claim to the extension of spent fuel at-reactor storage at least for 5 years.

Different possibilities were analyzed in order to enlarge the storage capacity at site.
One option was the reconstruction of the existing pools using compact storage racks.

The final decision was made in 1984: the original racks had to be dismounted from the spent fuel pool of the unit 2 and the new structure providing compact storage had to be mounted. This task was successfully accomplished by the scientific-, designing-, manufacturing-, erecting- and licensing organs involved and in July 1985 the first Hungarian compact storage system was ready to receive the assemblies unloaded from unit 2.

After the successful carrying out of the compact storage pool at unit 2 the other three storage pools were reconstructed, too.

In the new storage pool there are 706 cells, of which 650 pcs are normal storage cells and 56 pcs are hermetic casings.



Spent fuel management in Hungary

ITALIAN ACTIVITIES IN SPENT FUEL MANAGEMENT

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General

After the 1987 referendum the Italian Governement decided a moratorium on its nuclear energy activities; accordingly, the LATINA (GCR) power station was definively shut down and the stations of TRINO (PWR) and CAORSO (BWR) were put on standby.
In July 1990 the Governement resolved to definitively shut down TRINO and CAORSO and to decommission the four Italian power stations.
Meanwhile ENEA had also determined to decommission its nuclear fuel cycle pilot plants.

Power station fuel

Magnox fuel from the LATINA reactor has been transferred to Sellafield to be reprocessed by BNFL.
After the decision to decommission the power stations, the fuel at the light water reactors and that at the Avogadro Intermediate Storage Facility will be also reprocessed abroad. In Italy the direct disposal of spent fuel has never been considered.
Table I shows the amount of spent fuel from the Italian LWR reactors and its present location.

Fuel at ENEA's facilities

At ENEA's facilities there are small quantities of spent fuel for which reprocessing at the EUREX and ITREC pilot plants had been envisaged; these are shown in Table II.
It has been decided to shut down and decommission these plants, this fuel therefore will be sent to foreign plants for reprocessing.

Status of ENEA's research programme

Latina fuel transfer. The shipment to Sellafield of the LATINA Magnox fuel stored at the EUREX pool was completed in July

TABLE I. LIGHT WATER REACTOR SPENT FUEL IN ITALY

STATION	POWER MWe	TYPE	COMMERCIAL OPERATION	FUEL IN CORE, tU	AR STORAGE CAPACITY, tU	SPENT FUEL IN 1986, tU	STORED AR,tU	STORED AFR*,tU	REPROCESSING **, tU
GARIGLIANO	160	BWR	1964	-	-	68	-	68	-
TRINO	260	PWR	1965	35	50	58	23	15	20
CAORSO	860	BWR	1981	102	397	117	117	-	-

* Avogadro Facility

** BNFL - Sellafield

TABLE II. SPENT FUEL AT ENEA'S FACILITIES

PLANT	FUEL	DESTINATION
SALUGGIA	TRINO - 52 PWR cruciform elements - 1.93 tU MTR - 150 elements - 152 kgU	removal from pool or reprocessing reprocessing by DOE (USA)
ITREC POOL TRISAIA	ELK RIVER - 64 elements - 1.69 tU-Th	removal from pool
OPEC HOT CELLS CASACCIA	CIRENE and other - 120 pins - 92 kg U-Th-Pu	removal from Hot Cells

1990. Further information on this project will be presented in a report at this meeting.

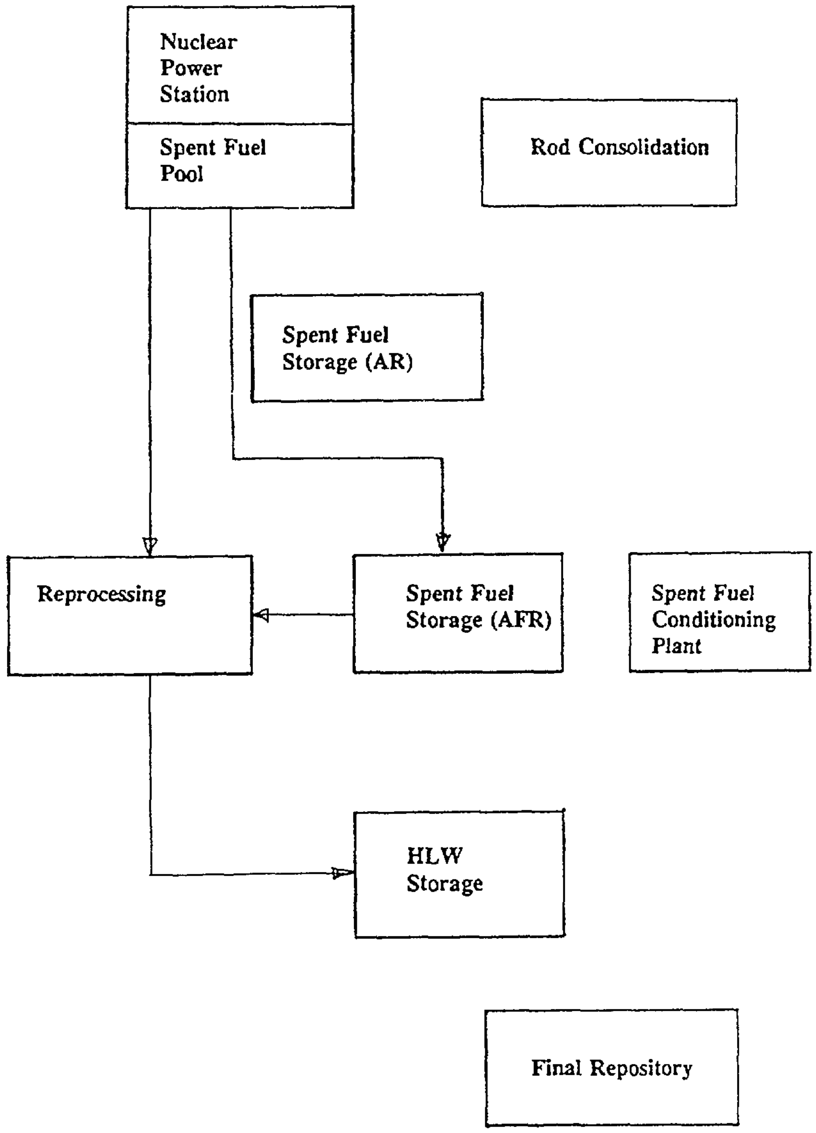
Trino fuel dry storage. The preliminary design of a cask for the dry storage of the TRINO fuel has been carried out; but this activity has now been discontinued in favour of foreign reprocessing of the fuel.

Elk River fuel dry storage. This fuel is being placed in sealed bottles and in the process of being monitored to check its integrity.

As in the case of the TRINO fuel, reprocessing abroad is being preferred to dry storage in a cask; this cask was in the preliminary design stage.

OPEC fuel storage. The pins of the PIE fuel elements at the OPEC cells have been put into 5 containers and stored in the warehouse of the facility. This material with the rest of the PIE fuel in the warehouse will be shipped for foreign reprocessing.

Future plans. Once the spent fuel is shipped from ENEA's facilities, no further plans are envisaged in this field at this stage.
Activity at ENEA will be devoted instead to waste treatment and to the decommissioning of the installations.



Spent fuel management in Italy

LWR SPENT FUEL MANAGEMENT IN JAPAN

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The strategy of spent LWR fuel management of Japan has its base in reprocessing of spent fuel and subsequent utilization of recovered uranium(U) and plutonium(Pu) as nuclear fuel.

To the end of 1990, Japan operated 39 nuclear plants with total installed capacity of 31.5GWe. In the year of 2000, the capacity is estimated to be approximately 50 GWe and the annual spent fuel arising will be 1200 tons per year. The accumulated quantity of spent fuel by the end of FY1989(March,1990) reached about 7300 tons, of which about 2250 tons of spent fuel being stored in water pools at reactor sites. The other part of the spent fuel has been shipped for reprocessing in domestic or foreign facilities.

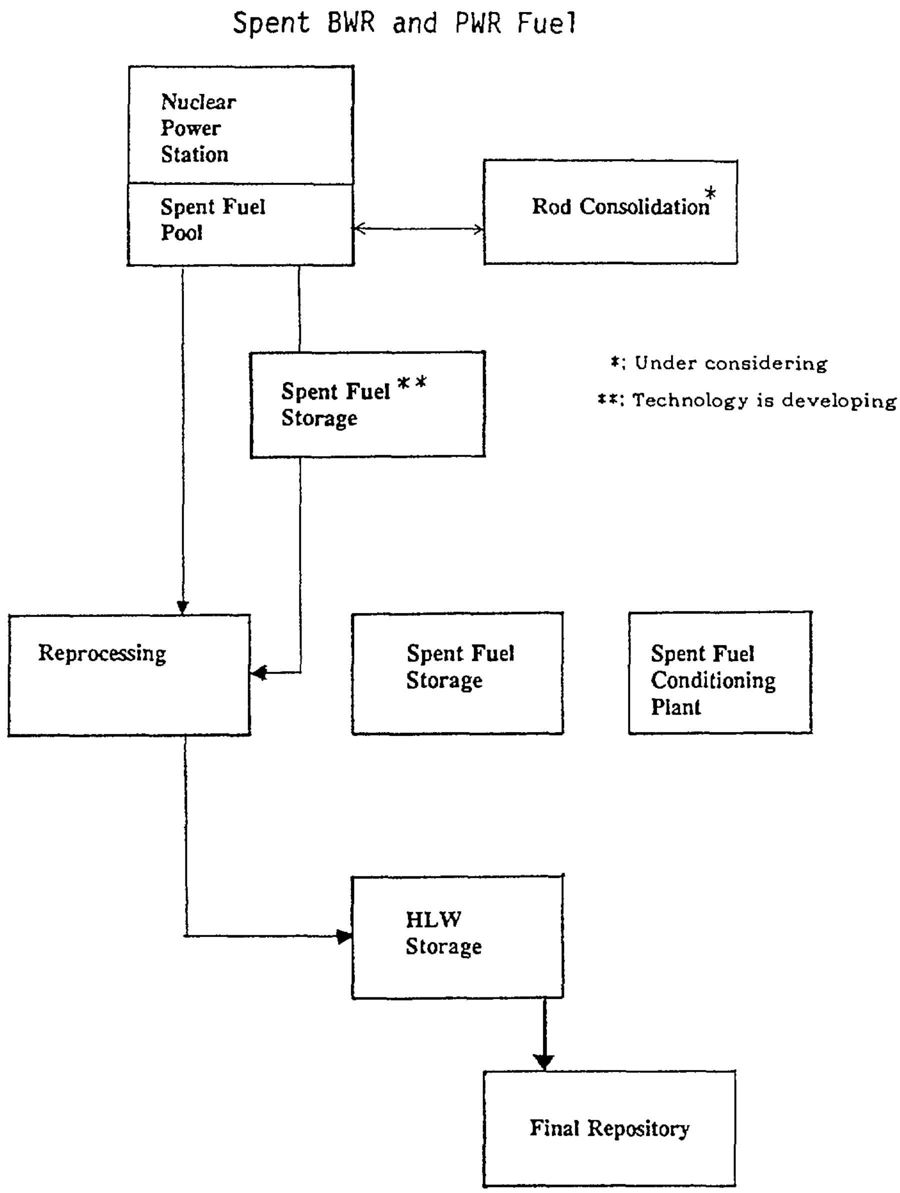
Under the policy of reprocessing all of the spent fuel and utilizing recovered U and Pu, the Tokai reprocessing plant(TRP) with capacity of 0.7 ton per day is operated by Power Reactor and Nuclear Fuel Development Corporation(PNC). Up to the end of 1990, about 510 tons of spent fuel, including a small amount of MOX fuel from an advanced thermal reactor(ATR,Fugen) has been reprocessed in the plant.

Construction of a commercial reprocessing plant with capacity of 800 tons per year is planned in Rokkasho-mura, Aomori-prefecture. Japan Nuclear Fuel Service Co., LTD is responsible for the construction and operation. Hot commissioning of the plant is scheduled to start at 1997.

On the other hand, most electric companies in Japan have contracts for reprocessing their spent fuel with foreign firms; COGEMA in France and BNFL in United Kingdom. About 5600 tons of LWR spent fuel is contracted with both COGEMA and BNFL, and 1100 tons of gas cooled reactor(GCR) spent fuel with BNFL, for reprocessing. To end of 1989, about 4500 tons of spent fuel from both LWR and GCR had been transported to the facilities of COGEMA and BNFL.

Although, Japanese national policy of the spent fuel management is that all of the spent fuel is reprocessed, spent fuel arising exceeds the amount of reprocessing now. This situation is afraid not to be eliminated in short term future. In such situation, a part of spent fuel is planned to be stored safely. So, enlargement of the storage capacity must be prepared. Recently constructed nuclear plants have spent fuel storage pools of which capacities are larger than former ones, and reracking was made on a part of the storage pools already constructed.

Studies on spent fuel storage are being conducted at JAERI, CRIEPI and private companies. The subjects have wide spectrum such as economic comparison among storage methods, fuel behaviors in inert and oxidizing atmospheres and data acquisition on cast iron casks for the quality assurance. Private companies are developing technology on vault facility and transportation and storage cask.



Spent fuel management in Japan

SPENT FUEL MANAGEMENT IN THE REPUBLIC OF KOREA

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In Korea, nuclear power has been an important part of the national energy system ever since the first commercial operation of nuclear power plant in 1978. Nine nuclear power plants, 8 PWRs and 1 CANDU, are now in operation with generating capacity of about 7,600 MWe. Nine nuclear power plants currently in operation account for about 36% of the total installed generating capacity. In addition, two more 990 MWe class PWRs will be completed in the middle of the 1990's. This nuclear power programme entails the management of spent fuels discharged from plant operation. They are currently stored in each unit's At-Reactor(AR) pools of which total accommodation capacity is about 2,730 MTU. The accumulated amount of spent fuels up to now is about 810 MTU and this accommodation is projected to reach about 2,810 MTU by the year 1995, about 3,500 MTU by the year 1997 and about 4,600 MTU by the year 2000. Moreover, the accumulative amount of spent fuel discharged from these 11 units for the whole life time will be approximately 11,500 MTU. This accumulative projection indicates shortage of Full Core Reserve(FCR) storage capacities of those 11 units in mid-nineties except Kori-2 and Yeongkwang-2.

Under the current situation in which the Korean government has not established a definite policy to either recycle or permanently dispose for the long term management of spent fuels, the best we can do is to store them safely in the meantime. Korean Atomic Energy commission(AEC) has set forth a resolution(July 1988) that Away From Reactor(AFR) storage be built by the end of 1997 as an interim storage facility to ease out the mid-term spent fuel management problem. Further Atomic Energy Act amended in May of 1986 promulgated the government responsibility for spent fuel management and designated Korea Atomic Energy Research Institute(KAERI) as the responsible body for the construction and operation of the storage facility.

Meanwhile, Korea Electric Power Corporation(KEPCO) will take care of the spent fuel accumulating in the existing AR pools by suitable means up to the time of handover to AFR. Examination of the suitable options for KEPCO's AR storage management shows the possibility of transshipment of Kori-1 excess spent fuel to Kori-3 and 4 at first hand taking advantage of their presence at the same Kori site.

At Uljin site, reracking with poisoned rack system should allow considerable expansion of existing capacities up probably to the end of 1997 as it is planned by KEPCO. Maximum use of existing system of cooling and purification is thought to be adequate without modification, even though further detailed checks would be needed.

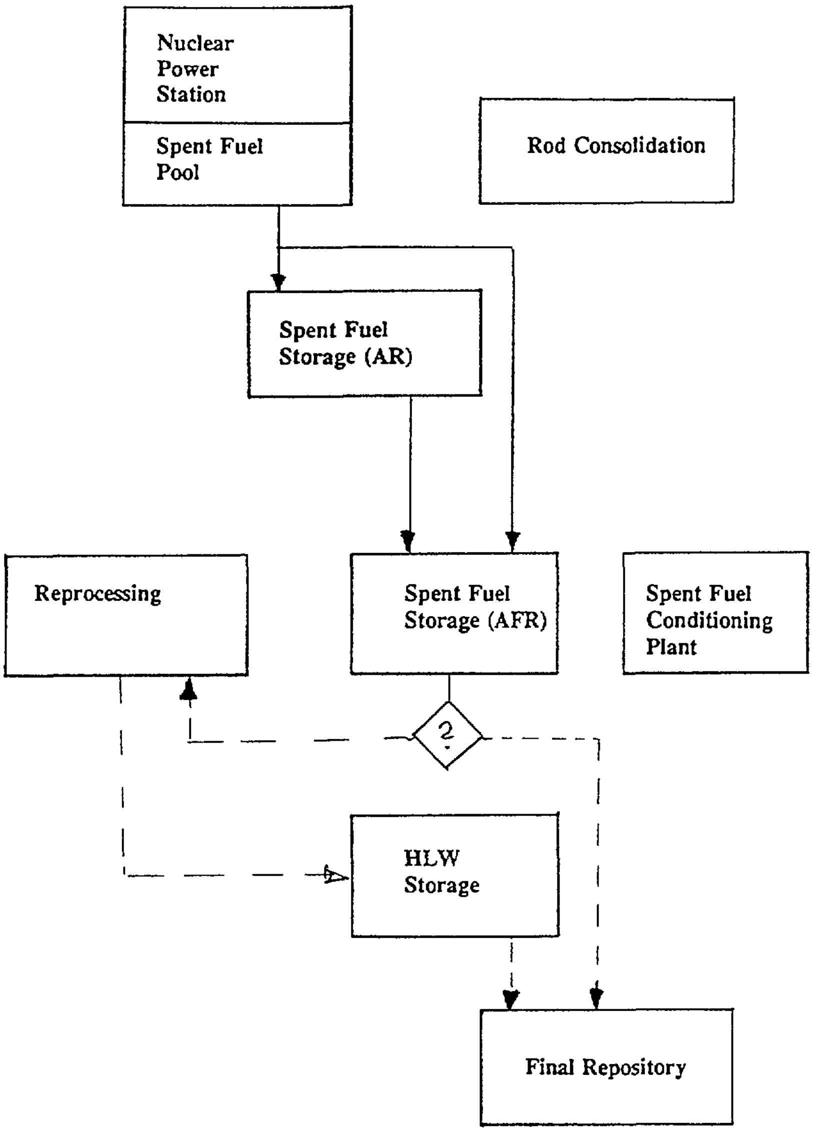
For the CANDU fuels at Wolsung site, possible capacity expansion with shorter tray stackup is estimated to be only marginal two more years. Considering the FCR capacity of Wolsung-1 up to early 1992, new option to manage the excess amount up to the target year 1997 is inevitable. KEPCO is presently interested in one of the dry storage concepts.

AR expansions for spent fuel storage are not the ultimate solution for accommodating the whole spent fuel up to the life time of each unit. KAERI has carried out feasibility studies on the problem for future interim storage strategies.

The wet type interim storage facility of which the conceptual design was finished in 1990 will be entered into basic and detail design phase to be followed by start of construction work provided that the suitable site is selected. The storage capacity of the first facility will be 3,000MTU.

To implement the spent fuel management programme, transportation system development is an important factor to be integrated in the programme. KAERI's shipping cask No 1, KSC-1, which can transport one spent PWR fuel assembly of any type including 14x14, 16x16, and 17x17 was developed in 1986 and since then has been employed to transport 5 spent PWR fuel asssemblies from nuclear power plant to the KAERI's post irradiation examination(PIE) facility for hot cell examination via road. Design, licensing and fabrication of KAERI's shipp-ing cask No 4, KSC-4 were carried out from 1987 to 1990. 4 PWR fuel assemblies can be transported by KSC-4. Using this cask, transshipment of spent PWR fuel assemblies from Kori-1 to Kori-3 nuclear power plant was carried out. A total of 28 PWR fuel assemblies has been transshiped successfully to date. Using the technology accumulated during development of KSC-1 and KSC-4 shipping casks, a number of large size shipping casks will be developed for transportation of spent fuel from nuclear power plants to the interim storage facility. Besides, an appropriate transportation system including spent fuel and cask handling device will be developed along with the porgramme implementation of the interim storage facility.

Spent PWR and PHWR Fuel



Spent fuel management in the Republic of Korea

SPENT FUEL MANAGEMENT IN THE UNION OF SOVIET SOCIALIST REPUBLICS

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To date nuclear power in the USSR is based on the reactors of WWER-440 and 1000, and RBMK-1000 and 1500 types. Power plants with breeders of BN-350 and BN-600 types are also in operation; 46 power units are working with spent fuel arisings of 1300 t/y.

The concept of spent fuel management adopted nowdays is to store fuel in AR or AFR facilities for 3-10 years before shipment for reprocessing or long-term storage(BEFAST).

Wet storage is the preferred mode and will evidently remain as such during next 20-30 years due to its high reliability and economical efficiency.

There are two basic types of pools currently in use: AR pools receiving fuel directly from the reactor for cooling during 3-year period, and pools, usually of AFR type for interim storage during 10 years or more (fig. 1 - 3).

The safety provisions in AR and AFR pools include:

- steady ventilation of the above water area due to a light removable ceiling in the pool building;
- SS lining of the pool inner surface;
- systems for leak localization and control;
- systems of water cooling and purification, maintenance of water level, pool filling and dewatering, technological and radiation monitoring;
- storage of defective fuel in sealed individual cans; storage of spent fuel in baskets placed at the pool bottom (WWER fuel) or in cans hunged from the ceiling (RBMK fuel) under a

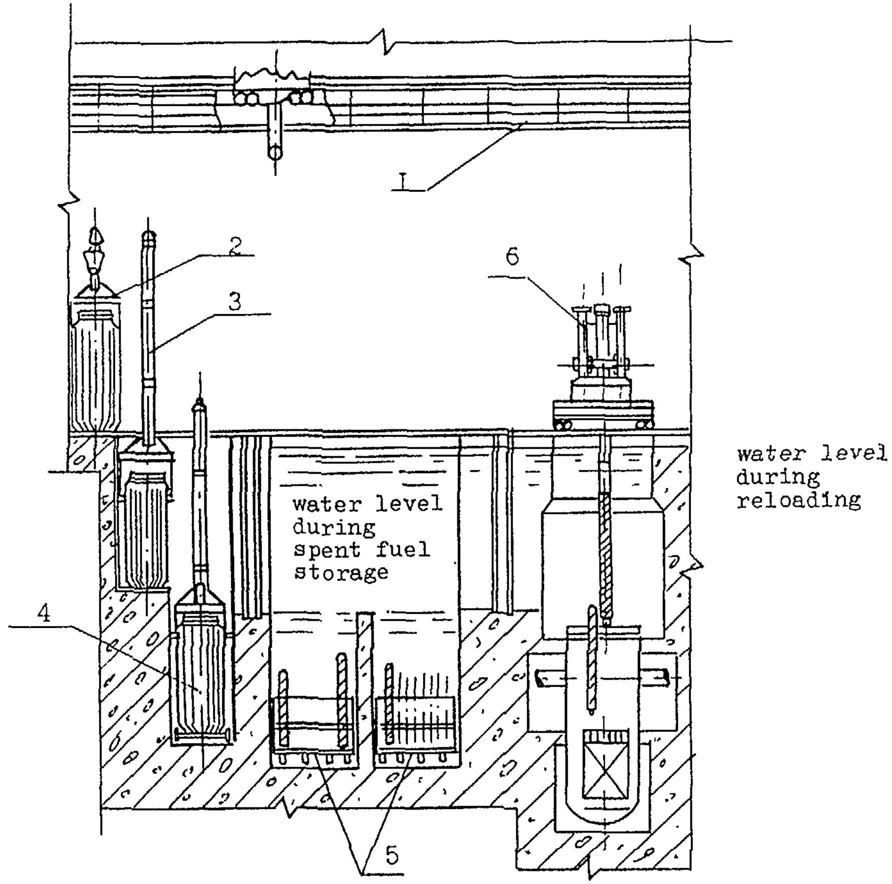

Fig. 1. AR cooling for SF of WWER-1000 reactor:

1 - circular electric crane, 320+160/2+70 load capacity;
2 - traverse for spent fuel container; 3 - rod for container;
4 - shipping cask; 5 - cooling pool racks; 6 - loading machine.

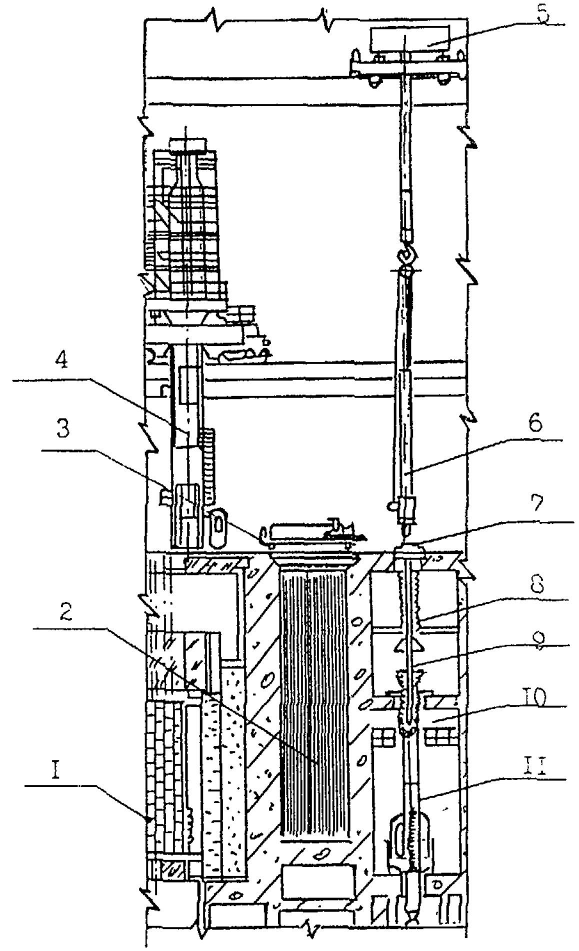

Fig.2. AR cooling pool of RBMK spent fuel:

1 - reactor; 2 - cooling pool; 3 - floor beam-crane;
4 - loading machine; 5 - travelling crane of 50/10 t load capacity; 6 - reloading container; 7 - charge device;
8 - guide shaft; 9 - basket; 10 - guiding device;
11 - transport container.

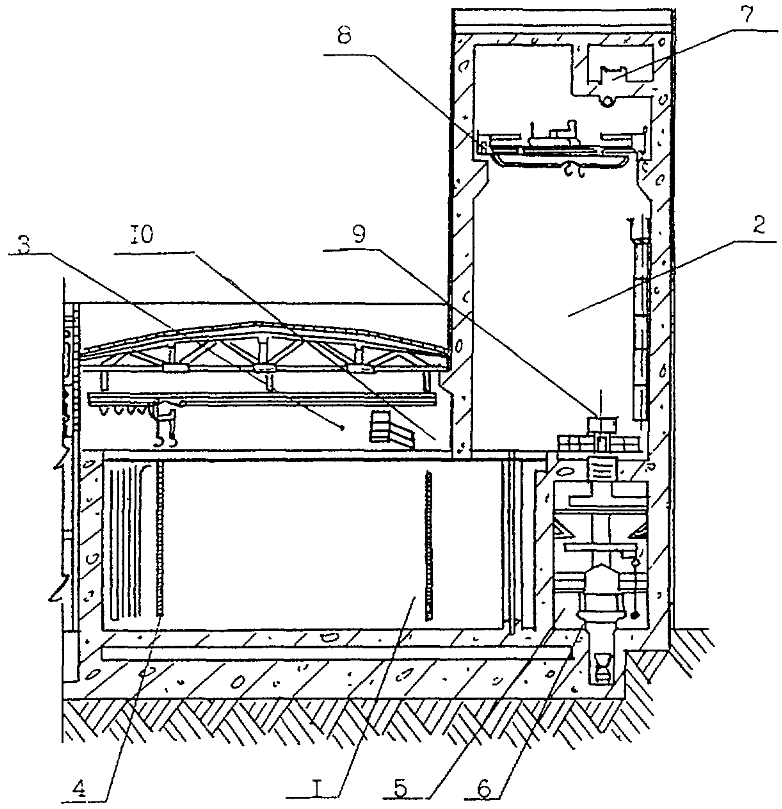

Fig.3. ISFSI for RBMK fuel:

1 - storage pool; 2 - main hall; 3 - storage section; 4 - cans with fuel assemblies; 5 - transport entrance; 6 - transport container; 7 - 15t cable trolley; 8-20/5t travelling crane; 9 - guiding device; 10 - transfer device.

water shielding layer of about 3 m thickness above the active part of assemblies.

Storage facility designs consider the risk of emergencies, such as:

-loaded containers hanging and dropping;

- cooling system failure ;

- power supply failure.

The permission for pool construction and operation is granted only after careful revision of corresponding design documents and approval by competent authorities (both,national and local), ex. GOSATOMNADZOR, GOSATOMTECHNADZOR,GOZATOMINSPEKTZIA.

Long-term operation experience shows high corrosion resistance of intact fuel. Moreover, no further deterioration of defective fuel has been observed after several years of pool storage.

Research programmes aimed at improving storage technologies and saving capital and maintenance costs have been fulfilled. These include:

- long-term (more than 10 years) storage behaviour of spent fuel with special emphasis on fuel with cladding defects;
- increase of the current pool capacity;
- water quality improvement; amendment of quality specifications;
- corrosion behaviour of pool component materials with the prospective of future replacement of the current materials for cheaper ones;
- contribution of pool water sludge to the radiation field growth in storage pools;
- dry transportation and storage.

It's been found out, that zirconium fuel cladding and SS pool components have perfect corrosion resistance in the pool water environment.

Deposition of corrosion products (CPs) suspended in the pool water contributes greatly to the growth of the surface oxide film and sorption of radionuclides. In general, purification systems are quite efficient in removing CPs from the pool water. However, a layer of sludge has always been found at the pool bottom.

Physicochemical studies of CP behaviour, in particular, of radionuclide sorption were carried out with the aim to develop methods for reducing radiation levels in long-term storage pools up to 30 Bq/l. These methods were tested at the storage facility of the Leningrad NPP.

Dense storage of RBMK fuel (3-5 t HM $/m^2$ of pool area) in currently operating facilities has also been considered.

Demonstration studies of dry storage after cooling in water pools are performed:

- with WWER-1000 fuel assemblies in TK-13 containers after 3-year wet storage. At the moment of loading into the container the fuel had 3% enrichment, 27.3 GWd/t burnup and ~ 22 KW heat output;
- with individual RBMK fuel elements after (8-10)-year cooling. In this case 3-year storage in air caused no serious impact on corrosion resistance and mechanical properties of fuel cladding.

During long-term storage in dry air and > 200°C oxidation of zirconium cladding and stress corrosion cracking of SS components is possible. In inert atmosphere the storage temperature 400°C is allowable. For Soviet designs of long-term storage facilities the recommended storage temperature is 125°C in dry air, and 350-400°C in inert atmosphere. The heat output from a spent RBMK assembly decreases from 0.07 KW after 5-year storage to 0.02 KW after 30 years. Hence, the temperature in the center of a dry storage cask won't exceed 270°C. This fact is very important from the viewpoint of the long-term dry storage of RBMK fuel assemblies, which are not reprocessed now. Also the technology of long-term dry storage in transportation casks is under active study now.

NATIONAL STRATEGY IN THE UNITED KINGDOM

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Strategy papers have been published in IAEA TECDOC papers 416 (1986) 487 (1988), and 580 (1990). The major features remain unchanged and are summarised below. Also attached is a flow chart summarising the overall spent fuel management routes currently pursued in the UK together with future options.

1 Reprocessing of Magnox fuel will continue to the end of the programme and some 2,000 tU of AGR fuel will be reprocessed in THORP. Revised contracts for these services and their associated waste treatment and storage services are being negotiated with BNFL by Nuclear Electric plc and Scottish Nuclear plc.

2 Alternative options for the management of the remaining AGR fuel beyond the current THORP commitment including prompt or deferred reprocessing, dry storage and direct disposal, remain under review. The final choice will principally be dictated by economic considerations.

3 Nuclear Electric's PWR spent fuel management strategy remains flexible but it is expected to make full use of the at-reactor pond storage facilities.

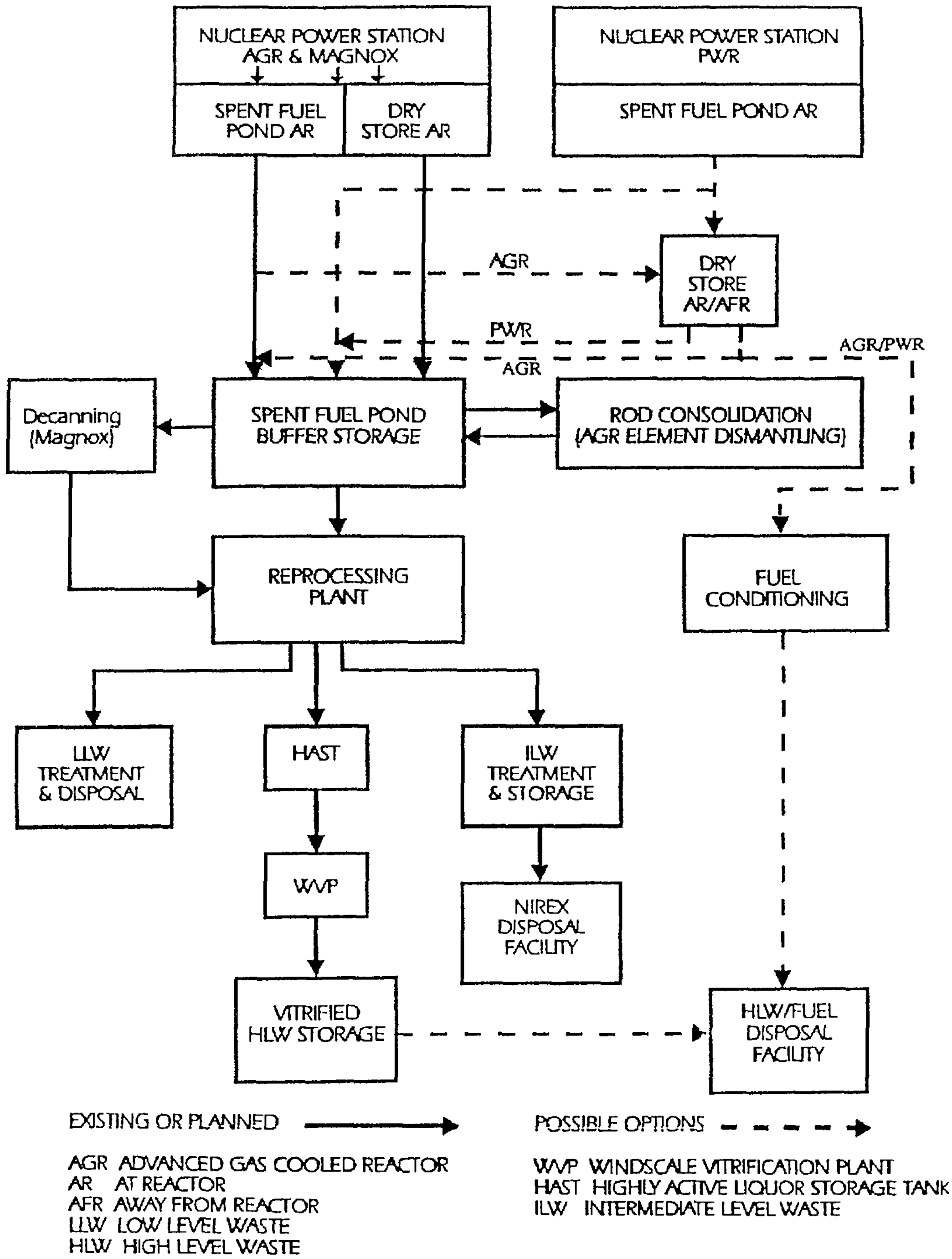
4 Uranium derived from Magnox reprocessing continues to be used in AGR fuel. The use of uranium from THORP reprocessing will depend on the balance of risks and reward.

5 The capacity of THORP in the first 10 years of operation has been increased from 6000 tU to 7000 tU. Sixteen out of the eighteen overseas customers who have rights under baseload agreements have taken up their share of the additional capacity. Fixed price offers have been made to a number of utilities including Nuclear Electric and Scottish Nuclear for post-baseload reprocessing in THORP.

6 A demonstration manufacturing facility for PWR MOX fuel is being established at Sellafield. This is a joint AEA Technology/BNFL project based on an existing AEA facility and will incorporate the BNFL short Binderless Route which has been under development since 1985. The plant will have an output of about 5 tHM/year and is expected to be operational in 1993.

7 Following a public inquiry into a joint application by AEA Technology/BNFL for outline planning permission for a demonstration fast reactor fuel reprocessing plant, (EDRP), sited at Dounreay, consent approval was given in late autumn 1989. However, Government funding of the prototype fast reactor (PFR) at Dounreay is to cease after 1994. Corresponding support for fast reactor fuel reprocessing is to cease beyond 1997.

8 Two candidate sites have been identified for the UK's repository for intermediate and low level wastes. These sites are Sellafield and Dounreay. Test drilling to determine the suitability of the Sellafield site is currently being carried out. NIREX advises that they expect the repository to become operational in the year 2005.



Spent fuel management in the United Kingdom

STATUS OF SPENT FUEL STORAGE IN THE UNITED STATES OF AMERICA

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Based on the rapidly growing need for additional on-site spent fuel storage, U.S. utilities are now actively planning their fuel storage strategies. Assuming maximum utility pool reracking and no new nuclear orders, the expected cumulative demand for additional on-site storage is 890 MTU by 1995, 2910 MTU by 2000, 6390 MTU by 2005, and 11,600 MTU by 2010. This on-site storage demand will be substantially reduced if the DOE's Monitored Retrievable Storage (MRS) facility can be put in place by 1998 as planned

Several utilities are already well along in either planning or implementing their long-term storage strategy. In addition to Virginia Power, Northern States Power has selected metal casks for their storage technology and is now involved in procurement and licensing activities. Baltimore Gas and Electric now has begun construction of a NUHOMS facility and will join Duke Power and CP & L as licensed users of this horizontal concrete silo technology. Vertical concrete casks are being planned for use both at Consumers Power and at Wisconsin Electric Power. Construction is also underway at the Fort Saint Vrain Plant of Public Service of Colorado to build a small modular dry storage vault designed by Foster Wheeler for HTGR fuel.

In the licensing area, there have been important recent actions taken by the NRC. The NRC has issued a rule that implements the Nuclear Waste Policy Act directive of approving storage technology at power plant sites without, to the maximum extent practical, the need for additional site-specific approvals by the Commission. This approach has been referred to as generic licensing. To date the NRC has only approved the action for metal casks having previously approved Topical Reports. Requests to be included under the rule are being submitted by vendors of concrete casks and NUHOMS storage modules.

In the area of spent fuel storage research and development, significant progress has occurred in programs under both DOE and utilities sponsorship. DOE's Behavior of Spent Fuel in Storage project has been conducting research to obtain data to support recommendations of limits for interim storage of spent light-water reactor fuel. For the last four years, spent fuel fragments and defected rod segments have been exposed to humidified air to obtain oxidation data relative to dry storage of spent fuel. High weight gains and cladding splitting due to oxidation of irradiated UO_2 have been observed while no dependence on burnup or cover gas moisture levels have been observed.

The current status is that the exposure of irradiated specimens to air was concluded at the end of FY 1990. Activities during FY 1991 are concentrating on the preparation of a report presenting the data collected and performing additional examinations on selected specimens from the oxidized samples. During FY 1992, the data report will be published and an additional report will be prepared presenting the results of detailed examinations on rod segments and selected bare fuel fragment specimens, and work will continue on evaluating the data relative to recommendation of storage limits. Finally, during FY 1993 the specimen examination and storage limits recommendation reports will be published.

Another technology milestone was reached at the end of 1990 when the testing phase of the cooperative concrete cask demonstration program involving DOE, EPRI, Wisconsin Electric, and Pacific Sierra Nuclear was completed. The test of the cask, holding 17 canisters of consolidated fuel, involved validation of the cask thermal and shielding capabilities. The cask successfully met all design objectives. The project also served to demonstrate the fabrication economics and the licensability of the cask design. Work is now in progress to reduce and analyze all test data and to prepare a report that will document the project results. The report will be available by the end of 1991.

Other R & D activities underway in the U.S. include the following:

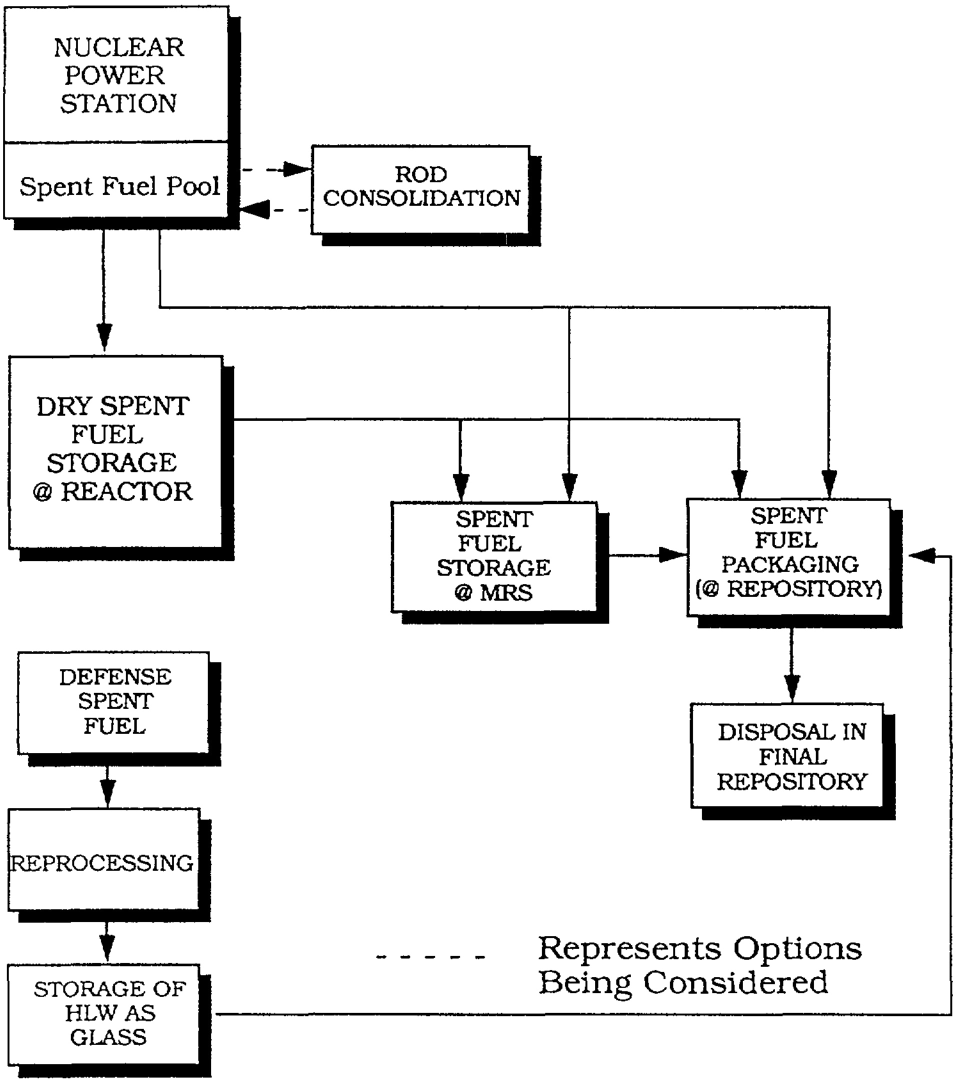
- Compaction of Hardware Scrap From Rod Consolidation
  This activity at the Millstone 2 site is designed to demonstrate that technology is available to achieve relatively high scrap compaction ratios of about 8 to 10. Previous demonstration attempts in the US have not been successful.

- Small Cask to Large Cask Transfer System
  A preliminary design has been completed that indicates the technical and economic feasibility of transferring uncanistered fuel from a small cask to a large cask. Such technology would allow utilities with crane or geometry restrictions to still store fuel in large efficient systems such as concrete casks.

- BWR Rod Consolidation
  Plans are to pursue an aggressive program to develop designs for performing rod consolidation on BWR fuel elements. These designs would use, to a large extent, automated and robotic equipment.

- Licensing of a Dual Purpose Cask
  A cooperative program involving Nuclear Assurance Corporation and their Spanish partner ENRESA, along with Virginia Power and EPRI, are undertaking to obtain from NRC both a PT 71 transport license and a PT 72 storage license for their 26 element NAC-STC cask. This represents the first effort in the U.S. to license a cask for dual purpose use.

- NRC Approval of thermal Hydraulic Codes
  DOE has submitted complete packages to the NRC on the PNL developed COBA-SFS and HYDRA-II thermal hydraulic codes. The intent of the project is to receive NRC approval for use of the codes in licensing and safety analyses activities. The NRC review is expected to be completed by the end of FY 1991.



Spent fuel management in the United States of America (December 1990)

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