

IAEA-TECDOC-1343

Spent fuel performance assessment and research

*Final report of a Co-ordinated Research Project on
Spent Fuel Performance Assessment and Research (SPAR)
1997–2001*



INTERNATIONAL ATOMIC ENERGY AGENCY

IAEA

March 2003

The originating Section of this publication in the IAEA was:

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

SPENT FUEL PERFORMANCE ASSESSMENT AND RESEARCH
IAEA, VIENNA, 2003
IAEA-TECDOC-1343
ISBN 92-0-102703-6
ISSN 1011-4289

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Printed by the IAEA in Austria
March 2003

FOREWORD

As of the beginning of 2002, more than 150 000 Mg U spent fuel is currently being stored in various storage facilities. Most of this fuel is under water, but dry storage is becoming a widely used technology with more than 9000 Mg U currently stored in various countries. In the future spent fuel storage capacity needs will continue to increase and some of these fuels will have to be stored for 50 years or more, before their reprocessing or final disposal takes place. As a result, interim storage for long periods of time will continue to be a key technology for all Member States.

A number of countries are planning or have already initiated research programmes on spent fuel storage performance, or are developing a high confidence database which evaluates the storage of spent fuel for extremely long periods of time, to determine their consequence on disposal.

Spent fuel storage technology (particularly dry storage) is undergoing a rapid evolution, new fuel and material design changes are coming on stream and target burnups are steadily increasing. These improvements will ask for new studies and potential adaptations of the storage technologies currently used.

As a consequence, the IAEA initiated in 1997 a Co-ordinated Research Project on Spent Fuel Performance Assessment and Research (SPAR) to collect and exchange spent fuel storage experience of the participating countries, to build a comprehensive international database supporting the licensing of present and future technologies; to carry out research work which will evaluate and justify the storage of spent fuel for extremely long periods of time (more than 50 years); and to assist in defining how the requirements for spent fuel storage and for the whole back end of the fuel cycle are connected.

The present publication is based on results obtained in the participating countries. The draft manuscript was prepared and discussed during the last research co-ordination meeting, held in Cordoba, Spain, 1–5 October 2001, and finalized at a consultants meeting in February 2002. The report provides an overview of technical issues related to spent fuel wet and dry storage and summarizes the objectives and major findings of research, carried out within the framework of the CRP.

The IAEA gratefully acknowledges the contribution of the CRP participants and the consultants who participated in the drafting and review of the report. The IAEA staff member responsible for this publication was H.P. Dyck of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

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1. INTRODUCTION AND OBJECTIVES

Several options for the ultimate management of spent fuel discharged from nuclear power plants are being implemented or are under consideration. At the present time, the two leading options are:

- Direct disposal of the spent fuel in a geological repository (once-through cycle), and
- Reprocessing of the spent fuel, recycling of the reprocessed plutonium and uranium, and final disposal of the wastes from the reprocessing operations.

However, delays in opening geological repositories in most countries or in implementing reprocessing in some countries mean that increased spent fuel storage capacity in combination with longer storage durations will be needed over the foreseeable future. Worldwide, the spent fuel production rate, now at about 10 500 t HM, is expected to increase to ~11 500 t HM by 2010. Because less than a third of the fuel inventory is reprocessed, ~8000 t HM/year on average will need to be placed in interim storage facilities.

Presently, spent fuel is mostly stored wet in at-reactor (AR) or at away-from-reactor (AFR) facilities. Because many AR pools are approaching their full capacity, even after extensive re-racking, storage of spent fuel under dry and inert atmosphere is being used increasingly.

The situation is further complicated by today's reliance on higher enrichment, higher burnup fuels as well as on mixed-oxide fuel, to generate electricity at a competitive cost. Given the much higher decay heat levels from these fuels, wet storage will remain the preferred approach for interim storage during the first decade after discharge. After sufficient decay ("cooling"), and especially for long storage durations (up to 300 years under some scenarios), dry storage under inert conditions or in air become the preferred alternative, given the passive nature of dry storage systems.

This publication is the last report on the IAEA Spent Fuel Performance Assessment and Research (SPAR) co-ordinated research project (CRP) — a continuation of the previous BEFAST-I to -III. Eleven participants from Canada, France, Germany, Hungary, Japan, the Republic of Korea, the Russian Federation, Spain, the United Kingdom, and the United States of America participated in the programme. In addition, a participant from Sweden contributed to the programme as well.

The overall objectives of this CRP were:

1. To carry out research work to evaluate the technical basis for storing spent fuel for extended periods of time, i.e. more than 50 years.
2. To evaluate and exchange data on spent fuel storage research and experience among the participating countries to build a comprehensive, international database supporting the licensing of present and future technologies.
3. To assist in defining how the requirements for spent fuel storage and the whole fuel cycle back-end are connected.
4. To exchange operating experience in spent fuel storage.

The specific objective of the CRP was to identify materials issues in long-term storage facilities.

2. HISTORY OF THE BEFAST AND SPAR CO-ORDINATED RESEARCH PROJECTS

Extended spent fuel storage is, and will remain, 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 close the nuclear fuel cycle, but generic questions related to spent fuel storage are common to all nuclear programmes.

The first phase of the BEFAST project (1981–1986) involved 12 organizations from 11 countries: Austria, Canada, the former Czechoslovakia, Finland, the former Federal Republic of Germany (FRG), the former German Democratic Republic (GDR), Hungary, Japan, Sweden, the USA and the former USSR. A subsequent project, BEFAST-II, Behaviour of Spent Fuel and Storage Facility Components during Long-term Storage, implemented during the years 1986–1991, involved organizations from 12 countries: Argentina, Canada, FRG, Finland, GDR, Hungary, Italy, the Republic of Korea, Japan, the UK, the USA and the USSR. BEFAST-III, implemented during the years 1991–1996 involved 15 organizations from 12 countries: Canada, Finland, France, Germany, Hungary, Japan, the Republic of Korea, the Russian Federation, Slovakia, Spain, the UK and the USA. There was also an observer from Sweden.

During the three co-ordinated research projects (CRPs) the participating countries contributed their R&D results on fundamental questions of spent fuel storage. The reports of the CRPs have been published as IAEA-TECDOCs [1–3].

Towards the end of the BEFAST-III project, it became apparent that the R&D component of the project was decreasing steadily; more emphasis was being placed on the operation and implementation of storage technology. The storage technology (particularly dry storage) was undergoing a rapid evolution: new fuel and material design changes were coming on stream and target burnup were steadily increasing. With the increased burnup came higher fission gas and fission product inventories, increased sheath (cladding) strains and increased cladding hydriding and oxidation. Because of all the new parameters that have surfaced during the course of BEFAST-III, a new project SPAR was proposed to address the effects of these new parameters on long-term storage and to determine their consequences on disposal.

The first phase of the IAEA Spent Fuel Performance Assessment and Research (SPAR) CRP (1997–2002) involved 11 organisations from 10 countries: Canada (2), France, Germany, Hungary, Japan, Republic of Korea, Russian Federation, Spain, Sweden, UK, and USA. Sweden participated in the project as an observer.

Three Research Co-ordination Meetings were held during the course of the SPAR CRP: the first in April 1998 in Washington, D.C.; the second in May 2000 in Ahaus, Germany and the third in October 2001 in Cordoba, Spain.

Major topics for both wet and dry storage during all four CRPs are summarised in Table I. It records detailed objectives and the shift in emphasis during the various phases of the Programmes between 1981 and 2002.

Table I. Research Subjects in the CRPs

Long-term Behaviour	B-I	B-II	B-III	SPAR	Surveillance	B-I	B-II	B-III	SPAR	Facilities & Operation	B-I	B-II	B-III	SPAR
Material aspect (cladding & components)	X			X	Monitoring; Wet + Dry -Environment - Components - Fuel assemblies - Workers' dose rate			X	X	Capacity enhancement - High density racks - Re-racking - Double tiering - Doped coolant - Rod consolidation		X		
Degradation mechanisms and models	X	X		X	Fuel conditions - Operational - Fabrication - Technology - Defected fuel rods and assemblies		X	X	X	Changing modes Wet–Dry		X	X	X
Validation - experimental - experience		X	X	X	Different reactor types		X	X	X	Handling of heavily damaged fuel System performance		X	X	X X

3. WET STORAGE

Vast quantities of zirconium-alloy-clad fuel now reside in pool storage (145 000 t HM). As of January 1, 2001 by far the largest quantities of spent fuel in wet storage reside in the USA (~42 500 t HM) and Canada (29 600 t HM). The Russian Federation has about 9200 t HM of RBMK-1000 fuel and 3800 t HM of WWER-1000 fuel in storage pools (both reactor types use Zr1Nb-clad fuel). Apart from Canada and Sweden (CLAB), the largest AFR wet storage facilities are generally associated with reprocessing sites: La Hague, Krasnoyarsk, Rokkasho Mura, and Sellafield.

The experience in wet storing spent nuclear fuel now spans some 40+ years. Wet storage continues to dominate as the primary method for storing spent nuclear fuel; ~93% of all spent fuel is wet stored. The benefits provided by this technology are mainly associated with cooling efficiency and shielding. Also, it facilitates safeguards and one-off fuel inspection/examination exercises. Despite the emergence of dry storage technologies, new and extensions to existing wet storage pools have been initiated or completed in France, Germany, Japan, Russian Federation, Sweden, UK and USA.

Reactors have been burning MOX-fuel since the early 1970s. Currently its usage is exclusive to Europe where there are 91 reactors with part-core MOX loading. Spent MOX behaviour in wet storage is similar to UO₂ fuel apart from the higher end-of-life radiation levels and associated heat of decay. For these reasons significantly longer wet storage would be required to achieve the comparable radiation & decay heat levels of UO₂ spent fuel prior to further processing (interim dry storage, reprocessing or direct disposal).

3.1. SPENT FUEL STORAGE EXPERIENCE

3.1.1. General performance

For primary barrier or containment purposes, cladding corrosion is the factor of most interest in wet storage. However, retention of fuel assembly structure integrity is the more important factor when retrieval is taken into consideration. A detailed review of the degradation mechanisms of various fuel types under wet storage conditions is given in Section 7.

For *zirconium alloy clad* fuel, data exists for continuous pool storage of greater than 40 years. This data indicates cladding corrosion to be extremely low (1×10^{-6} $\mu\text{m/a}$) and, therefore, is not viewed to be the time-limiting factor for prolonged wet spent fuel storage; even in poor pool chemistry conditions.

For *stainless steel clad* fuels, continuous storage experience of 27 years (LWR) and 23 years (AGR) exists. Although the general cladding corrosion rates for these fuels is significantly higher than zirconium alloy fuel (at ~0.1 $\mu\text{m/a}$), general cladding corrosion is not a time limiting factor for the storage durations currently envisaged (up to 100 years). For AGR fuel, particular attention to pool water chemistry is required as parts of the fuel stringer become sensitised during reactor operation; see Section 7 and Section 3.2 for further information.

Magnesium alloy clad fuel is particularly susceptible to cladding corrosion under wet storage conditions. Although a protective magnesium hydroxide film is initially formed, the presence of any aggressive ions in the water promotes the dissolution of the protective oxide film and leaves the cladding open to pitting attack. For this reason, Magnox fuel is stored in dosed pool water and storage duration tends to be limited; normally <5 years.

In general the wet storage of spent fuel only appears to be limited by adverse pool chemistry conditions or deterioration of the fuel storage pool structure. Other factors that may influence early fuel removal from NPP pools are removal of long stored fuel to maintain pool capacity, or complete closure of the pool on economic grounds at the time of reactor shutdown. In these cases, one of several options may be considered: interim dry storage, transportation to a centralised pool storage facility, reprocessing of the fuel, or direct disposal (if a route becomes available).

3.1.2. Wet storage experience in the participating countries

The following paragraphs provide examples of the application of wet storage in the countries participating in the CRP.

Canada

Canada has been storing fuel under water since the early 1960s. Fuel has been stored in water pools at CANDU reactor sites for about 40 years.

In 1977/78, AECL and Ontario Hydro initiated an experiment at AECL's Chalk River Laboratories (CRL) to assess the feasibility of storing irradiated CANDU fuel with zirconium alloy sheath underwater for at least 50 years. One hundred and seventy-six elements from 19 fuel bundles were originally selected for the program. They included prototype generating station fuel irradiated in light water loops in an experimental reactor at CRL and production fuel irradiated in NPD, Douglas Point and Pickering. The oldest bundle selected for the program was discharged from the NPD reactor in May 1962. All 19 bundles were dismantled and the fuel stored as individual elements in open-ended stainless steel tubes. Groups of eight tubes were loaded into either aluminium or stainless steel storage cans. Forty elements were stored in stainless-steel cans in a water pool at NPD; the rest were stored in the NRX water pool at CRL. Stainless steel and aluminium storage cans were selected for the program to investigate differences in storage material behaviour. The NPD pool was selected as one of the storage sites, since its water chemistry and fuel handling techniques were typical of those used in CANDU power stations.

The fuel stored at NPD in the wet storage program was transferred to the NRX pool at CRL in November 1987, when NPD was shut down. Subsequently, in 1994, a decision was made to shut down the NRX pool and as a consequence the wet storage program fuel was transferred to the NRU pool at CRL during February 1995.

The fuel originally selected for the program was characterised in 1977/78, which consisted of a destructive examination of specific fuel elements from each of the bundles plus a non-destructive examination on the remaining elements. The examination revealed no changes in the condition of the fuel or sheath from the original post-irradiation examinations following up to 16 years of storage under water.

In 1988, specific fuel elements from each of the bundles were retrieved for a detailed interim storage examination. That examination revealed no apparent change in the condition of the fuel or sheath in any of the fuel elements without cladding failures following up to 27 years of storage under water. X-ray photoelectron spectroscopy (XPS) examination of fuel fragments from an intentionally defected element that was stored for 21 years under water indicated that the surfaces of the fuel fragments were fully hydrated and significantly oxidised. However, the oxidation was restricted to a thin layer (i.e. nanometers) on the surfaces of the fuel

fragments as indicated by XPS and X-ray diffraction. An interim examination planned for 1998 was not carried out, partially because no observable changes are expected to occur over 10 years. The fuel remains in storage and future examinations will be scheduled as appropriate.

Eastern European Countries (Hungary, Russian Federation)

Apart from one experimental dry storage facility, all spent fuel in the Russian Federation is stored wet either at AR sites (Leningrad, Kursk, Smolensk and Novo-Voronezh NPPs), or at the AFR Mining Chemical Combine (Krasnoyarsk). The behaviour of the Zr1Nb clad WWER and RBMK fuel in storage has been good apart from corrosion in the RBMK spacer grids. Over ~40 years experience of wet stored fuel exists, there have been no known fuel cladding failures as a result of wet storage; under normal operating conditions.

Corrosion studies have been conducted on Zr1Nb alloy cladding specimen being cut from RBMK and WWER spent fuel assemblies of different burnup and cooling times. Results showed that burnup has a much more significant influence on the cladding plasticity than storage duration.

For RBMK fuel, a maximum oxide layer growth rate of 3–5 $\mu\text{m}/\text{year}$ has been observed in the spacer contact area of the spent fuel stored in open canisters with uncontrolled chemistry for 15 years of storage period.

For WWER-440 fuel, the measured oxide film thickness is distributed uniformly along the total cladding; its average value was 0.01 mm. In the location of the spacer grids, a local increase in the oxide film thickness was observed, with an additional value of 0.005 mm [4].

France

In France, because the back-end policy is based on prompt reprocessing, at-reactor spent fuel storage is limited to a short period. After two years maximum of cooling in the NPP pool, spent fuel is transported to the La Hague storage complex prior to reprocessing. The La Hague spent fuel reception, unloading and storage complex consists of the NPH wet cask unloading facility, the NPH storage pool, the T0 dry cask unloading facility and the C, D, E storage pools. The four stainless-steel-lined storage pools are interconnected and offer a total nominal capacity of more than 14 400 t HM.

Spent fuel originating from PWR and BWR NPPs of national and international customers is stored in baskets. This alternative to the usual rack system has been preferred because large quantities of fuel are handled frequently and baskets allow for easier reshuffling. Compact baskets have been designed to handle 9 PWR or 16 BWR fuel assemblies at a time. A storage density of 3 t HM/m², including all the basket transfer areas for pool operation, is achieved. Damaged fuel identified during unloading is placed into a sealed bottle prior to storage.

Germany

In Germany, a little less than 2900 t HM spent fuel is presently stored in at-reactor (AR) pools. This number might change a little with time. But it is foreseen that spent fuel will be stored in dry storage casks at the reactor sites (except Obrigheim, where only wet storage is planned), or at the central storage sites in Ahaus, Gorleben or Greifswald. In the future, the amount of spent fuel in AR wet storage will decrease, and finally all fuel will be stored dry.

This means that in Germany the amount of spent fuel stored wet will be limited. Nevertheless, Germany performed a detailed R&D work to assure that spent fuel can be stored wet without problems. The results of the work performed can be summarised shortly as follows.

Assessment of the ability of fuel assemblies to be safely held in wet storage — in particular, for long periods — has been based above all on the evaluation of corrosion mechanisms:

- Oxidation and electrochemical corrosion of cladding materials,
- Corrosive attack of structural components, and
- So-called crevice corrosion.

Oxidation is of no importance. Electrochemical attack can be suppressed through the selection of appropriate materials and by adequate control of the pool water chemistry. Corrosive attack on the components is negligible for several reasons: adequate material selection, in-pile passivation, and low temperatures. Crevice corrosion is prevented due to clean water conditions and controlled water chemistry.

All defect mechanisms associated with stress and strain in the Zircaloy claddings can be neglected, since these stresses are much less than the yield strength. A set of spent fuel rods containing incipient flaws as well as through-wall defects penetrating the cladding tube have been stored in a demonstration programme and periodically inspected for 18 years. The inspections have not revealed any unanticipated effects or changes. The demonstration programme provided strong evidence that wet storage of spent Zircaloy-clad fuel in water with adequate water chemistry condition and cleanliness is safe and reliable.

Japan

There are sixteen LWR nuclear power station sites in Japan, and one under construction. All sites have spent fuel storage pools. Approximately half of the cumulative amount of spent LWR fuel discharged from reactor has been shipped to domestic (JNC) and overseas reprocessing plants, and the remainder is stored at reactor sites. Most of these have been stored in pools, without problem.

The total controlled-capacity available for spent fuel storage at all LWR NPP sites is about 14 380 t HM. Due to the controlled capacity being approached at a number of NPPs, and to meet future demands, re-racking (AR) or modification to common-use for two or more reactors has been undertaken. Furthermore, a new pool has been constructed and common use of pools is planned.

Republic of Korea

The integrity of spent fuel rods in AR pool storage has been demonstrated by routine monitoring for over 20 years. No significant problems due to the spent fuel degradation in the pool have been encountered so far. There are no symptoms of fuel cladding or storage component corrosion. However, a top nozzle was separated from the PWR spent fuel assembly during a lifting operation at the Kori-1 spent fuel storage pool. The separation of the top nozzle occurred due to a fracture at the bulge joint area of the sleeve. It was revealed by detailed hot-cell examination that the sleeve failed due to inter-granular stress corrosion cracking. This fuel assembly was handled with a special tool. Recently, the Korea Atomic Energy Research Institute (KAERI) has developed a device for the in-pool reconstituting of sleeve-failed PWR spent fuel assemblies.

Spain

Spent fuel has been wet stored in Spain since 1970. All fuel storage is AR, and storage experience has been good, apart from an occasional top nozzle separation of early manufactured Westinghouse PWR fuel assemblies (see US contribution later). All fuel assemblies in storage have been moved at least once, when the AR pools were re-racked, without any problems occurring.

Sweden

In Sweden, the back end policy is based on final disposal. At the reactor, the spent fuel storage is limited to a short period. Cooling periods of nine months for BWRs, and 12 months for PWRs are required, prior to transport to the centralised storage (CLAB). The transportation of spent fuel is carried out in dry transport casks. The casks are cooled and filled with water before unloading in the unloading section of the CLAB facility. It is expected that CLAB will have to be operational until at least 2050.

The fuel assemblies are transferred from the casks into canisters that contain 25 BWR or 9 PWR FA respectively. When the canister is filled, it is transferred to the underground storage pools.

The current storage capacity is 5000 t HM; however, construction of a second storage cavern with a capacity of 3000 t HM started at the end of 1998 and is due for completion in 2004. As of middle of 2001, the storage records are as follows:

- 3410 t HM are stored,
- Spent fuel maximum burnup is 52 GW·d/t HM,
- MOX fuel is stored,
- Average collective dose is about 100 man·mSv/year,
- Release of radioactive nuclides to the environment is negligible,
- 16 years of successful operation.

United Kingdom

The wet storage of fuel discharged from Magnox and AGR NPPs can be conveniently divided into two phases. The first covers short-term storage at the reactor and the second covers the further interim storage away from reactor at Sellafield, prior to reprocessing. The only exceptions to the above are British Energy's Sizewell B PWR reactor, where all spent fuel is currently stored at reactor wet, and Magnox Generations Wylfa NPP (see Section 4).

The experience of wet storing commercial spent nuclear fuel on the Sellafield site spans some 45 years. There are four operational AFR storage pools at Sellafield: Fuel Handling Plant (Magnox/AGR), AGR Storage Pond (AGR), LWR Storage Pond (LWR/PIE), and Thorp Receipt & Storage (LWR/AGR). These pools store fuel from the first- and second-generation UK reactors, Magnox and AGR, and also irradiated fuel from LWR reactors, mostly from BNFL's overseas customers.

The main differences between fuel storage at Sellafield and other facilities are:

- Fuel storage is based upon containerised systems (Magnox and AGR containers, Multi-element bottles (LWR)). These facilitate:
 - Isolation of the fuel from the bulk pool water,
 - The chemistry within the container can be different from the bulk pool water,
 - The spread of corrosion products to the storage facility is minimised.
- A variety of pool chemistries are utilised depending on the properties of the fuel being stored.
- Apart from a few special transports, there are no individual fuel element handling operations, during storage.
- The pools are purged with fresh water; resultant effluent is either treated in the Site Ion Exchange Plant, or filtered and sentenced prior to sea discharge.

Fuel integrity/performance under extended storage periods has been monitored by the use of CCTV, activity release models, and full PIE programmes.

United States of America

Spent fuel storage experience from the civil reactor programme spans some 40 years. Apart from the General Electric Morris AFR at Illinois, all wet storage is AR in the USA. In general, the performance of wet storage systems has been excellent with only an occasional operational problem or mishap.

One such event occurred March 2001, when a top nozzle separated from a fuel bundle at the North Anna fuel pool of Dominion Virginia Power. The bundle was being returned to its storage location at the time when it fell some 11 feet back into its designated pool storage slot. The bundle was a Westinghouse 17×17 fuel assembly that had been removed from the core in 1984. The separation occurred at the bulge joint where the top nozzle thimble sleeves connect to the guide thimble tubes. Other similar failures have occurred at Prairie Island (US) and in France, Republic of Korea, and Spain. Stress corrosion cracking has been determined to be the failure mechanism in previous events.

3.2. POOL WATER CHEMISTRY

The role of water in spent pool storage is to:

- Facilitate heat removal from the spent fuel,
- Act as a biological shield,
- Maintain fuel cladding integrity,
- Facilitate spent fuel visual inspection.

To achieve these goals, water quality has to be optimised for the fuel being stored and for the pool components/systems. Additionally, consideration needs to be given to the exclusion of microbiological species (such as algae growth) to prevent loss of water clarity, etc. Factors that may be affected include water chemistry, water treatment, quality control methods and water temperature.

Operating experience gained since the 1950's shows:

- The ingress of aggressive ions such as chloride should be minimised,
- Where pool water quality has been maintained, spent fuel storage performance for all types of fuel has been excellent,
- Optimum pool water chemistry is linked to the fuel being stored,
- Apart from some experience with water-filled, canned RMBK fuel, the general experience with defected fuel is that the defect does not propagate during prolonged storage.

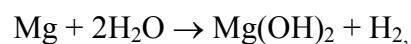
For stainless steel clad **AGR fuels**, pool water chemistry has to be specified to prevent corrosion of elements 1–5 of the original irradiation stringer through irradiation-induced intergranular stress corrosion cracking. To inhibit this failure mechanism, AGR fuel is stored in pool water dosed with sodium hydroxide (AFR) and boric acid/sodium hydroxide (AR).

All AGR stations use cooling pools, where spent fuels are stored in borated water. The water chemistry at AGR stations is dictated by criticality control. Boron is added in the form of boric acid. Sodium hydroxide is also added to return the pH of pool water to 7.

AFR pools contain non-borated water and are dosed with sodium hydroxide to maintain a nominal pH of 11.4 (target range: 11.35 to 11.45). Chloride levels of <0.1 ppm are achieved in enclosed pools and 1–2 ppm in open pools. The required chemistry for AGR fuels in the Thorp pools, which do not use sodium hydroxide as corrosion inhibitor, is established when the fuel is placed in containers for storage in the pool (pH ~7, Cl⁻ <0.1 ppm). For experimental purposes, individual containers can be sealed and monitored for activity release. This method was used to demonstrate that the AGR fuel remained intact during storage in the high quality water in Thorp storage pool even though a corrosion inhibitor was not being used.

For **CANDU fuels**, which use Zircaloy for cladding material, water chemistry specification is based on a long experience gained over the last 40 years in Canada and 12 years in the Republic of Korea. Spent fuel is stored in AR pools in demineralised water with typical specification given in Table II. Basically, water pH is maintained around 7 while chloride content is limited to a level less than 0.1 mg/kg. (0.2 mg/kg in the Republic of Korea; pH 5.5 to 9.0; and Cl⁻ <0.5 mg/kg at Ontario Hydro, Canada).

The storage of **MAGNOX fuels** requires particular attention in controlling pool water chemistry. Although magnesium alloy cladding dissolves in pure water by the following reaction:



a passive/protective magnesium hydroxide film is formed. This protective film, however, is also susceptible to dissolution from acidic species dissolved in the water:

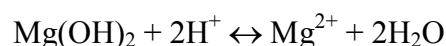


Table II. Typical examples of water chemistry parameters in storage pools

FUEL TYPE	CANDU		AGR	MAGNOX		LWR	
Pool type	At Reactor pool	At Reactor pool	AFR pool ¹	At Reactor pool	AFR pool ¹	CLAB pools Maximum	CLAB pools Normal
Storage structure			Vented containers			Ullaged Container	
Water	Demineralised water	Borated water	Demineralised water			Demineralised water	Demineralised water
pH	6.9	7	11.4	11.5	13.0 ²	8.6	5-6
Conductivity	200 µS/m	-	-	200 µS/m	-	< 500 µS/m	< 100 µS/m
Chloride	0.005 mg/kg (ppm)	0.5 ppm	< 0.1 to 1–2 ppm	< 1 ppm	0.1 ppm	< 500 ppb	< 5 ppb
Sulphate	0.02 ppm	0.5 ppm	0.1 ppm	< 0,5 ppm	0.1 ppm	< 500 ppb	< 5 ppb
Fluoride	0.06 ppm	-	-	-	-	< 500 ppb	< 5 ppb
Na	0.1 ppm	-	-	-	-	< 10 ppb ³	< 1 ppb ²
⁹⁹ Mo	8.6 × 10 ⁶ Bq/m ³	-	-	-	-	-	-
¹³⁷ Cs	0.3 × 10 ³ Bq/m ³	10 × 10 ⁶ Bq/m ³	1 × 10 ⁷ Bq/m ³ ⁴	60 × 10 ⁶ Bq/m ³	20 × 10 ⁶ Bq/m ³	-	2 × 10 ⁴ Bq/m ³
Oil				-	-		
Water activity	9 × 10 ⁶ Bq/m ³	40 × 10 ⁶ Bq/m ³	10.5 × 10 ⁶ Bq/m ³	-	20 × 10 ⁶ Bq/m ³		3 × 10 ⁶ Bq/m ³

¹ Purged pools, no in-pool ion exchange systems. Ion exchange if required takes place either on the pool water discharge filter or downstream of the storage pool in a separate facility, prior to sea discharge.

² pH inside ullaged containers, not pool.

³ Na⁺ and Ca²⁺

⁴ Pool contains small quantities of fuel with perforated cladding.

	RBMK		PWR and BWR			WWER	
Pool type	AR pool	AFR pool	AR pool	AFR pool	AFR pool (UK) ⁵	AR pool	AFR pool
Storage structure		Water filled cans			Container (MEB) ⁶		
Water	Demineralised water	Demineralised water	Borated water (PWR) Demineralised water (BWR)	Demineralised water	Demineralised water ⁷	Borated water	Demineralised water
PH	5.5–8	6–7.5	4.5–5.5	4.5–5.5	6.2–7.5	4.3–6.5	5.5–7
Conductivity	300 µS/m	300 µS/m		350 S/m	-	200 µS/m	200 µS/m
Chloride	0.1 mg/kg	0.05–0.1 ppm	0.15 ppm	< 0.1 ppm	0.1 ppm	< 0.1 ppm	< 0.1 ppm
Sulphate	-		0.15 ppm	< 0.1 ppm	0.1 ppm	< 0.150 ppm	
Fluoride	0.1 mg/kg		0.15 ppm (Republic of Korea)	< 0.1 ppm	-		
Na ⁺ and Ca ²⁺	-			< 0.5 ppm	-		
⁹⁹ Mo	-						
¹³⁷ Cs	-				1 × 10 ⁶ Bq/m ³		
Oil	< 200 ppm						
Water activity	< 5 × 10 ⁶ Bq/m ³	< 5 × 10 ⁶ Bq/m ³	< 2 × 10 ⁷ Bq/m ³ (Republic of Korea)	8 × 10 ⁶ Bq/m ³	2–3 × 10 ⁶ Bq/m ³	40 × 10 ⁶ Bq/m ³	40 × 10 ⁶ Bq/m ³

⁵ Purged pools, no in-pool ion exchange systems. Ion exchange if required takes place either on the pool water discharge filter or downstream of the storage pool in a separate facility, prior to sea discharge.

⁶ Pool also contains AGR fuel in containers.

⁷ MEBs can contain either deionized or borated water.

It is therefore important to minimise ions that would lead to cladding degradation; i.e. chloride, carbonate, and sulphate. To overcome these effects, Magnox fuel is stored in high purity caustic-dosed water to a minimum pH of 11.4. At AFR, these conditions are increased to pH 13.0 inside the storage containers. Fuel has been satisfactorily stored for up to 5 years at $\text{pH} \geq 11.4$ by dosing with sodium hydroxide and keeping $[\text{Cl}^- + \text{SO}_4^{2-}] < 1 \text{ mg/kg}$.

Light water reactor fuels (PWR or BWR)

Apart from a very small amount (three NPPs) of older stainless steel clad fuel, all LWR's use Zircaloy cladding. Spent fuel is stored in open structures (such as racks or baskets used worldwide), or in sealed containers (such as the Multi Element Bottle used in UK). Water purity is controlled to suppress conditions that might lead to a corrosive environment for spent fuel cladding and structures.

The AR pools for PWR reactors are filled with borated water, while demineralised water is used for BWR reactors as well as for AFR wet storage installations serving both LWR reactor types. Depending on the countries that operate Light Water Reactors, water chemistry for storage installations may be quite different. However, some general parameters can be defined: pH requirement is 4.5–5.5 and chloride and fluoride concentrations have to be limited to 0.1–0.15 ppm (see Table II). In Japan, the pH requirement for BWR stations is 5.3–7.5; chloride and fluoride concentrations are limited to 0.5 ppm, while each operating value is generally maintained below 0.05 ppm.

For some storage pools that use Boraflex neutron poison material, the increasing concentration of silica may become a concern. The silica results from the degradation of the silicon rubber polymer and the release of other silica that is present as a filler material.

For AR **RBMK** storage, elements are suspended from the pool floor beams. The fuel has zirconium-1% niobium alloy cladding (Zr1Nb), and the required water chemistry quality has been defined to prevent the corrosion of fuel materials.

In AFR storage installation, RBMK fuels are stored in water-filled cans. Water chemistry analysis in the cans shows it changes with storage time. The simultaneous effects of the absence of water mixing, filtration, and ion exchange lead to an increase in water corrosivity, radiolysis products, and halogen contents. As a result, corrosion becomes more intensive and diversified. The uniform corrosion rate in wet storage, however, is consistent with other zirconium alloy clad fuel. For the major portion of the FA, the corrosion film thickness on the cladding surface was 10–40 μm . However, some FAs developed significant nodules and one FA demonstrated fretting corrosion in the vicinity of the spacer grids. Stainless steel (X18H10T) shows a tendency to pitting corrosion. Alkali additives can be used for suppressing water corrosivity. Spacer grids have now been replaced by zirconium alloy components to overcome the problem.

WWER fuel with Zr1Nb alloy cladding is stored in borated water in the AR pools. Boric acid concentration is maintained within 12–16 g/kg; concentrations of halides (Cl^- and F^-) are kept below 0.1 $\mu\text{g/kg}$; pH varies in the 4.3–6.5 range. AFR pools are filled with demineralised water. Halide concentrations are kept below 0.15 mg/kg; pH values are within the 5.5–7.0 range.

3.2.1. Water quality control

The objective of water quality control is to maintain the conditions that (1) minimize corrosion of spent fuel, pool components, radioisotope build-up and inhibit biological growth in the storage water and (2) guarantee the clarity of pool storage water.

To maintain water quality, the following analyses are performed on a routine basis:

- PH,
- Conductivity,
- Water turbidity,
- Chemical composition (Cl^- , F^- , SO_4^{2-}),
- Isotope-specific activities.

Generally, a complete water activity content is checked weekly, this is increased to a daily analysis in Sellafield's case (UK) where pool water is discharged to sea after sentencing & treatment. The complete chemical analysis of the water is generally performed on a monthly basis. At AFRs (France and UK), specific parameters such as chloride, fluoride and sulphate levels are measured either weekly or daily.

Table III provides examples of the common parameters measured, techniques applied and frequency of analysis.

Table III. Water quality parameters

	FREQUENCY	ANALYSIS METHOD
pH	Daily	pH electrode
Conductivity	Daily	Conductivity meter
Turbidity	Daily	UV-Visible Spectrometer
Cl^- , F^- , SO_4^{2-}	Monthly (AR)	Ion chromatography
NO_3^- , PO_3^{4-}	Weekly/daily (AFR)	Inductive coupled plasma mass spec. (ICPMS)
Alpha, Beta, gamma general control	Weekly (AR)	Scintillation counters
	Daily (AFR)	Ion chamber
Complete chemical analysis	Monthly (AR/AFR)	ICPMS
	On request	Atomic Absorption or Emission Spectroscopy (AAS or AES)
		Plasma AES

Where water quality is not strictly controlled or has been allowed to deteriorate, a number of reported incidents have resulted. These include:

- Top nozzle separations, etc.
- Cladding failures, etc.

3.2.2. Sources of pool water activity

The main sources of pool water activity are from the leaching of corrosion products (commonly known as „crud“) adhered to the fuel, and fuel or fission product species diffusing through any cladding defects and through micro pores within the cladding. Water activity has to be maintained at a reasonably low level to limit dose to the operators and to minimise adsorption by in-pool structural components and ancillary equipment, as a means of facilitating final decommissioning activities. An example at full pool water activity analysis can be seen in Table IV.

Table IV. Example of full pool water activity analysis

	Activity, Bq/ml		Activity, Bq/ml
^{110m} Ag	0.040 698	⁹⁵ Nb	0.105 751
¹⁴ C	0.034 763	⁶³ Ni	2.187 500
¹⁴⁴ Ce	0.235 838	¹⁴⁷ Pm	0.004 601
⁶⁰ Co	5.638 653	²⁴¹ Pu	0.420 042
¹³⁴ Cs	0.089 831	¹⁰³ Ru	0.085 452
¹³⁷ Cs	3.937 652	¹⁰⁶ Ru	0.501 148
¹⁵² Eu	0.018 595	³⁵ S	0.031 119
¹⁵⁴ Eu	0.007 337	¹²⁵ Sb	0.228 827
¹⁵⁵ Eu	0.007 929	⁸⁹ Sr	0.099 563
⁵⁵ Fe	0.007 391	⁹⁰ Sr	0.286 458
³ H	0.451 346	⁹⁹ Tc	0.085 551
¹²⁹ I	0.019 285	⁶⁵ Zn	0.052 692
⁵⁴ Mn	0.025 567	⁹⁵ Zr	0.156 465

Denotes all results <= Limit of Detection.

Fuel crud

Fuel crud is the common name given to the residues deposited on fuel assemblies as a result of reactions of the coolant with the primary coolant circuit components. It is generally observed that fuel from gas cooled reactors has minor deposits, and PWRs are cleaner than BWRs. This is by virtue of the differences in materials used in construction and the chemistry of the cooling circuits.

In Sweden, crud removed from BWR fuel originating from the Ringhals-1 NPP has been analysed since 1978. This experience shows that the amount of deposited material is more or less linear with the residence time of the fuel in the reactor. The composition of the crud is feedwater water chemistry dependent.

Radionuclide content of crud

The radionuclide content of fuel crud mainly comprises of cooling circuit corrosion products that have been neutron-activated. In some cases fission products may be present in trace amounts where in-reactor cladding defects have been present.

Upon fuel discharge from reactor, typically the principal nuclides present are ^{54}Mn , ^{55}Fe , ^{58}Co and ^{60}Co , with ^{55}Fe being the dominant species. Due to the longer half-lives of ^{55}Fe and ^{60}Co , only these remain above background levels in 15-year cooled fuel. In the long-term (>100 years), the main activation species remaining is ^{63}Ni , which is present only in trace amount at reactor discharge. An analysis of Ringhal-1 crud is given in Table V.

Table V. Crud composition taken from the fuel cladding at Ringhals-1 NPP in 2000. Average of four (4) fuel assemblies with residence time from 2 to 5 years and burnup from 18.9 to 37.7 MW·d/kg U

Corrosion product	g/m ²
Co	1.3 E-2
Cr	1.6 E-1
Cu	1.8 E-2
Fe	5.3
Mn	8.3 E-2
Ni	7.6 E-1
Sb	2.3 E-3
Zn	1.9 E-1
Nuclide	GBq/m ²
^{58}Co	28.7
^{60}Co	52.6
^{51}Cr	15.8
^{59}Fe	8.0
^{54}Mn	13
^{124}Sb	2.8
^{65}Zn	0.6

3.3. INACTIVE CONTENT

BWR

Studies indicate BWR crud to be hematite in origin. Typical metal content is ~87 wt% Fe, in the form of red hematite — ~2 wt% Cu, ~4.4 wt% Zn, ~3.3 wt% Ni, and ~2.2 wt% Mn.

PWR

The major components of PWR crud are reported to be spinels of the nickel-substituted ferrite ($\text{Ni}_x\text{Fe}_{3-x}\text{O}_4$ with $0 \ll x \leq 1$) and magnetite (Fe_3O_4) types. Compounds of NiO, SiO₂, and Cr₂O₃, plus elemental Ni have also been identified. Typical metal content is about 78 wt% Fe, 20 wt% Ni, and 2 wt% other metals (mainly Cr, with Co at 0.03–0.11 wt%).

Gas cooled reactors (Magnox/AGR)

Fuel arisings from gas cooled reactors mainly suffer from carbaceous crud deposits from the carbon dioxide coolant reactions with inhibitors/graphite decomposition.

3.3.1. Sources of pool water particulates

The main sources of suspended solids are from particulates associated with the flask if pool discharged, from crud dislodged from individual fuel assemblies, transfers to storage, movement of the fuel, or open fuel baskets in storage. Flask-associated particulates include crud released from the fuel (thermal shock) when water is introduced to the flask internals if shipped dry, dirt on the flask externals, and entrained particulates from the flask liner.

In the case of open storage bays, airborne particulates are also introduced.

3.3.2. Water treatment

Water treatment is necessary to:

- Remove suspended solids in order to maintain pool water clarity,
- Remove dissolved radioactive species to minimise the dose to operators,
- Limit build-up of nutrients (phosphates and nitrates) to minimise biological growth,
- Limit build-up of aggressive ions (chlorides, sulphates, etc.) that can initiate corrosion of the fuel assemblies stored, or in-pool components.

Water treatment normally includes a mechanical treatment to remove the solid materials contained in the pool, in conjunction with a chemical treatment to extract both radioactive and non-radioactive chemical species dissolved in the pool water.

Mechanical treatment of the bulk pool water is generally performed by filters (pre-coated sand or mechanical), while chemical treatment is realised with ion exchangers (cationic and anionic resin types are used). In some cases, ion exchange is preceded by neutralisation. Generally only single bed organic ion exchange resins are regenerated when saturated. The resultant concentrate may include a boron recycling step, followed typically by evaporation and final encapsulation for disposal. Mixed bed filters and inorganic exchangers resins are also used. In these cases, there is no regeneration phase and the saturated beds are disposed of directly after encapsulation.

The build-up of particulates on the pool floor and walls is removed mechanically, by in-pool cleaners. A variety of designs have been deployed from modified leisure pool cleaners, simple suction devices to purpose-built two-stage cleaners, coarse and fine (cyclone) filters.

In the case of ion exchange there are three types of systems in operation. These include: out of pool ion exchange columns, ion exchange floated on top of a pre-coated mechanical filter and in-pool water treatment units; examples of the latter include cartridge systems and the combined ion exchange/cooling Nymphaea system (France).

Special attention is also required to avoid the growth of microbiological species that can reduce water clarity or even lead to microbial attack of storage materials. The main factors in limiting biological growth are to minimise the introduction of nutrients (especially phosphates), intensity of lighting in the storage area, and temperature.

Where bacterial growth does initiate, treatments have varied from the use of biocides to full-scale mechanical cleaning and collection of the biogrowth. Operating pools at high pH also prevents biological growth.

3.3.3. Water temperature

Water temperature requirements may vary with the specific plant design and with the type of storage installation (AR or AFR). Normal temperature limits are usually based on operating considerations, such as personnel occupation requirements and equipment operating limits rather than fuel corrosion. The temperature limits normally are about 45°C (max 65°C in the Republic of Korea). Other considerations that make it preferable to operate pools at the lowest practicable temperature include lowering the release rate of radionuclides from defective fuels, minimising bacterial or microbial growth, and lowering the humidity level in the storage area.

For AFR storage installations, spent-fuel cooling-time is greater, water temperature is lower and does not exceed 40°C in normal conditions. The influence of water temperature on the transfer of activity from the fuels to the storage water has been shown to double with each 10°C-rise in temperature. It is, therefore, advantageous to keep pool water temperature as low as reasonably possible, thereby delaying release, and taking advantage of the decay of the activity while it is fixed to the fuel surface. The load on the waste treatment systems becomes correspondingly lower [5].

3.4. POOL COMPONENTS AND MATERIALS

The choice of pool components and materials is dependent upon the type of fuel being stored, cost, and to facilitate final pool decommissioning. The latter property has come to the fore as experience in both in-pool performance and final pool decommissioning has been gained. This has led in most cases to the almost exclusive usage of stainless steel for spent fuel storage structures, heat exchangers, piping and fuel handling equipment.

3.4.1. Pool lining

There are two main methods of treating the internals of the reinforced concrete pool structure to make it watertight, that is to line with welded sheets of stainless steel or to coat with a water resistant paint system.

All LWR AR storage pools are stainless steel lined, while both stainless steel and epoxy liners are used in CANDU AR storage and epoxy/paint is used in Magnox pools. In the case of AFR storage installations all are stainless steel lined (France, Finland, Germany, Japan, Russian Federation, Sweden), with the exception of UK AFR storage pools; where most are only lined at the wind/water line, the rest is painted.

In case of concrete pools coated with epoxy (Canada), the concrete employed has been shown to have negligible corrosive ion leaching and permeability to water. However, when coated concrete is used, the cumulative dose rate on the epoxy has to be limited to prevent epoxy degradation. Measurable changes in epoxy liner properties have been observed after a 1 MGy radiation dose. With the extension of the operating life, this dose limit could be exceeded at some point after the design life of the pool is reached. Since 1988, some extent of radiation-

induced deterioration has been observed at the Pickering station, without any damage on the liner envelope. In the case when water temperature has been maintained higher than 32°C on a regular basis, a risk of minor damage to the coated concrete walls has been observed at the Pickering station in Canada. If epoxy liner radiation-induced deterioration continues, there is a possibility that water may eventually contact the structural concrete. Thus, a programme to investigate the long-term effect of water on concrete was considered.

To resolve the issue of epoxy degradation, alternative paint systems have been developed and deployed in the UK.

3.4.2. Storage racks

The designs/methods used for spent fuel pool storage can be fuel/facility type, and operator dependent. For example, RBMK (AFR) fuel is stored in stainless steel individual bottles suspended in a rack located at the pool surface, in comparison LWR (AR) fuel is typically stored in open top floor mounted borated stainless steel racks. In the case of AFR fuel interim storage, prior to reprocessing, the storage systems are designed to facilitate the bulk movement of fuel and as such tend to have a reduced capacity than can be achieved AR by the fixed high-density rack systems. In one country (UK), the storage systems mainly comprise of the „basket” used in transportation; i.e. the Magnox & AGR skips and the multi-element bottle (LWR).

The materials of construction are primarily either borated stainless steel, or stainless steel, usually in combination with a neutron absorbing material; for example Boral, Boraflex or Cadminox. One exception is the Magnox skip, painted mild steel, which is used due to concerns with respect to the potential for electro-coupling between the fuel cladding and stainless steel. The only other material that has found some use for structural components is aluminium; mainly for its corrosion resistance in deionised water when passivated.

4. DRY STORAGE

4.1. INTRODUCTION

Almost 20 years of favourable experience exists with the dry storage of spent power reactor fuel and about 30 years with research reactor fuel. Dry storage experience exists with fuel from a variety of reactor types (CANDU, HWR, PWR, BWR, WWER-440, WWER-1000, RBMK, MAGNOX and the HTGR). Since its conception, dry storage of spent fuel has evolved into a wide variety of systems. Examples of these are concrete canisters, steel-lined concrete containers and concrete CANSTOR modules (modular vault-like storage system) in Canada, concrete canisters in the Republic of Korea, vaults in France, Hungary, UK and the USA, and casks in Germany, Japan, Russian Federation (in development), Spain and the USA [1, 2, 3, 6]. At the present time, all countries participating in SPAR (Canada, France, Germany, Hungary, Japan, Republic of Korea, Russian Federation, Spain, United Kingdom and the USA) are engaged in various dry storage technologies. Almost all participating countries are also actively pursuing a dry storage research and development programme. So far, the results of the research indicate that fuel can be stored safely under the present conditions for many decades.

As regards licensing conditions for the dry storage facilities, different trends and licensing periods have been implemented throughout the SPAR countries as shown in the following table VI:

Table VI. Dry storage licensing conditions in the SPAR countries

Country	Initial licence period	Renewal period
Canada	2 years	2 years
Germany	40 years	N/A
Hungary	10 years	10 years
Japan	Not limited (similar to the reactor site licence)	N/A
Republic of Korea	Not limited (covered under the reactor site licence)	N/A
Russian Federation	1 year	3 years
Spain	20 years	20 years
United Kingdom	Not limited (covered under the reactor site licence)	N/A
USA	20 years	To be determined

N/A = not applicable.

Dry storage has become a mature technology and the quantities being placed into dry storage are increasing significantly. The inventory of spent fuel in dry storage in the above countries, as of 1 January 2001, was about 8693 t HM. The largest quantity is being stored in Canada, where since 1996, about 25 000 spent CANDU fuel bundles are being placed in dry storage annually. This is expected to increase when the Bruce Dry Storage Facility comes into operation by the end of 2002. At that point the OPG facilities will start placing in dry storage about 76 800 bundles per year. In Canada, the total inventory of spent fuel in dry storage at the end of 2000 was 4239 t HM. Also, in the USA, as many of the spent fuel storage pools reach their capacity, dry storage is a significant factor in the utilities' spent fuel storage strategy.

A helium storage environment is used in most of the systems, although air as well as helium is used in Canada. In the Republic of Korea only air is used to store CANDU fuel. Air is also being used for the experimental dry cask storage of RBMK-1000 fuel in the Russian Federation and a combination of CO₂ and air in the vault for the dry storage of MAGNOX fuel in the UK. Nitrogen is used in the MVDS in Hungary and at Ft. St. Vrain (USA).

Dry storage of spent LWR fuel in an inert atmosphere is licensed dependent on burnup and type of cask for temperatures up to 410°C in Germany and 380°C in the USA and Spain [7, 8]. Dry storage in nitrogen is licensed for a temperature of 350°C in Hungary for Zr/Nb clad fuel. Here, an application is being discussed with the regulatory body to allow raising this value, to at least 380°C. Dry storage is licensed in Canada and the Republic of Korea for temperatures up to 160°C for storage in air. In Canada, at the Pickering Station, storage is also licensed for temperatures of up to 360°C in a helium atmosphere.

In most cases, the dry storage containers are loaded with spent fuel under water. Following removal of the bulk water, the container requires vacuum or hot gas drying to prevent later aqueous corrosion and hydriding of the Zircaloy components and cask construction materials.

Care must be taken to ensure that the vacuum cycle is slow enough to allow full evacuation of all the water in the cask and also to prevent freezing of residual water due to evaporative cooling within the cask. Additionally, the drying level specification for reaching adequate dryness levels must be carefully defined. The drying processes utilised today are vacuum and hot-gas drying.

The **vacuum drying** process involves lowering the cover gas system pressure below the vapour pressure of the water at the drying temperature. The container is considered to be satisfactorily dry when the system pressure (e.g. ~6 to 10 millibar) remains constant for a specific time period (e.g. ~20 minutes). However, large amounts of residual water or relatively cold fuel may require a fairly long drying time or more sophisticated methods. The evaporation rate will also depend upon the temperature of the surrounding area. Vacuum drying is utilised in Canada, Germany, Spain and the USA.

In the **hot-gas drying** process, (process temperature 90–150°C), the fuel assembly is uniformly heated by the hot gas, to evaporate the water, in addition to blowing it away. This ensures complete removal of water residuals from all parts of the fuel assembly. The process time depends upon the heat input and flow rate of the hot gas. This process is utilized in Canada, Hungary, and the Republic of Korea. The fuel assembly container is considered to be dry, when the moisture content of the exhaust air equals the moisture content of the inlet air.

Most of the fuel in dry storage is clad with a zirconium alloy (Zry-2, Zry-4, Zr2.5Nb and Zr1Nb); however, dry storage experience also exists for magnesium and aluminium clad fuels. Average burnups of spent fuel presently in dry storage range from 4.5 to 33.5 GW·d/t HM, while the maximum burnups range from 7.5 to 50 GW·d/t HM. However, there is an almost universal tendency towards increasing the discharge burnup of the fuel elements. In Germany, for instance, average discharge burnups for PWR fuel assemblies have increased from 35 GW·d/t HM in 1983 to 50 GW·d/t HM in 1998, and a value of 65 GW·d/t HM is expected by 2003 in this decade. In the USA and other SPAR countries, most of the spent fuel assemblies are being, or will be in the very near future, discharged with burnup in excess of 45 GW·d/t HM. However, most presently licensed storage systems have burnup limitations of 45 GW·d/t HM, or less.

4.2. DRY STORAGE STATUS AND EXPERIENCE

Dry storage experience in the countries participating in SPAR CRP is described in this Section. Since the inception of the previous BEFAST project, other countries have also developed dry storage systems that are not reported here. As the amount of spent fuel stored dry in the SPAR countries is a large percentage of the total amount stored in dry conditions, this experience is considered to be representative of the total population. Total amounts of SF dry stored as of January 1, 2001 in the SPAR countries are shown in Table VII.

Canada — Atomic Energy of Canada Limited (AECL) has been storing spent fuel from its research reactors in a dry environment at the Chalk River Laboratories (CRL) for several decades. AECL has also designed reinforced concrete canisters (CCs) which have been used since the mid 1970s to store fuel from several NPPs. AECL has also developed two modular dry storage systems, CANSTOR (for CANDU fuel) [9] and MACSTOR (for LWR fuel) [10]. These are reinforced concrete modular structures that store spent fuel inside metal containers. A CANSTOR system was licensed in 1995 for the storage of spent CANDU fuel from Hydro Quebec's Gentilly-2 Nuclear Generating Station (NGS). Loading of the first CANSTOR module began in the latter part of 1995.

Table VII. Amount of spent fuel in dry storage

Country	t HM in dry storage (As of January 1, 2001)	t HM in dry storage (Forecast by 2010)
Canada	4 239	16 200
France	50	50
Germany	161	1700
Hungary	224	800
Japan	75	*
Republic of Korea	914	3 400
Russian Federation	0	6 000
Spain	0	220
UK	650	0
USA	2 430	~18 000

* The dry storage facilities at the Fukushima-Daiichi and the Tokai-Daini stations have capacities of 150 t HM and 260 t HM respectively. The amount of SF in dry storage in 2010, however, is not clear at this time.

Ontario Power Generation (OPG) uses Dry Storage Containers (DSCs) of its own design. They are transportable containers built of reinforced concrete, encased in (inner and outer) steel liners, which each hold 384 CANDU fuel bundles. OPG has had one operating dry storage facility at the Pickering site since 1995, and in January 2000 obtained a construction licence for a new storage facility at the Bruce nuclear site. The Bruce facility is expected to come into service by the end of 2002. The combined capacity of both facilities is 1.23 million bundles, and will provide sufficient storage for the rest of the stations' operating life.

Canada has not selected yet a strategy for long-term management of its spent fuel. This decision will be made under the framework of new nuclear waste legislation, which is expected to be enacted in 2002. Under this law the Canadian nuclear waste (spent fuel) producers will be required to create a separate waste management organisation (WMO) with the mission of develop and implement an integrated strategy for the long term management of all the spent nuclear fuel generated in Canada. The terms of this nuclear waste legislation are dictating research priorities in 2002. Current research being supported by the nuclear waste owners in Canada is focused on the requirements of the new law, the primary requirement being to issue a report examining options for the long-term management of nuclear fuel and recommending an approach. That report must be issued within three years of the law being enacted. Subsequently, the government will use the results of the study to select a preferred approach for management of the spent nuclear fuel, for which the WMO will become the implementing organisation.

France — Dry storage has been developed in France at the CASCAD facility for spent fuel that does not require prompt reprocessing. As of 1 January, 2001, approximately 50 t HM of heavy-water reactor fuel from the decommissioned EL4 Brennilis reactor had been stored in the vault at the CASCAD facility at Cadarache. The maximum capacity of the facility is about 100 t HM. Spent fuel is canisterized at the reactor (within a dry cell). Canisters, vacuum dried and filled with helium, are transferred to the dry-storage installation. Canisters are stored in storage wells ventilated by natural convection.

Germany — In Germany, based on their own practical experience, research and development, and demonstration programmes, only casks are licensed for dry storage [11, 12, 13]. Germany has licensed three central storage sites at Ahaus, Gorleben and Greifswald. The storage capacity at Gorleben has been increased from 1500 t HM to 3800 t HM using the CASTOR V/19 cask. Having completed the licensing of Ahaus for the CASTOR V/19 cask, the total storage capacity in Germany has been increased to 8345 t HM.

A basis for cask licensing is the integrity of the spent fuel structure and fuel rod cladding structure and the exclusion of systematic fuel rod cladding failures [14]. A systematic failure of the rod cladding in a non-corrosive environment is avoided by limiting stress and strain, whereby material's properties have to be considered.

Hungary — A modular dry vault storage (MVDS) started operation at Paks to store spent WWER-440 fuel in 1997, with an initial capacity of 162 t HM (1350 assemblies). An extension for 216 t HM (1800 assemblies) has been commissioned in 1999, with additional 4 vaults under construction presently. Although the general experience with dry storage in nitrogen-filled storage tubes has been very good, a review of emerging dry storage technologies is presently under way to decide whether additional cost savings may be achieved through use of an alternate technology.

Japan — A dry cask storage facility for the Fukushima-Daiichi power station of the Tokyo Electric Power Company started operation in September 1995. The facility has a capacity of 20 casks. These casks are installed horizontally in the building. The building presently stores five large metal casks, each one with a capacity of 52 BWR assemblies, and four smaller metal casks, with a capacity of 37 BWR assemblies [15]. A dry cask storage facility at the Tokai-Daini power station of the Japan Atomic Power Company will start operation in March of 2002. The capacity of the facility will be 24 metal casks, in which 61 BWR assemblies can be loaded into each cask. These casks will be stored vertically in the building.

Republic of Korea — Concrete canisters (AECL design) are being used for the dry storage of the CANDU spent fuel bundles at Wolsong. The CANDU spent fuel bundles are stored in a dry air environment in concrete canisters. Spent fuel is usually cooled in the water filled spent fuel bay for at least six years before being moved to the dry concrete silos. During pool storage, the fuel bundles are stored in trays (the defective fuel bundles are stored in a specially designed can), and later the bundles are loaded into a stainless-steel basket. The fuel in the basket is dried and the basket is seal-welded before the emplacement in the canister of the dry concrete silos. The silo, designed by AECL, is a cylindrical type device that is 6.5 m high with a 3.1 m outside diameter. A silo can contain as many as nine fuel baskets with 60 fuel bundles in each basket. A total of 140 concrete silos were installed at the Wolsong site. Dry storage started in 1992 and by the end of August 2001, 57 780 CANDU spent fuel bundles were being stored in 107 concrete canisters.

Russian Federation — A metal-concrete cask has been designed to store and transport RBMK-1000 fuel. In parallel, the design of a cask storage facility has been completed and its construction has already started at the Leningrad NPP. A similar facility has been designed for the Kursk and Smolensk NPP. Additionally, there are plans to build a Centralised Storage Facility by 2007 to store RBMK-1000 and WWER-1000 fuel, this project being the second phase of the RT-2 reprocessing plant of the Mining and Chemistry Combine (Krasnoyarsk). The capacity of this facility will be 33 000 t HM. A collection pad for interim storage of metal-concrete casks with spent fuel from submarines will be commissioned at RTP

Spain — Dual-purpose (storage and transport) metal casks have been designed, licensed and manufactured to store spent fuel from the Trillo Nuclear Power Plant. Each cask can store, in a helium environment, 21 PWR fuel assemblies with a maximum burnup of 40 GW·d/t HM, 4% initial enrichment and a minimum of 5 years cooling time. The casks are initially licensed for a period of 20 years for the storage mode and for 5 years for transport. The first loading is scheduled to start in 2002.

United Kingdom — The general experience with dry storage of MAGNOX fuel at Wylfa Power Station, both in the short term CO₂ cooled stores and in the two longer-term air-filled stores, has been very good.

The original design for Wylfa Power station provided, uniquely among the MAGNOX reactors, for discharge of the irradiated MAGNOX-clad, uranium-metal, fuel elements into one of three carbon dioxide filled dry stores for short-term storage (cooling and iodine decay), before transport to Sellafield for reprocessing. Later it was appreciated that an increase in storage capacity at Wylfa would make this facility, the largest of the MAGNOX stations, less vulnerable to transport or reprocessing difficulties. Accordingly, one and then a second dry store (DSC4 and DSC5) were built. The overall design of the cells was dictated by the need to access the existing fuel discharge route, and they were built butting on to the existing reactor. The chosen storage medium for already substantially cooled fuel was air with restrictions on element temperature of <150°C in the store. Under normal circumstances, the air moisture level was less, than 30 000 ppm water, or 50% relative humidity, whichever is the less.

The fuel elements are held in skips, which consist of 12 × 16 arrays of blind tubes, open at the top, and cooled by air which passes from below over the outside of the tubes, each of which holds one element. The air pressure is maintained slightly negative with respect to atmospheric pressure via an exhaust fan and a line of filters. This ensures that any leakage is inward, and that all air released to atmosphere is filtered. As fuel elements are added and removed from the store, the skips move along conveyor belts, i.e. the location of the fuel elements with regard to the building walls is not fixed.

The Wylfa Diverse Discharge Route (DDR) has been constructed beneath dry cell 4 to enable export of fuel direct from the DSC4 and 5 into off site transport flasks. The DDR comprises a shielded Flask Filling Cell (FFC) with remote fuel handling manipulator and outside the FFC, a number of flasks operating and handling stations together with a flask export station.

United States of America

Status

Most nuclear power plants in the USA were not originally designed with a pool storage capacity sufficient to accommodate the spent fuel generated over the operating life of the power plants. Utilities originally planned for spent fuel to remain in the spent fuel pool for a few years after discharge, and then to be sent to a reprocessing facility. Since reprocessing has been eliminated, and no other option for spent fuel disposition currently exists, utilities have expanded the storage capacity of their spent fuel pools by using high-density storage racks. This has been a short-term solution, however, given that many utilities have reached, or soon will reach, their spent fuel pool maximum storage capacity.

Utilities have developed independent spent fuel storage installations (ISFSIs) as a means of expanding their on-site spent fuel storage capacity on an interim basis until a geological

repository is available to accept spent fuel for permanent storage. By the end of 2001, 19 independent spent fuel storage installations were licensed. In 2001, approximately 860 t HM of spent fuel were transferred to dry storage, a 60 per cent increase from the previous year. US utilities loaded a record 80 canister or casks systems during 2001, bringing the total number of spent fuel containers to 302. This represents ~3300 t HM, or slightly less than 8 per cent of the total spent fuel inventory.

Loading of canister or cask systems is expected to be well over 100 systems in 2002 alone. It is now estimated that by 2010, almost one-third of US spent fuel will be in dry storage. This growth reflects the increased availability of newly licensed dual-purpose (storage and transportation) systems, the increasing number of plants that are losing full core reserve, and the decommissioning needs for shutdown plants.

Dry storage beyond 20 years

The US Nuclear Regulatory Commission (NRC) currently limits the licence term for an independent spent fuel storage installation to 20 years from the date of issuance. Licences may be renewed by the Commission at or before the expiration of the licence term. The first 20-year licence (Surry) will be expiring in January 2006. In preparation for licence renewal applications, the NRC Spent Fuel Project Office is preparing the regulatory guidance for allowing dry storage beyond 20 years. The licence renewal is not intended to be an exercise in re-licensing, and does not provide an opportunity to impose requirements, that are beyond those that were met by the design, when it was initially licensed by the NRC. However, the effects of aging on dry cask storage systems, structures, components, and support activities need to be sufficiently understood and managed such that the ISFSI can be operated during the renewal period without undue risk to the health and safety of the public. Therefore, the current licensing basis for the ISFSI will be carried forward through the renewal period, and while the duration of the period has not yet been determined, it could conceivably be anywhere from 20 to 100 years.

Dry storage of spent fuel with burnup in excess of 45 GW·d/t HM

Due to the trend to longer fuel cycles, reactor operators are driven toward fuel designs that can reach higher burnup levels. Presently, PWR fuel designs are available with peak rod burnup of 62 GW·d/t HM (or ~55–60 GW·d/t HM assembly average), and BWR fuel designs with peak pellet burnup of ~70 GW·d/t HM (or ~55 GW·d/t HM bundle average). Approximately 5000 PWR assemblies and 1000 BWR bundles, with assembly-average burnups in excess of 45 GW·d/t HM, have been discharged to wet storage at the US reactor sites. Starting in 2001 for PWRs, and just a few years later for BWRs, the majority of assemblies to be discharged in any given year is, or will be, characterized by assembly-average burnup greater than 45 GW·d/t HM. By 2007, over 90% of assemblies to be discharged will have burnups greater than 45 GW·d/t HM.

Presently, the Nuclear Regulatory Commission's Standard Review Plan (NUREG-1536) does not address dry storage of spent fuel having burnups in excess of 45 GW·d/t HM. In Interim Staff Guidance 11 (ISG-11), Rev.1, the NRC Staff proposes that the relevant Standard Review Plans be modified to permit the storage and transportation of high burnup fuel assemblies under the following conditions:

“... the staff has reasonable assurance that fuels having average assembly burnups exceeding 45 GW·d/t HM can be safely stored and transported if the following acceptance criteria are met:

- I. A high burnup fuel assembly (i.e. burnups greater than 45 GW·d/t HM) containing Zircaloy clad fuel may be treated as intact if both of the following conditions are met:
 - A1. No more than 1% of the rods in the assembly have peak cladding oxide thicknesses greater than 80 micrometers.
 - A2. No more than 3% of the rods in the assembly have peak cladding oxide thicknesses greater than 70 micrometers.
- II. A high burnup fuel assembly should be treated as potentially failed fuel if either of the following conditions are met:
 - B1. The fuel assembly does not meet both criteria A1 and A2; or
 - B2. The fuel assembly contains fuel rods with oxide that has become detached or spalled from the cladding.

The administrative controls section of the SAR technical specifications should specify a program to be implemented by the cask licensee to assure the criteria described above are met prior to loading the cask with high burnup fuel. As part of this program, the licensee may use cladding oxidation thickness measurements or predictions based on consideration of reactor operation variables affecting peak cladding oxidation; i.e. in-core flux, length of cycle, number of cycles, power excursions, coolant temperature and amount of time at that temperature, the coolant water chemistry, and the cladding material. In cases where there are no previously documented measurements of the oxide thickness to validate cladding oxidation predictions, the program may have to incorporate peak cladding oxide thickness measurements.”

This interim staff guidance leads to the requirement that calculations or measurements of zirconium oxide thickness be performed for all fuel above 45 GW·d/t HM. Since analytic models do not account for the large scatter observed in the corrosion performance of zirconium alloys, it may be impossible to show that 3% or 1% of rods will not exceed 70 micrometer or 80 micrometer, respectively, without actually conducting measurements in every case. The number and scope of such campaigns would be prohibitively expensive. In addition, any assemblies containing structurally sound rods, but with oxide thickness in excess of 70 micrometer or 80 micrometer would have to be treated as damaged assemblies for storage purpose.

US utilities are making the case for removing the oxide limits presently in ISG-11, Rev. 1, because excessive in-reactor corrosion is prevented by adherence to approved oxide limits. In a few limiting cases, spallation, or flaking, of Zircaloy-4 cladding has been observed. However, severe spallation is not normally observed. Except when severe spallation is observed, US utilities argue that there is no need for special restrictions.

4.3. RESEARCH ACTIVITIES RELATED TO DRY STORAGE

Research activities relevant to dry storage have been carried out since the early 60s and have been reported in earlier BEFAST projects and elsewhere.

Canada — AECL and Ontario Hydro (OPG’s predecessor) conducted a dry-storage experimental program at the Whiteshell Laboratories from 1978 to 1996. Long term experiments were carried out in which fuel bundles discharged from OPG’s Pickering and

Bruce stations were characterized in detail and placed in dry storage, in concrete canisters, under controlled conditions [16]. Experiments were run keeping the test bundles in dry air and moist air at 150°C and also at environmental temperatures. Intentional cladding defects were made in the outer ring elements of half of the fuel bundles to evaluate the performance of failed fuel. Fuel bundles were removed from the canisters at intervals of a few years for interim examinations. Fuel elements were subjected to both non-destructive and destructive tests.

The results from these tests provided a measurement of the oxidation rate of failed CANDU fuel under conditions of storage in air. Results indicated that although the fuel matrix underwent considerable oxidation in the elements with cladding failures, the rate of oxidation was too low to cause expansion of the fuel matrix and further cladding damage. The fuel elements without cladding failures did not suffer any noticeable defects over several years in dry storage [17, 18].

Germany — Dry storage was from the beginning a necessary element in the back end of the fuel cycle strategy of Germany. Therefore, an extended R&D was performed to define the conditions for safe storage of spent fuel under dry conditions. The detailed results from the different investigations were put together early into performance prediction codes and those codes were validated in a demonstration programme using original spent fuel and representative storage casks.

Presently, dry spent LWR fuel storage is licensed for single fuel assemblies with rod burnup up to 65 GW·d/t HM. This allows dry spent fuel storage of reloads with batch averages up to 55 GW·d/t HM. Closing the nuclear fuel cycle by direct disposal of the spent fuel, combined with prior long-term dry storage, requires that the end-of-life conditions (EOL) of the fuel assembly (FA) be considered for assessing performance of the disposal approach. In order to take care of FA with higher burn up, the loading schemes for transport and storage need to be defined in more detail on specific, instead of enveloping, EOL data for better response to all licensing requirements. Optimized cask loading schemes, taking into account the available FA with specific neutron and gamma emissions as well as the results from statistical performance calculations, provide a better usage of all transport and storage casks. Additionally, post-pile creep behaviour of highly irradiated commercial cladding materials needs to be analysed to extent licensing to even higher burnups.

Japan — R&D for dry cask storage has been carried out mainly by the Central Research Institute of Electric Power Industry (CRIEPI) [15]. The first R&D program, undertaken between 1987 and 1991, dealt with metal cask storage technology and its results provided reference data for the licensing of dry cask storage technology in Japan. The main subjects studied were: (1) fuel cladding temperature tests and analysis considering heat transfer within the cask, (2) creep tests of fuel cladding, (3) containment tests of the cask lid structure, (4) cask drop tests, (5) material tests of ductile cast iron, (6) building collapse and heavy objects drop onto cask, and (7) cask toppling by earthquake.

CRIEPI also carried out from 1992 to 1996 an advanced research program on spent fuel storage technology. Major activities of this new program dealt with the evaluation and related tests on dry storage methods suitable for spent high burnup fuel and MOX fuel in terms of safety and economy, which included application of burnup credit and temperature analysis. The activities also included various designs and development projects for advanced dry storage technologies, such as a new type of fuel basket with efficient heat removal and subcriticality, and an advanced concrete cask with improved safety and economy.

A further R&D program for interim storage of “Recycled Fuel Resources” (i.e. spent fuel) was started in 1997 and will run until 2003. This program deals with verification tests related to concrete modular storage technology such as horizontal concrete silos and concrete casks and also deals with integrity of spent fuel during dry storage.

Republic of Korea — Dry storage has been regarded as a safe and economical method for long-term storage of spent fuel in the Republic of Korea. However, the higher temperatures associated with dry storage, as compared to wet storage, are considered to be a possible point of vulnerability. For this reason, KAERI (Korea Atomic Energy Research Institute) has undertaken a research program to study fuel degradation mechanisms. Oxidation tests were performed on irradiated and non-irradiated UO₂. The tests results included the fact that a big difference in the oxidation behaviour was observed between the irradiated and non-irradiated fuel. A failure analysis computer code was also developed as part of the research program.

SFINEL (Spent Fuel Integrity EvaLuator) is an integrated computer code system for predicting spent fuel rod integrity by taking account of burnup history [19]. This code has combined two separately developed programs by restructuring them into a single module with a graphical user interface. One program is SPENFIP (SPENt Fuel rod Internal Pressure evaluator), which was prepared for PWR fuel rod internal pressure evaluation in consideration of the cyclic power history. The other one is SIECO (Systematic Integrity Evaluation Computer) program written by KAERI for estimating spent fuel cladding integrity during dry storage. In fact, the main structures of SPENFIP and SIECO programs are based on the GT2R2 (Gapcon Thermal 2) and DATING (Determining Allowable Temperatures for dry storage of spent fuel in Inert and Nitrogen Gases) respectively, which both were originally developed by PNL.

Russian Federation — A research program is underway, since 1999 and will run until 2005, with the aim to study the theoretical and experimental assessment of permissible dry storage conditions for spent fuel. The program is studying the degradation mechanisms of the Zr1Nb fuel cladding, the structural stability, mechanical properties and corrosion resistance of fuel structural materials and cask components, as well as the way to deal with damaged fuel. Initial results show that creep of this cladding can be neglected if the temperature in the cladding in dry storage is limited to 330°C–380°C, depending on specific fuel characteristics. Research activities are carried out to assess terms and conditions of dry storage of SF in metal-concrete casks, and to provide data to augment the technical basis and criteria for evaluating the safety of SF storage and transportation systems and for extending dry cask storage licences. On the basis of the present knowledge about the RBMK fuel condition after in-reactor operation and long-term wet storage, it has been concluded that RBMK fuel may be safely stored in dry facilities for 10 years in an inert gas environment at temperatures of up to 300°C.

Spain — A recent R&D program performed on six high burnup (>50 GW·d/t HM) fuel rods that had been dry stored in air for more than 25 years, at temperatures not in excess of 175°C, has proven that the fuel cladding has not suffered any degradation under these conditions.

United States of America

Storage of spent fuel beyond 20 years

The NRC, EPRI, and the DOE Offices of Civilian Radioactive Waste Management (DOE-RW) and Environmental Management (DOE-EM) are participating in a cooperative research

program (Dry Cask Storage Characterization Program) to determine the long-term integrity of dry cask storage systems and spent nuclear fuel under dry storage conditions. The program objectives are (1) determine the long-term integrity of dry storage cask systems and spent nuclear fuel under dry storage conditions, and (2) provide data to augment the technical bases and criteria for evaluating the safety of spent-fuel storage and transportation systems, and for extending dry cask storage licences.

The Castor-V/21 cask at INEEL was selected for study under this programme. A summary of the scope of work performed on the Castor-V/21 cask and fuel can be found in Ref. [20]. No significant degradation of the Castor V/21 cask system was detected. Supporting evidence for this lack of significant degradation are (1) gas analyses showed neither signs of air ingress into the container, nor signs of cladding failure leading to fission product release, (2) visual examination of the cask lid O-rings suggested they were in adequate condition to maintain a seal; (3) no major crud spallation from the fuel rod surfaces; and (4) all materials inside the cask, including the assemblies, appeared the same as they did in 1985.

Storage of spent fuel with burnup greater than 45 GW·d/t HM

From a dry-cask licensing point of view, NRC requires that "...the spent fuel cladding must be protected from degradation that leads to gross ruptures." [21]. To support their case, applicants need to supply data and calculations to show that creep strains accumulated during dry cask storage do not lead to gross rupture. Most of the creep strain will be accumulated during the first 5 years of dry cask storage.

Given the wide variability in thermal creep resistance of unirradiated cladding and the limited database for irradiated cladding, it is desirable to expand the database for irradiated cladding. In particular, the creep behaviour of high-burnup rod cladding vs. hydrogen content and neutron damage is not well characterized. An experimental programme, sponsored by NRC, EPRI, and DOE, was initiated in 2001 on cladding materials irradiated in several US reactors. The main objective of the creep tests is to generate data on primary and secondary creep strain vs. time. Materials from rods having experienced burnups from 35 to >70 GW·d/t HM will be tested.

With regard to the application for licence extension to beyond 20 years, temperatures are expected to be too low (<200°C) for creep to be a relevant failure mechanism. However, having enough residual creep life after 20 years of dry cask storage provides some level of confidence in making a case for longer periods of storage. Therefore, the demonstration of significant creep life after 15 years of dry cask storage will directly support the industry approach for the current licence period of 20 years. From the DOE perspective, the transition from dry cask storage to the permanent repository site results in an elevation of cladding temperature to 350–360°C. Thus, significant residual creep life is directly relevant to the data needs of DOE as well.

Burnup Credit. US industry licensees have indicated to NRC that issuance of ISG-8, Rev. 1 has provided the impetus to prepare a new generation of high-capacity cask designs using BUC in the safety evaluations. However, adherence to the recommendations of ISG-8, Rev. 1 unnecessarily limits the population of SNF that can reside in a BUC cask, and restrict the use of BUC casks for fuel with high initial enrichment values. Therefore, NRC's Office of Regulatory Research (RES) conducted a research program to clarify guidance on acceptable technical approaches, to develop approaches for expanding the range of BUC, and to address regulatory needs for safe, simple, and cost-effective implementation of BUC. Since the

implementation of BUC is an international activity, input and experience from domestic and international experts and organizations were sought and incorporated into the RES program. The expert panel was convened to participate in a process of developing Phenomena Identification and Ranking Tables (PIRT). RES programs have used “PIRT” panels to provide input and information on several technical issues during the past few years. The PIRT panel for BUC consisted of 16 experts: seven international participants from countries with significant interest and experience working on BUC issues, and nine participants representing a variety of domestic organizations and areas of expertise. The main goal of the PIRT panel was to identify phenomena, parameters, procedures, etc. that influence the determination of the neutron multiplication constant (k_{eff}) for spent fuel in a cask environment, provide a graded (e.g. high-importance, moderate-importance, or low-importance) ranking of the phenomena and, as appropriate, judge the uncertainty associated with each phenomenon. Besides its primary objective, the PIRT process can also facilitate a beneficial exchange of information and ideas that leads to improved understanding of the issues and practical approaches for effective implementation of BUC within the licensing process. The activities of the PIRT panel are documented in Refs [22, 23].

5. SOURCE TERMS

Spent fuel main characteristics relevant to the design of storage systems (thermal power and radiation characteristics) depend on the initial conditions (fuel type and geometry, initial enrichment, etc.), irradiation history (residence times at the core, linear power rate, partial and accumulated burnups, re-shuffling schemes, etc.) and time since its last discharge from the core.

These characteristics may be obtained either by calculations based on historical records of measurements made during fabrication of the fuel and on real reactor operation history while the spent fuel considered was in operation or by calculations based on direct (burnup meters) or indirect measurements made at the time of interest. The accuracy of the results obtained in the first case will depend on the degree of administrative control maintained over the records and on the accuracy of reactor measurements. In the second case, it will depend on the uncertainties inherent in the measurement techniques employed.

Calculation of spent fuel characteristics after completion of the in-service operation is a mature technology, which is commonly performed by validated codes. Highly developed and complex codes are used at the NPPs, to track the evolution of each of the fuel elements at the core, to assist the operators in the design, heat transfer analysis and fuel management of their reactors. They involve the use of multi-group neutron spectrum and cross sections to calculate the nuclide composition of the fuel both as a function of time and each core region.

However, these codes are expensive to use and do not calculate the entire number of nuclides present in the fuel elements that might be of interest for the out-of-reactor fuel management in the back-end of the fuel cycle, this fact being the reason why simpler codes were developed for that reason. Among these, the most popular ones are the ORIGEN (Oak Ridge Isotope Generation, developed at ORNL, USA), KORIGEN (FZK, Germany) and other codes, like APOLLO developed in France, or the HELIOS developed in Sweden. They consider, in general, a single decay and photon database and cross section libraries, which have been specifically developed for each type of reactor. However, the specific fission and capture cross sections are strongly dependent on each burnup (i.e. neutron spectrum change during irradiation) and international benchmarking and validation studies have been, and are still

today being, carried out to compare results obtained with the different codes. Experimental studies are also being performed to determine the fission and capture cross sections of different isotopes as a function of the neutron flux and energy.

An important aspect of these codes is that specific results strongly depend on input data, particularly on the composition of the input nuclear material. In this regard, trace constituents may play a significant role in out-of-reactor situations (as interim storage or final disposal) and it may be important to detail impurities present in fresh fuel and in the assembly structural materials, if a complete and detailed isotopic characterisation is needed.

Another intrinsic feature of these codes is that large time steps may be used in the decaying phase (post irradiation decay is a simple calculation generally accomplished using a single decay step). However, multiple small time steps should be used to describe their irradiation history to prevent large errors in the results.

Output generally comprises fractional isotopic compositions, radioactivity, residual thermal power and neutron and photon production rates.

Three of the main safety objectives to be considered in spent fuel interim storage are to guarantee:

- Subcriticality,
- Heat dissipation, and
- Radiation shielding.

Therefore, the following source terms must be well known:

- Initial and final fissile material enrichment,
- Residual heat power, and
- Radioactive inventory.

5.1. SUBCRITICALITY

Criticality calculations are usually performed taking into account the initial enrichment (^{235}U or ^{235}U & Pu fissile) of the fresh fuel element and using validated codes such as KENO and MCNP. Considering fresh fuel is a very conservative assumption, which in many cases penalises the capacity of the storage system. Although the use of burnup credit (considering the decrease in reactivity of the fuel, once it has been burned in the reactor) has been licensed in many countries for wet storage, the use of this approach for dry storage systems is still today the subject of co-operative international efforts.

The plutonium and the uranium composition changes with burnup and that of plutonium changes with cooling time. The fissile plutonium (^{239}Pu and ^{241}Pu) inventory decreases with higher burnup and also with cooling time because ^{241}Pu decays to ^{241}Am . The ^{236}U content increases with burnup and will be for burnup greater than 50 GW·d/t HM larger than the ^{235}U fraction. The fissile material composition (^{235}U and Pu) will always be lower in the spent than in the fresh fuel.

5.2. THERMAL CALCULATIONS

From a thermal power point of view, fission products and transuranics are the most relevant contributors. During almost the first hundred years of cooling time, spent fuel thermal power is due to fission products decay, mainly to the two decay chains $^{90}\text{Sr} \rightarrow ^{90}\text{Y}$ and $^{137}\text{Cs} \rightarrow ^{137\text{m}}\text{Ba}$. Thermal power is due to the transuranic elements, mainly to ^{241}Am , from 200 to 1000 years cooling time.

Spent fuel residual thermal power will increase with the burnup. Its variation will be higher during the first 300 years cooling time and also when the initial enrichment is also higher.

The thermal power at the discharge of spent fuel decreases with cooling time. Table VIII. shows an example for an uranium fuelled 1000 MW(e) reactor and a maximum burnup of 33 000 MW·d/t HM.

Table VIII. Spent fuel decay heat after discharge

Time after shutdown	Initial thermal power, %	Thermal power, MW
1 second	17%	500
1 minute	5%	150
1 hour	1.5%	45
1 day	0.5%	15
1 week	0.3%	9
1 month	0.15%	4.5
1 year	0.03%	1
10 years	0.003%	0.1
100 years	0.001%	0.03
1 000 years	0.0002%	0.006

The amount of decay heat is influencing a lot of decisions in the back end of the fuel cycle, especially after longer time periods. The uncertainties of the computer codes might increase with time. Therefore, validation of the codes in use might be necessary. The Swedish performed such work by calorimetric studies in CLAB with 15 years old fuel [24, 25]. Germany used the Swedish data also for code validation. For the time being, a project for determination of the residual heat from spent fuel assemblies by a calorimetric method has started. The project will be completed in 2003.

5.3. RADIATION SOURCES

The radioactive inventory of spent fuel is due to the decay of fission products, actinides, mainly transuranics, and activation products. The contribution of the latter ones is lower than the others. This radioactivity decreases with cooling time, as shown in Table IX.

Table IX. Radioactivity decrease with cooling time

Time after shutdown, years	Relative activity
0	1
5	5×10^{-3}
50	1×10^{-3}
100	3.33×10^{-4}
1 000	1.33×10^{-5}

During the first 150 years of cooling time, the radioactivity of spent fuel is due to the decay of fission products mainly to the $^{137}\text{Cs} \rightarrow ^{137\text{m}}\text{Ba}$ and $^{90}\text{Sr} \rightarrow ^{90}\text{Y}$ radioactive chains. From 200 to 500 years of cooling time, the ^{241}Am , daughter of ^{241}Pu , is the main radionuclide with alpha decay to ^{237}Np .

It should also be mentioned that the noble gas activity in the spent fuel is decreasing fast with time. The half-life of ^{85}Kr is 10.756 years. As a consequence, for longer time storage safety concerns related to the gaseous activity released from the spent fuel under certain circumstances are less meaningful as they are in the beginning of spent fuel storage.

Alpha radioactivity from the decay of the actinides is lower than beta-gamma radioactivity from fission products' decay during the first 200 years of spent fuel cooling time. Later, alpha radioactivity due to the actinides is the main contribution to the total radioactivity of spent fuel.

5.4. ASPECTS ASSOCIATED WITH HIGH BURNUP FUEL

To achieve higher burnup, UO_2 -FAs need a higher ^{235}U -enrichment, and MOX-FAs a higher Pu_{fiss} -content. The consequence is generally a higher thermal load of the fuel to obtain a higher FA-burnup within the same irradiation time. If a certain amount of MOX-FAs replace U-FAs, differences of the reactivity behaviour vs. time have to be also considered. The MOX reactivity curve is flatter than that for Uranium (Figure 1). The point of intersection depends on the relation between ^{235}U -enrichment and Pu_{fiss} -content. A high preponderance of the Pu_{fiss} -content shifts the point of intersection to the left and leads to a considerable higher thermal load of the MOX-rods at burnups beyond 20 to 30 $\text{GW}\cdot\text{d}/\text{t HM}$.

For dry storage the evolution of the decay heat and the neutron activity with burnup is important to assess cask temperatures and shielding capacities. A burnup increase of 25% from 55 to 69 $\text{GW}\cdot\text{d}/\text{t HM}$ increases the decay heat of both UO_2 and MOX FAs by roughly the same amount (See Fig. 2.) [26]. However, the decay heat of a UO_2 -FA with around 2 kW at the beginning of storage after 5 years cooling time is less than a half of a MOX FA of comparable range of burnup. With increasing cooling time the decay heat decreases, however more and more slowly. Whereas for a UO_2 -FA it takes about 10 to 20 years to drop below 1 kW, it takes roughly 100 to 200 years for a MOX FA.

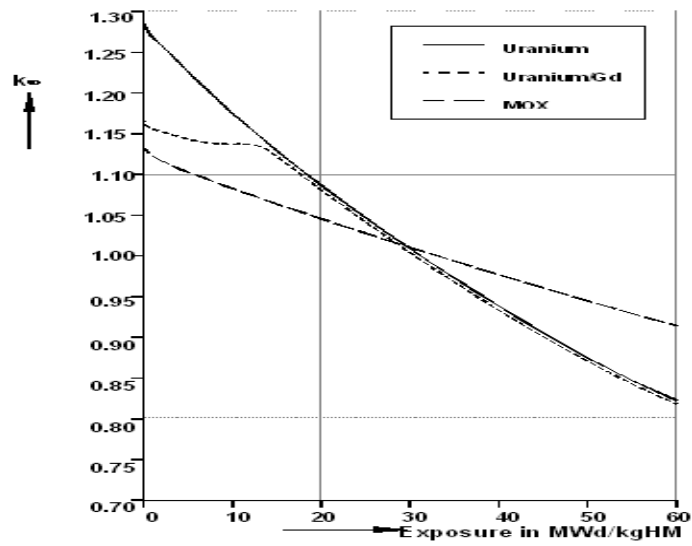


FIG. 1. Reactivity of MOX and UO_2 fuel depending on burnup.

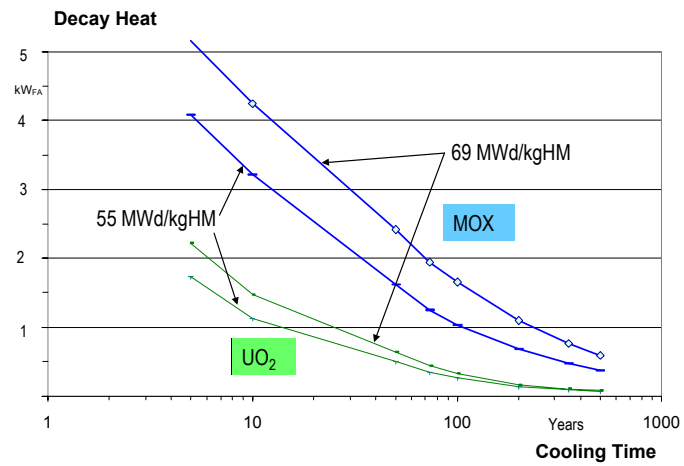


FIG. 2. Decay heat of UO_2 and MOX FAs depending on burnup and cooling time (FA-Discharge Burnup 55 and 69 GW·d/t HM).

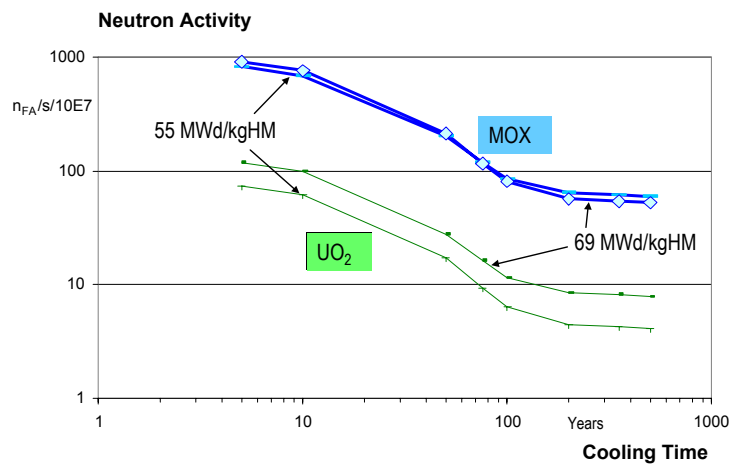


FIG. 3. Neutron activity of UO_2 and MOX FAs dependent on burnup and cooling time (FA-discharge burnup 55 and 69 MW·d/kg HM).

The neutron activity exhibits somewhat different features than the decay heat (Fig. 3.) [26]. Between 10 and 100 years cooling time it drops by one order of magnitude and above 100 years the reactivity remains relatively constant. The difference between MOX and UO₂ FAs amounts to roughly a factor of ten.

With regard to a storage extension, the graphs show that it takes about 100 years cooling time for both decay heat and neutron activity of MOX fuel to reach the UO₂ level at the beginning of storage.

5.5. ASPECTS ASSOCIATED WITH BURNUP CREDIT

Nuclear criticality safety was established as a discipline more than 50 years ago in response to several accidents that had occurred in nuclear weapons programmes. The importance of the safe handling of fissile materials was recognised at an early stage, both by the scientific community and the responsible authorities. Based on intensive experimentation with a large variety of configurations and materials substantial progress has been made in developing nuclear data and computer codes to evaluate criticality safety for nuclear fuel handling. The accuracy and reliability of computer code calculations has been extensively improved over the years. However, the criticality safety analysis of spent fuel systems has traditionally assumed that the fuel is fresh. This “fresh-fuel assumption” provides a well-defined, bounding approach to the criticality safety analysis. It eliminates all concerns related to the fuel operating history. However it ignores the decrease of reactivity with fuel burnup. This results in significant conservatism in the calculated value of the system's reactivity. The state-of-the-art calculation tools for criticality safety evaluations have led to reduction of the uncertainties in the evaluated neutron multiplication factors. It has allowed rational and more economical designs for manipulation, storage and transportation of fissile materials. Improved calculation methods allow taking credit for the reactivity reduction associated with fuel burnup.

The concept of taking credit for the reduction of reactivity is commonly called burnup credit. The reduction that occurs with fuel burnup is due to the change, i.e. reduction, in concentration of fissile nuclides and the production of actinides and fission product neutron absorbers. This leads to a reduction in the analysis conservatism while maintaining an adequate criticality safety margin.

Many burnup credit applications were licensed early in the 1980s. Since the mid-1980s, the US utility industry, the US Department of Energy (DOE), and the US Nuclear Regulatory Commission (NRC) have actively considered the incentives, benefits, and obstacles associated with implementing burnup credit in the criticality safety evaluation for storage, transport, and disposal of spent nuclear fuel (SNF).

Burnup credit methodologies have been accepted for the criticality safety analysis of on-site wet storage of spent fuel at different power plants operating in several countries. Some applications, e.g. in Spain were licensed more than 10 years ago, and were considered acceptable at that time on the basis of the overall conservatism of the analysis [27, 28].

Burnup credit is defined as the consideration of the reduction in reactivity, associated with the use of the fuel in power reactors. Changes in the isotopic composition during fuel burnup, which result in a reduced reactivity can be conveniently characterized by the reduction of the net fissile content, the build-up of actinides, the increase of the concentration of fission products, and the reduction of burnable absorber concentration where applicable. In practice, the conservative use of burnup credit requires consideration of all fissile isotopes, and allows

consideration of any neutron absorbing isotopes for which properties and quantities are known with sufficient certainty. The different levels of burnup credit, which are commonly used are described as follows:

- Credit for the net decrease of the fuel fissile content, taking into account both burnup and buildup of the different fissile nuclides (*net fissile content level*).
- Credit for the net fissile content and for the absorption effect of actinides (*actinide only level*).
- Credit for the actinides and the neutron absorption in fission products (*actinide plus fission product level*).
- Credit for the presence of integral burnable absorbers in the fuel design (*integral burnable absorber level*). This credit uses the maximum reactivity of the fuel, which is often not the initial reactivity. Although not really consistent with the definition of burnup credit, it is generally considered to be a level of burnup credit because fuel depletion calculations are needed to determine the reactivity state as a function of burnup.

The incentives first emerged with spent fuel storage pools. Lack of off-site alternatives (i.e. reprocessing, permanent disposal, or interim storage) provided significant incentives for utilities to obtain optimum use of the fixed pool storage capacity currently in place. This situation was aggravated by the demand to optimize pool storage space towards increased initial enrichments. This trend continues to the present. Thus the simple, yet conservative, assumption of using unirradiated fresh fuel for the criticality safety analysis became a significant economic barrier to continued operation of reactor power plants.

Efforts were initiated to evaluate the incentives and seek resolution of technical issues associated with the use of burnup credit in SNF storage and transport casks. In contrast to many countries where burnup credit is desired primarily to increase the allowable enrichment within existing cask designs, the USA nuclear industry is seeking to develop a new fleet of storage and transport casks that are optimized for the anticipated SNF contents. Rail casks with capacities of 32 PWR assemblies are being designed — an ~30% increase over existing storage cask concepts. These increased cask capacities can enable a reduction in the number of casks and shipments, and thus have notable economic benefits while providing a risk-based approach to improving safety. Arguments for improvement in safety have noted that the fewer shipments required with burnup credit cask designs will reduce the radiation exposure to both workers and the public as well as reducing the potential for a transport accident involving a cask.

Incentives for use of burnup credit in BWR applications have not been as significant as for PWR applications. The reason for this reduced incentive is that BWR fuels have less reactivity than PWR fuels and increased use of neutron poisons in intervening regions between assemblies have proven effective for maximizing capacities and allowing fairly high initial enrichments. Thus, the incentives are largely limited to reducing the cost of neutron poison plates and allowance for higher initial enrichment fuel (up to 5 wt% ²³⁵U).

The implementation of burnup credit has been dictated in many countries by different needs. Some examples are as follows:

- Introduction of higher enriched fuel in the existing storage, reprocessing or transport systems. In these cases, the use of burnup credit may alleviate the need for new installations or for extensive changes to existing facilities or equipment for spent fuel management activities.

- Burnup credit can increase the storage capacity (both on-site and independent) by allowing smaller centre-to-centre distances in the fuel storage systems.
- Burnup credit can be used for new casks, to increase cask capacities over current design capacities, to reduce the number of shipments needed.

In the area of spent fuel storage, burnup credit and its resultant capacity improvements can alleviate or minimize the environmental impacts associated with expanding or building new storage pools or dry storage facilities. Burnup credit can be used to maintain production rates at existing reprocessing facilities, even while fuel enrichments increase, thus avoiding the environmental impacts of constructing new facilities, or expanding old ones. For disposal of spent fuel, burnup credit is considered to be a necessity for any viable disposal scheme. Ignoring the reduced reactivity from burnup credit could lead to larger disposal sites and unnecessary use of land.

Application of burnup credit requires knowledge of the isotopic inventory of the irradiated fuel for which burnup credit is taken. This knowledge is gained by using depletion codes. The uncertainty of a depletion code is controlled and established through verification of that code, usually by comparison with suitable and appropriate experiments. In-core reactor measurement data are important for verifications of depletion codes. For burnup credit applications, particular significance is attached to comparisons of calculated to measured isotopic concentrations.

Due to the depletion analysis, that has to be performed to determine the isotopic content, the results of the criticality calculations become dependent on the reactor operation conditions assumed for the depletion analysis. Due to the wide variety of fuel irradiation histories, it is necessary to look for a bounding irradiation history given by those fuel operation conditions that lead to the highest spent fuel reactivity in the criticality analysis.

The outcome of performing a burnup credit criticality analysis is the determination of a burnup credit loading curve for the spent fuel management system of interest.

The use of burnup credit implies a verification of the fuel burnup before loading for transport, storage, disposal, or reprocessing each assembly, to make sure that the burnup level achieved complies with the criteria established.

Criticality Safety Regulations for fuel storage and transport do not prohibit the use of burnup credit in the criticality safety analysis of spent fuel systems. In particular, IAEA Safety Series 116 [29] explicitly describes the implementation of burnup credit in storage facilities, and IAEA Safety Standard Series No. ST-1 [30] contains the requirements for transport. Likewise, national regulations do not prohibit the use of burnup credit for criticality safety analysis of spent fuel systems.

However, given the difficulty of validating the analysis codes used to demonstrate criticality safety and the challenge of demonstrating that all the relevant effects have been considered, no unrestricted burnup credit methodology has yet been approved. The regulatory authorities of different countries have approved the reactivity effect of a limited number of nuclides. Which of the burnup credit levels have been approved, depends on the function and the characteristics of the given spent fuel management system.

To assist operators and dry storage and transport system designers in the implementation of burnup credit, several devices to corroborate the declared characteristic values of an irradiated

fuel assembly have been developed. Among these, the PYTHON (developed in France in collaboration between EDF and CEA), the BNFL burnup monitor, and FORK (USA) devices are the most well-known ones.

United States of America

In the USA, the regulations for transport and dry storage of SNF are promulgated in 10 CFR Parts 71 and 72, respectively. Current industry practice is to design spent fuel canisters that can be inserted into either dry storage or transport cask designs, thus eliminating the need for further handling of spent fuel assemblies once they have been loaded in a canister at the facility. The transport regulations of Part 71 provide the limiting condition for criticality safety, including consideration of water in-leakage to the canister. Although neither Part 71 nor Part 72 has any specific requirement that would prevent burnup credit from being implemented in the safety analysis, the historical practice to criticality safety analyses for transport and dry cask storage has been to assume the SNF is unirradiated, with uniform isotopic compositions corresponding to its maximum initial fresh fuel enrichment in ^{235}U .

Because there has not been a licensing application using BUC to date, there has been no regulatory experience in the USA with licensing of a PWR or BWR transport cask. The US industry and the US DOE have supported a significant number of technical investigations, focused primarily on PWR fuel, to provide a foundation for implementation of BUC in the USA. Based on this information, the SFPO issued Interim Staff Guidance 8 (ISG-8, Rev. 0), entitled Limited Burnup Credit in May 1999. Supported by initial confirmatory research, Revision 1 to ISG-8 (ISG-8, Rev. 1), entitled Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel Transport and Storage Casks, was released in late July 1999. ISG-8, Rev. 1 provides increased flexibility in the guidance for licensing of SNF casks. Some of these recommendations restrict the amount of burnup credit that can be used in the safety assessment (e.g. credit for fission products is not allowed) and some restrict the SNF population (type and range of burnup that would be allowed in a BUC cask). These recommendations were based on: 1) technical information available to NRC at the time ISG-8, Rev. 1 was issued; 2) consistency with the industry standards developed for criticality safety of fissionable material (ANSI/ANS-8.1) and LWR fuel (ANSI/ANS-8.17) in operations outside reactors; and 3) recognition that experience and additional research will provide a basis for additional guidance.

Tables X–XII provide a summary of BUC use and practice in the SPAR countries.

6. FUEL INTEGRITY

6.1. DEFINITION OF FUEL INTEGRITY

The terminology used to describe fuel integrity varies between countries. For example, in Germany or in Central and Eastern European countries, any assembly containing fuel rod cladding defects would be classified as “failed.”

In the USA, “damaged” fuel is defined as fuel with cladding defects greater than hairline cracks or pinholes [31], (at least in the regulatory context applicable to spent fuel storage and transportation.) NRC Interim Staff Guidance 1 provides qualitative guidance with regard to the classification, and subsequent handling, of spent fuel in three categories: intact (i.e. no defects or defects no greater than hairline cracks or pinholes), damaged (i.e. containing defects greater than hairline cracks or pinholes), and debris (such as fuel rod segments).

Table X. Use of burnup credit: national practices and status in wet and dry storage

COUNTRIES	WET STORAGE					DRY STORAGE				
	PWR	BWR	RBMK	MOX	WWER	PWR	BWR	RBM	MOX	WWER
FRANCE	AP	-	-	UD	-	-	IC	-	IC	-
GERMANY	AP	AP ¹	-	IC	NO	UD	UD	-	IC	IC
HUNGARY	-	-	-	-	IC	-	-	-	-	IC
JAPAN	IC	IC	-	UD	-	IC	IC	-	UD	-
KOREA, REP. OF	AP	-	-	-	-	IC	-	-	-	-
RUSSIAN FED.	-	-	AP ²	-	IC	-	-	-	-	IC
SPAIN	AP	AP ¹	-	-	-	IC	IC	-	-	-
SWEDEN	-	AP ¹	-	IC	-	-	-	-	-	-
UK	RR	UD	-	UD	-	IC	IC	-	IC	IC
USA	AP	AP ¹	-	-	-	RR ³	IC	-	-	-

Note: The Table contains information from countries participating in IAEA meetings on burnup credit implementation and from personal communications.

AP = Approved.

UD = Under Development.

NO = Applicable but not intended.

IC = Interest/Considering, or Applicable.

RR = Regulatory Review.

- = Not Applicable.

Table XI. Use of burnup credit: BUC level allowed in wet and dry storage

COUNTRIES	WET STORAGE					DRY STORAGE				
	PWR	BWR	RBMK	MOX	WWER	PWR	BWR	RBMK	MOX	WWER
FRANCE	A	-	-	-	-	-	-	-	-	-
GERMANY	FP	BA	-	ND	-	A	A	-	ND	ND
HUNGARY	-	-	-	-	A	-	-	-	-	A
JAPAN	ND	ND	-	ND	-	-	-	-	-	-
KOREA, REP. OF	FP	-	-	-	-	-	-	-	-	-
RUSSIAN FED.	-	-	A	-	-	-	-	-	-	-
SPAIN	FP	BA	-	-	-	ND	ND	-	-	-
SWEDEN	-	BA	-	ND	-	-	-	-	-	-
UK	ND	ND	-	ND	-	-	-	-	-	-
USA	FP	BA	-	-	-	ND	ND	-	-	-

A = Reduction of the actinides concentration only considered

FP = Reduction of actinides and build-up of fission products considered

BA = Credit for the presence of integral burnable absorbers

ND = Not decided

¹ Credit for the presence of integral burnable absorbers.

² For Smolenskaja.

³ Use of burnup credit for loading single purpose PWR casks is implemented.

Table XII. Use of burnup credit for spent fuel transportation: national practices and status (as of July 2000)

COUNTRIES	TRANSPORTATION ¹							
	WET				DRY			
	PWR	BWR	MOX	WVER	PWR	BWR	MOX	WVER
FRANCE	AP	IC	IC	IC	AP	IC	IC	IC
GERMANY	-	-	-	-	AP	IC	IC	IC
HUNGARY	-	-	-	-	-	-	-	IC
JAPAN	UD	UD	-	-	-	-	-	-
KOREA, REP. OF	IC	-	-	-	IC	-	-	-
RUSSIAN FED.	-	-	-	AP ²	-	-	-	IC
SPAIN	-	-	-	-	IC	IC	-	-
SWEDEN	-	-	-	-	IC	IC	-	-
UK	RR	UD	UD	-	IC	IC	IC	IC
USA	-	-	-	-	RR	UD	-	-

AP=Approved.

IC=Interest/Considering, or Applicable.

RR = Regulatory Review.

UD = Under Development.

- = Not Applicable.

1 Wet/Dry refers to the assumed condition of the package as transported. Regulatory assumptions for the safety analysis are typically performed wet.

2 For Kola.

Additionally, in some countries, the term “failed fuel” is also used for fuel elements or fuel assemblies having undergone geometric rearrangement or mechanical damage, which has resulted in a change to the fuel element or fuel assembly reactivity or handling capability.

In the context of this document, “failed” fuel will be used to describe any fuel assembly in which:

1. Cladding containment barrier has been breached, or
2. Structural deformations that affect the handling of the spent nuclear fuel (SNF) structure have occurred, or
3. Mechanical defects of the SNF structure have occurred.

It should be noted that a fuel assembly can undergo minor mechanical damage (e.g. grid strap damage), or structural modification, without necessarily being classified as “failed.” Examples of the later include re-built or re-constituted fuel assemblies and consolidated or dismantled fuel. It is also possible for cladding damage, detected during operation (e.g. a micro-crack) to close during post-reactor storage by “autogenous healing”, or through corrosion product growth/deposition over the affected area.

Fuel assembly performance within the reactor core has significantly improved over the past few years. Fuel assemblies are routinely inspected for evidence of structural deformity or damage to either cladding or spacers, in addition to on-line monitoring for fission product release. The fraction of failed fuel rods is currently equivalent to about ten per million [32].

6.1.1. Loss of fuel cladding integrity

Some of the corrective measures that can be taken upon detection of a fuel cladding failure are:

1. Restoration of the primary containment barrier by placing the whole fuel assembly in a bottle, canister or container.
2. Removal, replacement and conditioning of the failed pin(s), followed by further fuel assembly irradiation.
3. Leaving the fuel as-is, and managing the risks in terms of potential activity release.

6.1.2. Loss of fuel assembly lifting features

In cases where a fuel assembly top nozzle has become detached or lifting features have been damaged, but fuel cladding remains intact, one of the following options can be taken:

- Replacement with a surrogate top nozzle,
- Place fuel assembly in a can or sheath; i.e. provision for alternative lifting arrangements and structural integrity.

6.1.3. Geometrical rearrangement of a fuel assembly

Where a fuel assembly has been geometrically rearranged or fuel pins have become detached, but fuel cladding remains intact, one of the following options can be:

- Recovery to a can,
- Customised engineered mechanical constraint,
- Leave in-situ for final disposition at a later time.

It is the responsibility of the operator in charge of spent fuel management, assuming that it is not already prescribed in the terms of the licence, to choose between the options available. The operator will either achieve passive safety by implementing preventive measures, or manage the situation through operational procedures, achieving active safety. In any case, safety objectives related to radiological protection of the personnel and public have to be achieved.

6.2. DETERMINATION OF SPENT FUEL INTEGRITY

6.2.1. At reactor detection systems

All reactors have at least one on-line monitoring system for the detection of in-core fuel failures as a means of controlling the contamination of primary cooling circuits and subsequent dose to personnel during maintenance operations. In addition to the in-core systems, most reactors also have secondary systems associated with defuelling and in-pool operations to check cladding/fuel assembly integrity. The later range from visual inspection [33] and endoscopy [34] to sophisticated Sipping Test Apparatus [35, 36, 37]. Some of the at-reactor systems in routine operation are summarised:

During operation, the ratio of $^{131}\text{I}/^{133}\text{I}$ in the reactor coolant is a rough indicator of fuel leakage. In BWR off-gas and PWR reactor coolant, the ratio of $^{133}\text{Xe}/^{135}\text{Xe}$ also is a useful

indicator of fuel leakage. For example, a ^{133}Xe -detection rate of 2 E7 Bq/s is equivalent to the build-up of about 10g of material in the reactor core.

If there are high levels of fissile material on the fuel assembly surfaces (“tramp”), the activity released from this may obscure or hide the simple ratios mentioned above.

Diffusion analysis is a more sophisticated method to analyse if there are fuel leaks in a background of fission products from tramp uranium. This method is used in BWR as well as PWR. Using five to eight Kr- and Xe-nuclides with half-lives ranging from 3 minutes to 5 days, a simple mathematical model can distinguish the contribution from tramp uranium (recoil distribution) and leaks (diffusion). In a low background of tramp uranium, this method is a good predictor of leaking fuel. The opposite, declaring the core free from leaking fuel, is a more difficult task and must be combined with data from a shutdown transient (see below).

If the caesium isotopes ^{137}Cs and ^{134}Cs are released, the ratio may be used to estimate the burnup of the leaking rod. However, this method should be applied with care, since multiple leaks may yield a mixture of different burnups. This is especially so in the case of BWRs, where there may be significant axial variation of the burnup.

In BWRs, zonal movement of the control rods (also called flux-tilt) at reduced power may allow a very precise identification of the leaking fuel within the core. If there are multiple leaks, or one dominating leak, this method requires great skill in order to achieve unambiguous results.

During cooldown and depressurisation of the reactor coolant, leaking fuel releases a strong burst of activity (noble gases, iodine and caesium). This transient release may be used as a sensitive qualitative indicator of leaking fuel in the core. The absence of such a release is a strong indicator that the core is free of leaking fuel.

Today, sipping is performed in-core in BWRs and RBMKs as well as in PWRs and WWER, during refuelling operations. Both sipping methods (Telescope Sipping, In-mast Sipping) take advantage of the depressurisation of the fuel when it is lifted from its core position, either just for sipping, for unloading, or for shuffling in the core. The depressurisation usually releases a small burst of gas, containing Kr- and Xe- isotopes. The gas is collected and brought to a detector, either beta or gamma sensitive. Such systems are used in France, Germany, Hungary, Russian Federation and Sweden.

Usually, the leaking assemblies are verified by an additional box sipping, where water and gas samples are collected and analysed. The analysis of water samples also enables a comparison of the caesium-ratio from operation and shutdown. This is used to establish if there is more than one leaker in the core.

In **PWRs** the sipping is done during unloading, it requires only about one additional minute per assembly and is done routinely, since the cores are always unloaded.

In **BWRs** partial sipping is common, especially in combination with a previous flux-tilt to define a narrow area in the core where the leaking fuel is.

In both reactor types, the presence of high levels of tramp material in the core makes it difficult to detect pin-hole leakers, as they can be masked by the fission product release from the tramp material.

In Swedish facilities, leaking assemblies are inspected visually to identify the position of the leak. Usually the leaking rod is removed and replaced by a dummy or one of similar burnup, to allow further operation of the fuel. If visual inspection is inadequate, Eddy Current testing of each rod is performed. Of course, this operation requires the dismantling of the assembly. After the repair, the assembly is sipped by the Box Sipping method to verify that it does not leak any longer. In this situation, caesium isotopes (or ^{85}Kr) are the only nuclides available for analysis.

In Korean PWR reactors, underwater TV cameras, sipping test equipment and failed fuel rod detection systems (FFRDS) are used for the identification of failed fuel. When an increase in fission gas release is indicated, by continuous monitoring of the coolant activity, discharged fuel assemblies are examined using an underwater TV camera.

A failed fuel rod detection system using ultrasonic test technology is employed during the refuelling period at each PWR plant. After the performance tests, the FFRDS was proved to be satisfactory. The integrity of spent fuel rods in AR pool storage has been demonstrated by routine monitoring for over 20 years. No significant problems due to the spent fuel degradation in the pool have been encountered and there are no symptoms of fuel cladding or storage component corrosion.

RBMK-1000 reactors monitor fuel cladding integrity through continuous gamma monitoring of the steam supply pipes connected to each reactor channel. The gamma intensity between channels is compared to establish whether a particular stringer has failed. Additionally, fuel assemblies are checked during reload within the reloading machine by gamma activity measurements; if a reading $>20\text{mR/hr}$ is observed the assembly is considered to be failed [38].

WWER reactors, in the Eastern European countries, have a failed fuel detection system supplied with the reactor. The system comprises four sealed tubes for liquid sampling located on the fuel discharge route between the reactor and the spent fuel pool. During refuelling an assembly is placed in one of the tubes and water is circulated inside the tube for 30 minutes. After this time the liquid is sampled for ^{131}I , ^{133}I , ^{134}Cs , and ^{137}Cs isotopes. If the count is above a set value, the fuel is declared as failed and is placed in a sealed bottle within the spent fuel pool.

Some of the countries using Russian-designed reactors (e.g. Hungary) have retrofitted an additional sipping system, similar to those that operate at most Western reactors measuring significant Cs or I activity increases during refuelling operations.

The on-line sampling capability used in **Magnox** reactors analyses for beta decay of precipitated noble gas daughter products. The system, an electrostatic precipitator, otherwise known as the "Burst Cartridge Detection System" (BCD), is coupled to each reactor channel or groups of channels. The technique, only applicable for channel reactors, works by electrostatically precipitating onto a wire the charged daughter products of noble gas nuclides that decay within the precipitation chamber. The wire is driven into a beta detector that detects beta decay of the precipitated daughters. In the case of **advanced gas cooled reactors**, the BCD system is only brought into play, when a failure has been detected, by the high-level reactor gas activity monitor (GAM), in one of reactor quadrants. GAM is a multi-

channel gamma spectrometer that automatically determines the concentrations of individual noble gas nuclides in the reactor coolant. The capability also exists for individual channel sampling for Iodine to be carried-out.

In **CANDU** reactors, the iodine content in the primary heat transport (PHT) system is monitored to provide an indication of failed fuel in the core. Some reactors are equipped with a delayed-neutron detection system that identifies the channel containing failed fuel. However, a leaking fuel bundle is detected by on-line measurement of the increase in iodine concentration during the transfer of the fuel bundle from the fuelling machine to the reactor storage pool. In Canada, fuel bundles identified as containing failed fuel elements are segregated for assessment and not knowingly loaded in either dry storage containers or storage baskets.

6.2.2. Away From Reactor detection systems

Away from reactor detection systems are geared to undertake one of three roles:

1. Confirmation of shippers data on receipt.
2. Confirmation of continued fuel integrity during storage.
3. Confirmation of fuel assembly structural integrity prior to fuel retrieval.

At Swedish facilities, sipping tests are performed on casks prior to shipment to the CLAB storage facility. When the transport casks with the spent nuclear fuel arrive at CLAB, the vacuum is broken and the assemblies are cooled down by injection of water into the cask. The presence of any leaking fuel elements is evaluated by beta sensitive detectors, measuring ^{85}Kr in the gas, during the cask cool down procedure. If any shipment has indications of leaking fuel, it is completely sipped in a box sipping equipment.

In the majority of cases AFR facilities have a set of acceptance criteria, which consignors have to work to. As such, the service provider tends to rely on the quality of reactor operators or shippers data, and no routine checks are undertaken on receipt. In some cases random sampling, for example flask liquor or gas sampling, may be carried-out to confirm the fuel has not failed during transport; this is likely to be undertaken when there is a significant change in the burnup of the fuel being shipped. At Cogema's (TO) dry receipt facility, which is equipped with a krypton analyser, routine sampling of individual fuel assemblies is done prior to placing the fuel in storage.

The need to monitor the condition of fuel assemblies during storage can be a licence condition for the storage facility in some countries. In wet storage an early indication of bulk fuel cladding failures or loss of ideal storage conditions will be picked up through routine pool water sampling. In the case of dry storage systems, apart from MVDS types where individual or bulk channels can be analysed for krypton release by mobile sampling units, there is no routine capability for in-situ fuel integrity monitoring. In addition to daily/periodic sampling schedules, further proof of fuel assembly integrity may be obtained through post storage examination in hot cells of a sample of the fuel being stored.

When fuel retrieval is required, especially in cases where storage duration has been prolonged, visual inspections of tie rods, top nozzle/guide tube interfaces and handling features is carried-out using closed circuit television (CCTV) to confirm fuel assembly integrity.

6.3. CRITERIA

6.3.1. Utilities

All utilities have a threshold or cut-off limit in terms of acceptable cooling circuit contamination levels that can be tolerated before the reactor needs to be shut down. These threshold levels vary from one country to the next. For example, EDF will operate up to 1 000 000 MBq/t total activity or 40 000 MBq/t ¹³¹I before the reactor is shut down, and the Hungarian trigger level for possible leakers is 370 MBq/t total iodine activity (3700 MBq/t suggests serious fuel failure).

6.3.2. Transportation

Even though transport casks are designed to accommodate fuel assemblies with up to 100% defected pins, the individual transport authorities, the cask owner and the receiving AFR are unlikely to approve such shipments on dose grounds. In the majority of cases, known leaking fuel assemblies are required to be conditioned prior to shipment, or there will be a limit set by the appropriate transport authority on an acceptable level of activity within the cask body.

6.3.3. AFR facilities

AFR facilities generally have an acceptance criteria for failed fuel in terms of quantities and packaging required; in some instances this will be governed by the AFR licence.

7. FUEL ASSEMBLY DEGRADATION MECHANISMS IN DRY AND WET STORAGE

The significant physical and chemical processes that may affect the integrity of the spent fuel structure in interim storage are reviewed in Section 7.1. A detailed discussion of the most important degradation mechanisms affecting the spent fuel cladding under storage conditions is presented in Section 7.2.

7.1. DEGRADATION MECHANISMS OF FUEL ASSEMBLY COMPONENTS

Maintaining fuel assembly structural integrity is essential to ensure the subsequent safe handling of the spent nuclear fuel. There are two important factors to consider:

1. Maintenance of the original fuel bundle geometry, to avoid variations in fuel reactivity and to ensure that subcriticality is maintained.
2. Maintenance of fuel the bundle structural integrity to avoid fuel drops during handling, or nozzle separation.

7.1.1. LWR

LWR fuel assemblies comprise fuel rods and numerous structural components. Some modern fuel assemblies used in the European market are displayed in Figure 4.

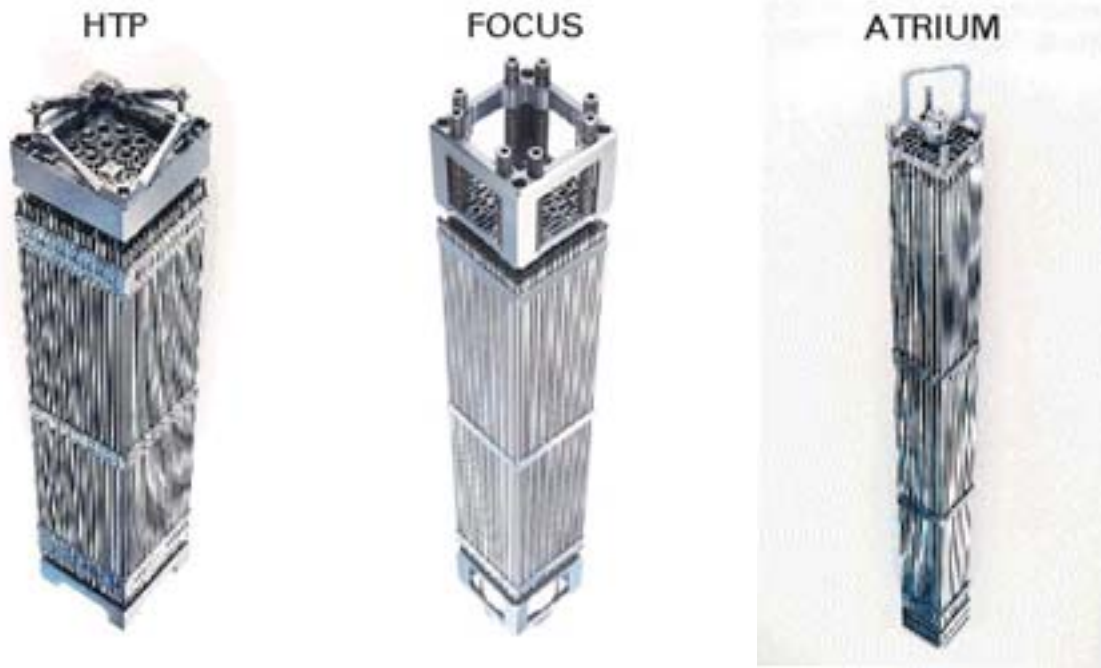


FIG. 4. Fuel assemblies for BWR and PWR.

All PWR and modern BWR fuel assemblies contain the fuel rods in a skeleton structure. Older BWR fuel assemblies have tie rods, containing fuel like a normal fuel rod, joining together the top and bottom ends of the fuel assembly structure. The different structural components of the fuel assembly, like spacers, guide tubes, top and bottom end nozzles, tie plates and fuel channels are manufactured mainly from Zircaloy or other zirconium alloys, stainless steel, and Inconel. At the end of the fuel assembly service life, all materials are passivated by oxidative corrosion.

In wet storage, electrochemical corrosion between the different materials can be excluded, as well as oxidative corrosion in the pool water. The wet storage performance of the LWR fuel structure is, in general, excellent. In a few cases, involving non-stabilised stainless steels, sleeve corrosion has been reported for PWR fuel assemblies with Zircaloy guide tubes and stainless steel upper tie plates. This corrosion phenomenon is related to the non-stabilised stainless steel components with a higher carbon content. This effect can negatively impact spent fuel handling at the back end of the fuel cycle when chloride ions ingress into the pool water has occurred.

In dry storage of the spent LWR fuel, no degradation mechanisms of the fuel assembly structure are known to be of concern and no detrimental experience has been reported.

7.1.2. AGR

The main components of the AGR fuel assembly comprise 36 fuel pins held in a graphite sleeve by a bottom grid plate and two braces along the length of the assembly (Figure 5). Except for the outer graphite sleeve, all other components are made from highly alloyed stainless steel. In-reactor, up to seven AGR fuel assemblies are joined together by a stainless steel tie bar passed through the centre of each assembly to form a fuel stringer ~7 m in length.

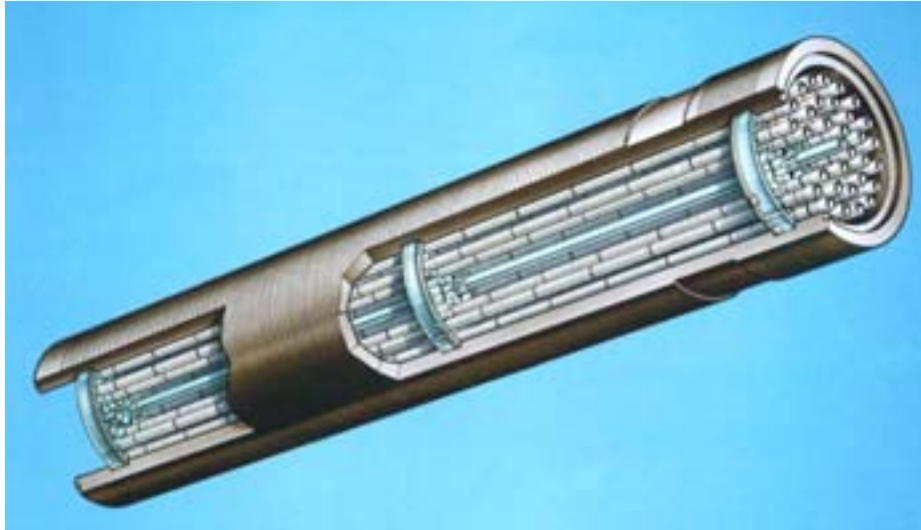


FIG. 5. AGR fuel assembly.

In spite of the AGR being a gas cooled reactor, the spent fuel is stored wet. The precautions used to ensure a satisfactory fuel rod performance in wet storage also ensure the stability of the AGR fuel element structure.

7.1.3. Magnox

The name Magnox originates from the magnesium alloy can, encasing the metal fuel rod. Magnox reactors are named after the Magnox-A 180 alloy developed to achieve low neutron capture cross-section and resistance to CO₂-coolant corrosion. The alloy also has good ductility and creep strength. The designs of Magnox fuel elements, although similar, are tailored for each reactor. Each element basically comprises an inner natural uranium rod machined with groves to reflect its core position, and one of either two types of outer hollow extruded and machined magnesium can (polygonal or ‘herringbone’); a cone end fitting, ceramic insulating discs and machined end cap complete the element. The interspace is charged with a small amount of helium prior to the outer can being locked into the groves in the uranium rod by externally pressurizing the sealed element. In the reactor, the elements are placed on top of one another forming stringers within the graphite channel. The purpose of machining groves into the natural uranium rods and locking the external cans into them is to minimize localized stressing of the cladding during irradiation (See Figure 6).

The end cap remains to be the only structural component of MAGNOX fuel. Since it is machined from magnesium alloy, it has the same degradation mechanisms as described in Section 7.2.3.

7.1.4. CANDU

The fuel bundles used in modern CANDU reactors are composed of either 28 or 37 fuel rods, held together in a cylindrical array by welds that attach the end caps of each fuel rod to two Zircaloy endplates. Ontario Power Generation’s eight Pickering nuclear reactors use 28-rod bundles, while the eight Bruce and four Darlington reactors use 37-rod bundles. CANDU-6 reactors also use 37-rod bundles.

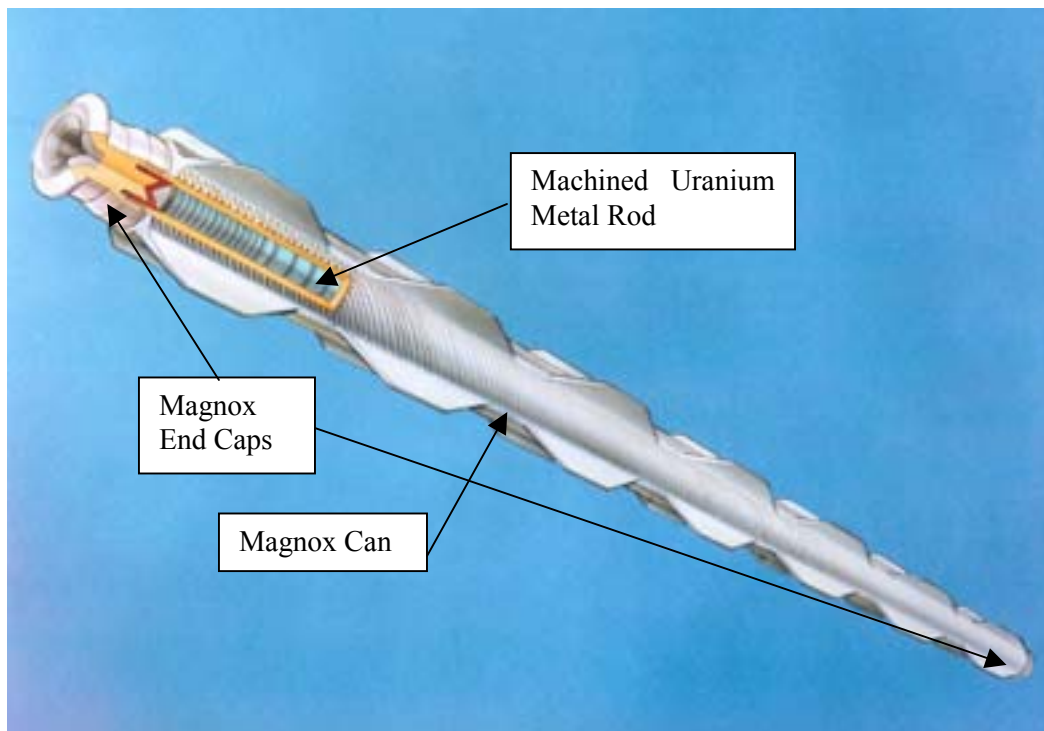


FIG. 6. Magnox fuel.

Each fuel rod consists of a Zircaloy-4 tube containing a string of UO_2 pellets. Zircaloy-4 endcaps are resistance-welded to each end of the tube in a He atmosphere to seal the pellets within the tube. The overall length of the fuel bundles is approximately 50 cm. Depending upon the location of the fuel rods within the bundle assembly, either spacers or bearing pads are brazed to the outside surface of the cladding. The spacers are designed to maintain coolant flow around the rods while the bearing pads protect the cladding from wear due to friction with the pressure tube during reactor operation.

Zircaloy-4 cladding is fabricated by cold-extruding Zircaloy tube shells. Several cold-working and annealing stages are required to obtain a microstructure and crystallographic texture that give the required combination of strength and ductility. The cladding texture and microstructure are designed such that any hydride platelets that may form in the cladding will lie predominantly in the circumferential-axial plane. Hydrides with this orientation are the most innocuous with respect to delayed hydride cracking or stress fracture of the cladding.

After final sizing, the outside diameter of the cladding is 15.19 mm for 28-rod fuel bundles and 13.08 mm for 37-rod fuel bundles. The cladding wall thickness used for all fuel rods is 0.38 mm.

Zr-5Be brazements are used for attaching the Zircaloy spacers and bearing pads to the Zircaloy fuel sheath. The induction brazing process used for this operation heats the sheath surrounding the appendages into the beta phase region of the phase diagram. This causes changes in its metallurgical structure and mechanical properties. A variety of structures are produced by the heating, ranging from a fully transformed beta structure near the appendages to short-term annealed or recrystallized structures near the edges of the heated zones. The large transformed beta grains tend to lower the strength and ductility compared with that in as-received tubing. However, no greater tendency is found for defects to occur in the brazing.

7.1.5. RBMK and WWER-440/1000

Table XIII presents the main characteristics of fuel assemblies from Russian power reactors.

The RBMK-1000 FAs differ from the WWER FAs by their size (about 10 m). The assembly consists of two fuel bundles connected by a tie rod (suspension). Each bundle comprises 18 fuel rods held in a skeleton structure by 9 spacer grids and a central tube. Today all components are fabricated from Zr1Nb alloy. The fuel is uranium dioxide with an enrichment of up to 2.6% ²³⁵U and all fuel rods contain erbium as absorber with a content of about 0.41%.

Degradation mechanisms

Based on operational data of more than 30 WWER reactors, it can be concluded that WWER fuel does not have any identified generic degradation mechanism. The very few events observed in the SPAR countries can all be attributed to special cases.

Table XIII. Main characteristics of Russian fuel assemblies

Characteristic	Reactor		
	WWER-440	WWER-1000	RBMK-1000
Outer diameter of fuel rod, mm	9.1	9.1	13.6
Fuel rod cladding thickness, mm	0.65	0.67	0.90
Material of cladding	Zr1Nb	Zr1Nb	Zr1Nb
Material of spacer grid	SS	SS	Zr2.5Nb
Structural material (tubes at FA centre or corner)	Zr1Nb	Zr1Nb	Zr2.5Nb
Burnup			
– Present	40–43	45–49	24
– Planned	45–50	55	35

7.2. DEGRADATION MECHANISMS AFFECTING THE FUEL CLADDING

Maintaining fuel cladding integrity is important because the cladding acts as the primary containment barrier for both the fuel pellets and the products of fission over the storage period and during subsequent downstream fuel handling operations. This is valid for "unfailed" fuel assemblies used in all the types of power reactors considered: LWR, AGR, Magnox and HWR. For CANDU and old BWR fuel, cladding integrity may also be important for fuel assembly integrity, because some of the fuel pins also act as structural elements for the whole fuel assembly.

7.2.1. LWR

Apart from a limited number of early fuel designs, all current LWR fuel rod claddings are zirconium-based alloys. In the Western countries, Zircaloy-2 and Zircaloy-4 are commonly used, whereas in Eastern European countries Zr-Nb-alloys are the preferred materials. It is observed, however, that the use of Zr-Nb-alloys is increasing in Western countries, because of their excellent corrosion behaviour during reactor operation.

7.2.2. Wet storage

7.2.2.1. Mechanisms

The degradation mechanisms that may affect cladding integrity during wet storage are as follows:

1. Mechanical stresses in the cladding.
2. Uniform (aqueous) corrosion.
3. Pitting, galvanic, and microbially-influenced corrosion.
4. Hydriding.

Mechanical stresses

The most important contribution to the mechanical stress imposed on the cladding tubing during long-term storage results from the internal gas pressure of the fuel rods. The internal gas pressure is composed of the He fill gas- and the fission gases released during in-service operation. For most of the fuel, even for higher burnup fuels, the hoop stress in the cladding is well below 80 N/mm². The yield strength ($\sigma_{0.2}$) of the cladding is over 400 N/mm², thus yielding a satisfactory safety margin. Therefore, processes straining the cladding can be ruled out at the relevant temperature and stresses of interest.

Oxidative corrosion

The corrosion behaviour of zirconium alloys in pressurised water and water vapour is well known for temperatures above 200°C. Investigations at other temperatures show that the temperature dependence of the corrosion rate follows the Arrhenius law in the linear, post transition corrosion regime. If the corrosion rate under normal storage conditions at 30°C to 45°C is compared to the corrosion rate under operational reactor conditions, the result shows that the corrosion rate in the pool water is a factor of 10¹⁰ lower. Putting this in context, the corrosion resulting from one full power day is equivalent to the amount that would occur over a period of several millions years in storage. This leads to the conclusion that zirconium alloys are virtually immune to (uniform) oxidative corrosion under wet storage conditions. However, this only holds for de-mineralised water of well-controlled chemistry.

Electrochemical corrosion

Zirconium alloys are near the noble end of the galvanic series in contrast to the aluminium alloys, which are near the active end of the series. Electrochemical contact between Zircaloy and Al would generate enough potential to oxidise the Al and hydride the Zircaloy. The effects of electrochemical interactions between Zircaloy and Al, however, are mitigated by the passivation effect of the oxide layer on Zircaloy surfaces, which acts as a barrier to hydrogen uptake. Therefore, for spent fuel, direct contact between bare Zircaloy surfaces and Al does not occur in the pool. Passivated stainless steels are also near the noble end of the galvanic series. Therefore electrochemical corrosion has never been observed when direct contact of both Zircaloy and stainless steel occurs in nuclear fuels, either in-reactor or in wet storage.

Hydriding

The average hydrogen concentration in zirconium alloy claddings at the end of in-reactor operation is between 250 and 550 ppm, depending on the material and the burnup achieved. When the fuel is cooled to the storage pool temperature, most of the hydrogen precipitates in

the form of hydride platelets. The formation of so-called pits by an Oswald ripening cannot be expected at the water temperature of interest to long-term storage. Due to the small heat up in the fuel bundle in water and the low temperature, H redistribution by thermal diffusion can be ruled out.

Cladding integrity-related properties such as yield stress, rupture stress, and uniform elongation depend on the hydrogen concentration in the cladding tube. Yield stress and rupture stress increase, and ductility decreases with increased hydrogen content in the cladding. The long-term storage temperature lies below the ductile/brittle transition temperature of hydrided Zircalloys. This means that the cladding will withstand elastic deformation up to the raised yield stresses. Plastic deformation under those conditions would be negligible. However plastic deformation does not occur during handling of the fuel assembly, since bowing of the fuel rods is mechanically prevented by the fuel assembly structure. Therefore, if the fuel assemblies are handled and stored under normal conditions, the hydrogen in the cladding does not exert any negative influence on the handling and storage of the spent LWR fuel.

7.2.3. Dry storage

7.2.3.1. Storage modes and mechanisms

For the purpose of assessing the performance of spent LWR-fuel in extended storage, three different temperature regimes can be defined, based on available experience:

1. **Mode I:** fast rate of decrease in fuel temperature, usually occurring between maximum dry storage temperature and 300°C.
2. **Mode II:** medium rate of decrease in fuel temperature occurring between 300°C and 200°C.
3. **Mode III:** slow rate of decrease in fuel temperature occurring below 200°C.

Potential mechanisms that may affect the cladding integrity of LWR fuels during dry storage are:

1. Creep.
2. Cladding oxidation.
3. Mechanical crack propagation and uniform and localised fission product corrosion.
4. Hydrogen-induced defects (delayed hydride cracking, hydrogen diffusion in thermal gradients, embrittlement, etc.).

A graphic presentation of these effects can be seen on Figure 7 [39].

Creep

Thermal creep is generally considered to be the limiting degradation mechanism for evaluating cladding integrity during dry storage. At temperatures between 300 and 400°C (Mode I), which prevail at the start of dry storage, the cladding undergoes strain due to creep. Creep strain is largely determined by the fuel rod internal pressure and fuel rod temperature-time history over the course of dry storage. If creep were allowed to proceed to rupture, it would most likely lead to a pinhole type of rupture. Nevertheless, it is desirable to avoid any loss of cladding integrity. This can be accomplished by confining creep deformation to its primary and secondary regimes.

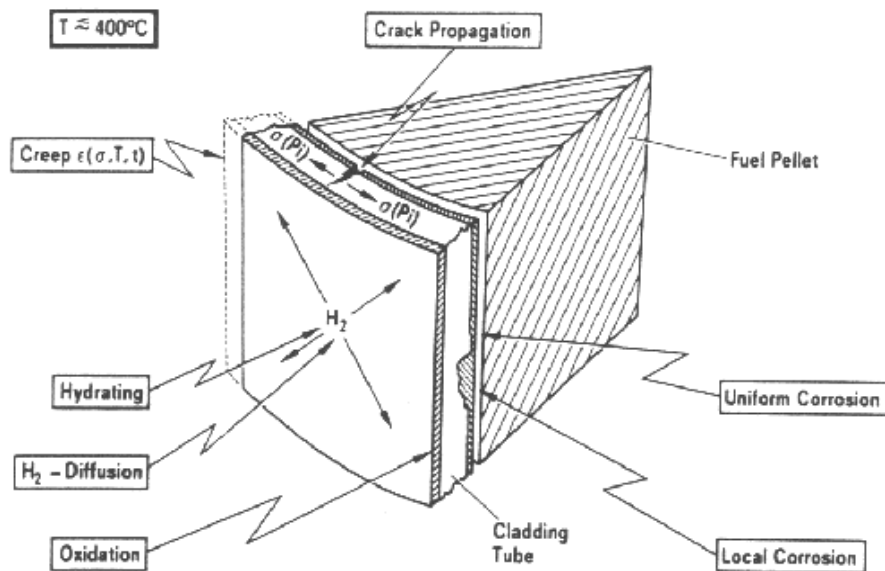


FIG. 7. Mechanisms that may affect spent fuel cladding performance during dry storage.

If it can be shown that, during dry storage, the creep strain does not exceed the critical strain domain leading to plastic instability, tertiary creep with its subsequent fuel rod deflection can be excluded. This approach is in compliance with standard creep engineering practices.

Implementation of this approach in Germany led to a methodology that relies on the following considerations:

1. Availability of an experimental database of creep test results of unirradiated cladding obtained under internal pressurisation conditions, which led to a numerical model to predict creep strain [40].
2. Availability of a correlation that allows the prediction of post-pile creep from creep of unirradiated material. As can be shown by comparison of the creep of unirradiated and irradiated Zircaloy, the creep for unirradiated material always describes the post-pile creep conservatively (Figs 8–9).
3. Availability of a database of the total *strain capability* of fuel rod cladding under dry storage conditions. For a burnup less than 40 GW·d/t HM, data from post-pile burst test indicate that all cladding will reach about 1% uniform strain before rupture. Because the rupture strain capability of the cladding is expected to decrease with burnup, the very conservative measurements provided by burst testing with its very high strain rates was replaced by accelerated creep testing with much lower strain rates. Siemens performed such a test program under contract to GNB, providing the spent fuel with a burnup up to 64 GW·d/t HM [41] from a Siemens high performance programme. This programme provided the basis for the present German licence to store spent U-FA and MOX-FA with an average batch burnup up to 55 GW·d/t HM (Fig. 10.)

Oxidative corrosion

Normal storage under inert atmosphere conditions and appropriate drying procedures rule out the presence of oxidising substances. Therefore, dry storage results in no further increase in oxide layer thickness over and above the condition existing at final discharge from the reactor.

In addition, storage systems include provisions to detect and correct abnormal conditions within a short time. Therefore, Zircaloy oxidation is not a limiting factor for cladding degradation during normal dry storage operation of LWR fuel.

Access of air is possible if the sealing system of the dry storage system fails, an unlikely event for existing system designs. If the stored spent fuel is free of defects, no alteration of the UO_2 -fuel will occur, because the Zircaloy-cladding can easily withstand the oxidative corrosion at the temperature levels expected under the storage Mode II and Mode III. There is also experimental evidence [42], that oxidation will not result in cladding failures for the temperatures expected under Mode I. In the range of 400°C to 200°C , both cladding and fuel oxidation rates decrease by an order of magnitude for every $\sim 10\%$ drop in temperature as expressed in $^\circ\text{C}$.

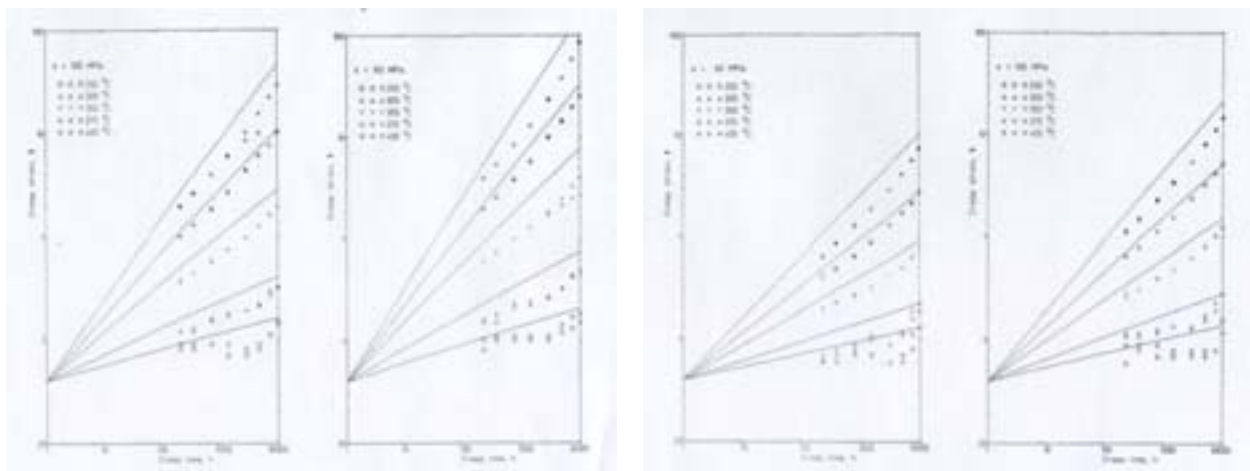


FIG. 8. Experimental creep data and calculated values using the creep equations derived from the experimental database.

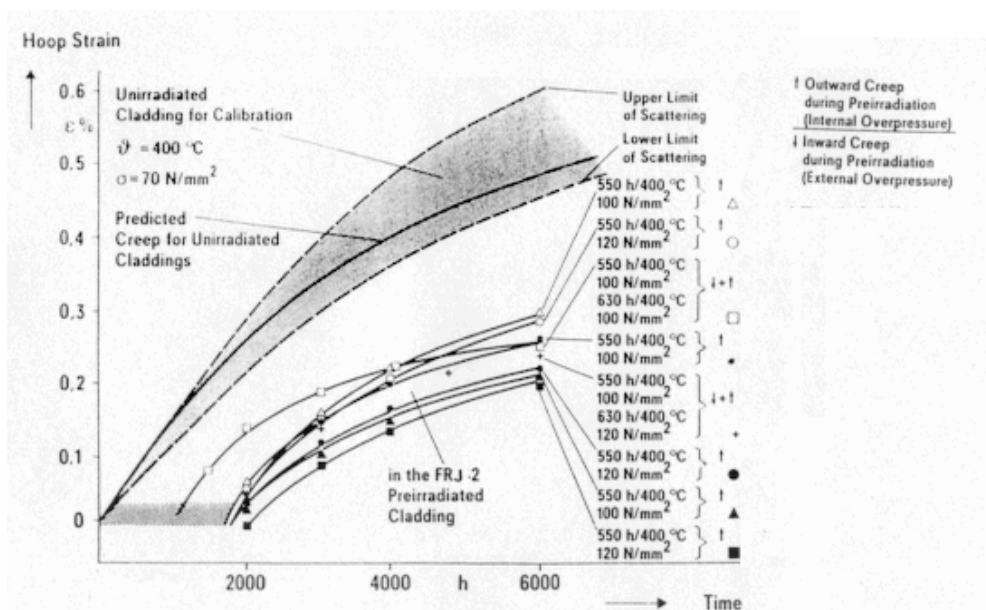


FIG. 9. Comparison of creep strain of irradiated and unirradiated Zry-4-cladding.

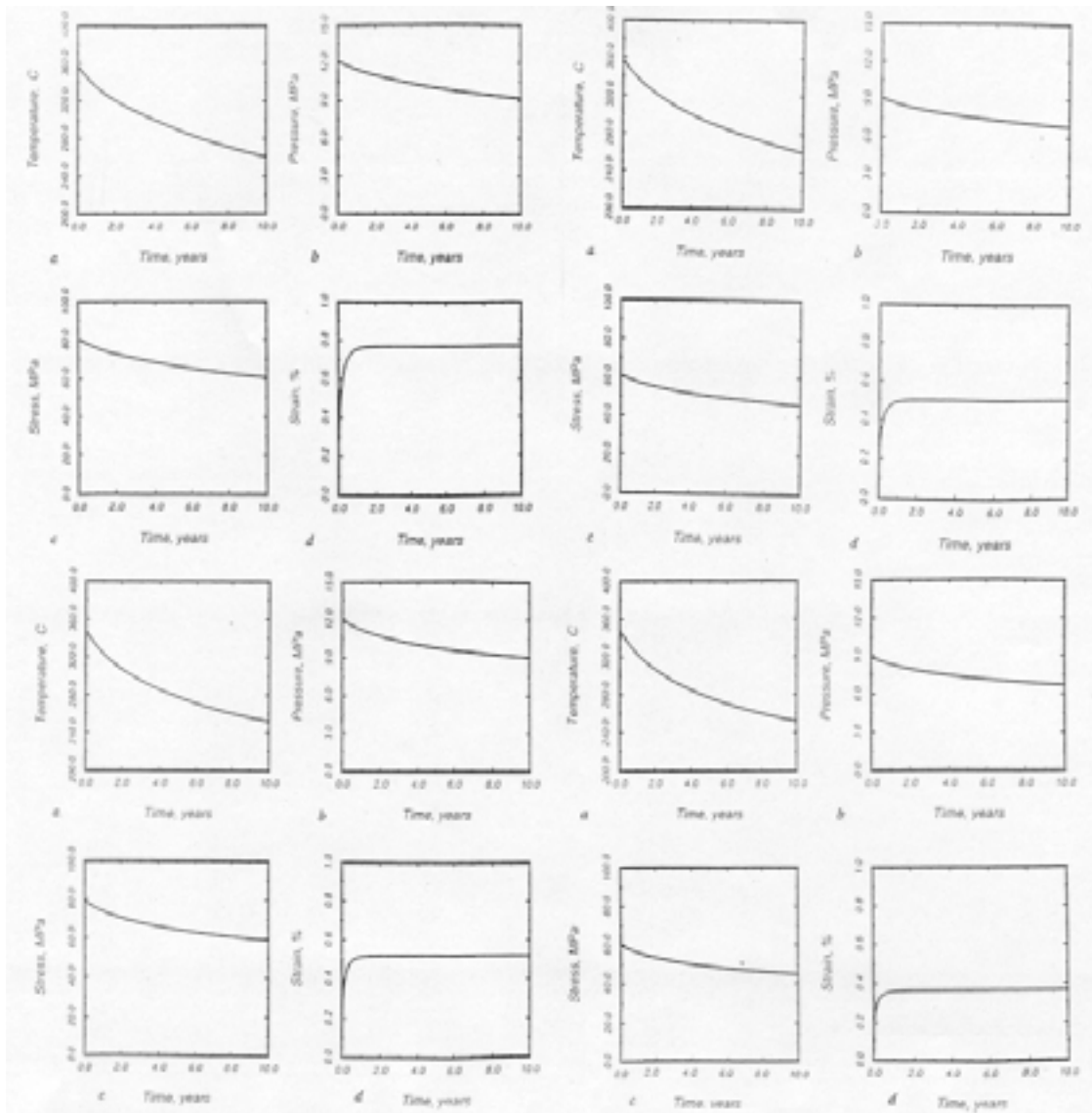


FIG. 10. Spent PWR MOX-FR (top)/UO₂-FR (bottom), predicted strain in dry storage, 55GW-d/tHM (left: strong cladding, right: fast creeping cladding).

In case of defective fuel rods, no fission product releases are expected under Mode III, because the UO₂ oxidation rates are very small below 200°C. When the fuel does not oxidise, no restructuring takes place, and hence no fission product release is expected. If the temperature exceeds 200°C (Mode I and Mode II), UO₂ conversion to U₃O₈ will occur, and as a result ⁸⁵Kr will be released.

Fission-product-induced cladding corrosion

Iodine-induced stress corrosion cracking (SCC) occurs only within a specific high temperature range, in the presence of chemically active iodine and adequate stresses. In spent fuel dry storage, iodine is not present in a form that could trigger SCC. As the combination of SCC agent and the stress conditions required for crack propagation are normally absent, it can, therefore, be concluded that cladding failure via this mechanism is not expected to occur.

Fission product behaviour can be characterised by a high stability against release. UO₂-fuel crystallises in a CaF₂-type lattice and contains a plurality of vacancies, which aid in the

retention of fission products generated by reactor operation. Also, each fissioned U-atom leaves an additional vacancy in the lattice. As a result, even in high burnup fuels, not all vacancies are occupied, and the UO₂-crystal has sufficient capability to retain the fission products. A complete discussion of this topic is available in Ref. [39].

In summary, the fission products generated in the UO₂-fuel under in-service conditions are practically immobile in the UO₂-fuel lattice during storage. Consequently all fission-product-driven defect mechanisms such as stress corrosion cracking, uniform fuel rod internal fission product corrosion of the cladding, and localised fuel rod internal fission product corrosion of the cladding are not active.

Delayed hydride cracking

If the hydrogen content of a Zircaloy cladding exceeds its solubility (~160 ppm at 400°C), the hydrogen precipitates in the form of Zr-hydride platelets. Fuel rods with a burnup exceeding 50 GW·d/t HM usually have greater hydrogen contents than lesser burnup fuels. The hydride platelets reduce the ductility of the cladding, when their orientation is perpendicular to the applied stress, and this may induce crack propagation in the cladding. The cladding failure mechanism referred to as delayed hydride cracking (DHC) is described below.

Crack initiation: Incipient cracks at the inner cladding surface may result in DHC, if hydrogen precipitates at the crack tip, which causes a decrease of the critical stress intensity factor. Radially orientated hydrogen precipitates may serve as incipient cracks, only if the temperature of the cladding is less than ~160°C, which corresponds to the ductile-brittle transition for the precipitated zirconium hydride platelets.

Hydride re-orientation: The cladding tube manufacturing process generates a texture in the cladding that causes the hydride-platelets to precipitate in a predominantly tangential orientation. Re-orientation of hydride platelets has the potential occur under specific temperature and stress conditions. For the grain sizes typical of LWR cladding, those temperature and stress conditions are: at 400°C; $\sigma = 120 - 180 \text{ N/mm}^2$, and at 250°C; $\sigma = 250 - 350 \text{ N/mm}^2$ [43]. Such stresses in the cladding are generally precluded under dry storage conditions, especially in Germany, through fuel licensing limitations.

Critical flaw size: Based on Canadian results [43, 44, 45], the critical crack size can be calculated by the following equation:

$$A = C \times (K_{IH}/\sigma)^2 / \mu\text{m/}$$

with $C=0,4$

This equation can be used to generate the following data shown in Table XIV:

Table XIV. Critical crack depth for delayed hydrogen cracking

T, °C	K _{IH} , MNm ^{-3/2}	σ, MN/m ²	Critical crack depth
150	5 (irradiated)	120	694 μm

The stress intensity factor for the unirradiated material results from experimental investigations. The value for the irradiated material status is assumed to be only half of that from unirradiated material. As a result, the critical crack depth is 694 μm . Neither incipient cracks from such depth, nor re-oriented hydride-platelets have been observed in spent fuel at the end-of-life condition, or after dry storage.

Altogether it can be concluded that delayed hydrogen cracking will not occur to any appreciable extent in dry storage, even for the hydrogen contents typical of higher burnup fuels. This conclusion is in agreement with the limited available experience that DHC has not been observed worldwide in LWR cladding.

Creep versus cavitation damage

The diffusion-controlled cavity growth (DCCG) mechanism was adopted by the Lawrence Livermore National Laboratory, and implemented by the NRC, to set peak cladding temperature during dry storage. It is also being used in the Pacific Northwest National Laboratories' Commercial Spent Fuel Management model (CSFM) as a predictive tool for creep-rupture [46]. However, cladding rupture from this mechanism is a sudden, non-ductile type of failure with no early manifestation of the damage, and does not accurately identify a creep type failure. By definition, the latter is a macroscopic phenomenon that does not progress without external evidence of damage (creep strain).

The cavitation model is very sensitive to the input data, and the model needs a great number of input parameters, which are not at all well known. Small variations of these inputs result in large variations of the conditions under which the spent fuel cladding would fail. Selecting the input data in a conservative manner leads to model forecasts predicting fuel rod failures under conditions, where the overall experience reports safe and reliable dry storage performance.

Burnup increase

Higher burnup results in higher decay heat and increased fission gas release. This will generate higher stresses for longer periods of time in the fuel rod cladding. In addition, the longer residence time of a fuel assembly in the core to achieve the higher burnup tends to reduce the residual wall thickness through waterside corrosion. Therefore, stress is further increased. MOX fuel already exhibits this at the same burnup as U fuel: increased decay heat and increased fuel rod internal pressure. High-burnup and MOX fuels will, therefore, experience higher stress and strain under dry storage conditions.

Pressure increase due to He-generation

For very long storage periods (hundreds of years), the production of helium from alpha decay in MOX fuel becomes comparable to the amount of fission gases produced during reactor irradiation. Calculations show concentrations of up to 0.2 g/kg U after 200 years of storage. Helium has a smaller atomic radius than the fission gases and therefore diffuses much faster. The diffusion coefficient of He is reported to be far below 10^{-18} cm^2/s under temperature conditions typical of dry storage [47, 48]. Assuming a diffusion coefficient of only 10^{-18} cm^2/s , the typical diffusion length is about 1/10 of the size of a fuel's grain after 100 years of storage. Thus, helium release via diffusion to the grain surface is negligible.

7.2.4. AGR

The cladding material used in the manufacture of AGR fuel pins is 20Cr/25Ni, which is a Nb-stabilised stainless steel. During reactor operation, stainless steel components in the 340–520°C temperature range (peak at 420°C) become sensitised, primarily as a result of radiation induced segregation. The cladding of elements (1–3) of the seven stack (usual) fuel stringer, are affected by this mechanism. Sensitisation, coupled with stress induced at the anti-stacking grooves of the fuel pin, can result in cladding failures for pins stored in moist air conditions or in pond storage environments (with high chloride ion concentrations); through the failure mechanism known as irradiation induced intergranular stress corrosion cracking (IGA). This has implications for potential air ingress at reactor shut downs, initial cooling/storage at reactor in dry buffer storage tubes and interim storage ponds.

For short to medium term fuel storage durations, IGA is the prominent defect mechanism for AGR fuel cladding. Unlike Zr-based alloys, if the storage durations become prolonged (i.e. hundreds of years) general waterside corrosion becomes a significant factor, especially where initial in-reactor cladding wall thinning has been significant. This failure mechanism also affects the fuel element stainless steel structural materials (braces etc.), in this case it is most dominant in the materials of fuel elements 3–5 of the stringer.

Sensitisation of stainless steel is a widely recognised and researched topic. It is caused by the free carbon in the alloy, reacting with the chromium-alloy, to form chromium carbides on the metal grain boundaries. The formation of chromium rich carbides on the grain boundaries denudes them of the chromium alloying additions and the stainless steel considered becomes sensitised to localised grain boundary corrosion in environments, where the bulk material is passive. Thus the cladding becomes susceptible to intergranular corrosion attack. To limit chromium carbide precipitation in AGR cladding, the final heat treatment was reduced to 930°C. Secondary effects, such as carbon pick-up from the coolant and radiation induced segregation, however, have been found to be the over-riding factors in AGR cladding sensitisation.

The problem of IGA initiation, during spent AGR fuel wet storage, has been inhibited by caustic dosing to pH11.4. Inhibitor levels are a function of both chloride ion content and pool water temperature; pH11.4 represents an optimised level for pool storage conditions at Sellafield. Although the long-term use of sodium hydroxide as an inhibitor was originally thought to have implications, through dissolution of the protective oxide film on the steel with a resultant increase in general waterside corrosion, recent corrosion studies have not seen any evidence of accelerated general corrosion [49].

7.2.5. Magnox

7.2.5.1. Wet storage

In wet storage MAGNOX-cladding corrodes according to the reaction with water, forming $Mg(OH)_2$ and hydrogen. In principle if the pH is increased to greater than 11.5, the solubility of the magnesium hydroxide layer is low, effectively forming a passive surface film on the metal. Research work, however, shows that this is not the full story and other factors, such as micro-cracking and the presence of carbon in the oxide layer, result in corrosion rates being up to three times higher than unirradiated material. Cladding corrosion is also influenced by the presence of water impurities, notably chloride, which leads to pitted areas even at low concentrations. To minimise these effects chloride and sulphate ion levels are maintained at levels <1ppm in pond purge water. Pitting corrosion may cause failure of the cladding during

longer storage. Crevice corrosion between the splitter blades and the cladding might also occur. For the reasons outlined above, MAGNOX spent fuel is usually reprocessed within two years of reactor discharge, however, current storage practices show good performance and storage periods >5 years should be expected.

7.2.5.2. Dry storage

Since the cladding was developed for CO₂-cooling during reactor operation, dry storage under CO₂ is suitable for long term storage. Corrosion rates in CO₂ are negligible for temperatures below 350°C. Storage in air is feasible when the fuel temperature is below 150°C; the safety case contains restriction due uranium hydride ignition if bare metal was present. In the case of dry Magnox fuel stored in dry air, fuel deterioration will occur through pitting attack, if relative humidity is >50%, or there is water ingress.

7.2.6. CANDU fuel degradation under dry storage conditions

7.2.6.1. Cladding degradation mechanisms

The potential mechanisms of concern for degradation of CANDU fuel, during long-term dry storage, are related to creep and hydrogen effects. Typically, spent CANDU fuel has much lower burnup than LWR fuels, this results in lower fission product inventories and lower residual heat, which result in lower storage temperatures. The effect of the lower temperatures is beneficial for most cladding degradation processes, except for phenomena, associated with hydrogen effects. Hydride precipitation and thermal cycling effects are enhanced in a temperature range close to the ductile-brittle transition, which is expected to prevail from the start of the dry storage period.

In the case of fuel elements with cladding defects developed during reactor operation, the potential exposure to oxygen, under abnormal storage conditions, could lead to cladding rupture as a result of fuel pellet oxidation and expansion. This is, however, a process that would occur only under abnormal conditions and would affect an extremely small fraction of the fuel and will not be examined here. The degradation processes that may affect fuel under normal storage conditions are discussed in further detail below.

Irradiation of CANDU Fuel

The bundle irradiation time in CANDU reactors ranges from 170 to 650 days, during which the UO₂ undergoes a burnup of up to approximately 300 MW·h/kg U. During in-reactor service, the fuel cladding collapses onto the fuel pellets, due to the 10 MPa hydrostatic pressure, applied by the primary heat transfer system. This collapse enhances the heat transfer from the pellets to the heat transfer fluid, but also increases the pellet-cladding interaction, which is also affected by the thermal expansion of the UO₂, and the fracturing of the pellets.

Thermal expansion of the fuel pellets and fission gases released from the fuel apply circumferential and longitudinal stresses to the cladding. The ceramic UO₂ has a low thermal conductivity and as a result, during irradiation the pellets have higher centreline temperatures and lower surface temperatures. The combination of thermal characteristics and properties, derived from the manufacturing processes, result in the fuel pellet shape changing from a straight cylinder to an hourglass shape, during in-reactor operation. This results in the formation of annular cladding ridges, which subsequently develop into regions of higher residual stress.

Interactions with the fuel pellets, heat transfer fluid and pressure from the fission gases generate stresses and resulting strains in the cladding during and after irradiation. In addition, the high-energy (>1 MeV) neutron flux in the reactor core causes radiation damage, altering the mechanical properties of the cladding. Using a correspondence of a 1×10^{24} n/m² fast fluence to a burnup of 24 MW·h/kg, the cladding is estimated to receive high-energy neutron fluences ranging from 1 to 15×10^{24} n/m². In the current spent fuel inventories from Canadian reactors, the fluences received peak at two values, with approximately one third of the cladding receiving $\sim 5.7 \times 10^{24}$ n/m² and two thirds $\sim 9.7 \times 10^{24}$ n/m².

7.2.6.2. Cladding properties after irradiation

Exposure to high-energy neutron irradiation in the reactor causes radiation damage to the fuel cladding in the form of defects and dislocations in the zirconium grains. The changes in the cladding mechanical properties are generally an increase in the strength and a decrease in the ductility of the material [50]. The mechanical properties of Zircaloy-4 fuel cladding were measured before and after irradiation in the as-received, alpha annealed, and beta heat-treated conditions. The beta heat treatment gave the material a microstructure, similar to that near the brazed spacers and pads on the fuel cladding. The specimens were irradiated to a fluence of 2.0×10^{24} n/m², E >1 MeV, at temperatures from 125°C to 250°C, and were subsequently tested at 20°C and at 300°C. The results of these measurements (closed-end burst test properties) of Zircaloy-4 cladding are given in Table XV.

The data in Table XV show that the yield and ultimate tensile strengths of the cladding increase and that the uniform and total elongations decrease with irradiation. Similar behaviour has been observed in fuel cladding irradiated at 300°C by Coleman [51] and Yasuda [52]. Yasuda et al. also observed that the increase in yield strength and the reduction in total elongation increased more rapidly with fluence below 10^{24} n/m² (E >1 MeV), and increased only slowly with fluence above this value.

To evaluate the potential for rupture of the fuel sheath, the stresses and strains, to which the sheath is subjected following reactor irradiation, are required. These stresses vary widely depending on the total burnup and the operation history of the fuel, on the temperature of the fuel and on the location being considered on the fuel sheath. A series of calculations were done of stresses and strains at 180°C, which is an estimated limit for the maximum fuel sheath temperature of 10-year old CANDU fuel stored in a DSC. The maximum hoop stresses through the length of the fuel element, out of reactor, were found to be from 24 to 44 MPa.

The maximum value of stresses calculated at end-cap/sheath region, under the same conditions considered above, were less than 70 MPa. In this area the radial and axial stresses were slightly higher, reaching 82 and 89 MPa, respectively.

The above estimated stresses are lower than the ultimate hoop strength values reported by Hardy by factors of 3 to 10. Therefore, fast tensile failures of the fuel sheath will not occur.

In previous assessments, the long-term effect of residual stresses on creep and cladding rupture was examined by various methods, including the Larson-Miller creep rupture parameter approach [53] and a mechanistic approach, as described by Cunningham et al. [54]. Predicted total creep strains have been estimated after 100 a, and the effect of strain localization has been evaluated. The results, quoted below, include creep and mechanical property data from AECL and other studies on Zircaloy fuel sheaths and materials. In addition, the effects of pre-irradiation have been considered, in particular, the possibility of reduced ductility due to strain localization.

Table XV. Mechanical Properties of Zircaloy-4 Cladding Measured Using Closed End Burst Tests Before and After Irradiation (Hardy 1970[50])

Initial Condition	Test Temp (°C)	Pre-irradiation Properties				Post-irradiation Properties			
		Yield Strength 0.2% Offset (MPa)	Nominal Ultimate Hoop Strength (MPa)	Uniform Circumferential Elongation (%)	Total Circumferential Elongation in 48 mm (%)	Yield Strength 0.2% Offset (MPa)	Nominal Ultimate Hoop Strength (MPa)	Uniform Circumferential Elongation (%)	Total Circumferential Elongation in 48 mm (%)
As-received	20	730	810	4.0	9.0	870	925	0.9	3.1
Alpha Heat Treated	20	540	610	8.7	37.0	680	730	1.1	23.3
Beta Ht. Treated	20	425	500	6.2	18.4	600	605	1.0	12.5
As-received	300	440	480	2.0	4.9	550	570	0.7	2.4
Alpha Heat Treated	300	180	260	11.3	63.5	410	435	0.8	38.0
Beta Heat Treated	300	185	250	7.1	26.3	330	355	0.7	11.0

7.2.6.3. Creep rupture

When a material is required to support stresses below its ultimate tensile strength for extended periods, it may deform by a creep process. Under certain conditions this deformation can lead to fracture (creep rupture) of the material. The presence of incipient defects in the material may affect creep rupture by increasing local stresses.

Direct measurement of the expected low creep rates under storage conditions is not practical, therefore, different analytical approaches to predicting creep rupture lifetimes need to be used, usually based on material behaviour and properties determined under different conditions. The Larson-Miller approach has been used to extrapolate creep behaviour observed at higher temperatures and pressures to those corresponding to dry storage conditions, but recently criticisms have been raised and highlighted the difficulties with applying this approach to creep rupture under dry storage conditions. A mechanistic approach has also been used, where rupture lifetimes are based on an understanding of grain boundary creep, which is thought to predominate under dry storage conditions.

The possibility of cladding failure by creep rupture during dry storage has been extensively examined in the US dry storage program, as well as for CANDU fuel [55].

In their report on the degradation of spent LWR fuel in dry storage, Cunningham et al. [54] used a mechanistic model to estimate creep rupture times. The following discussion, applied to CANDU fuel, follows their approach.

At high temperatures and high stresses, such as those used by Coleman et al. [51] and Peehs and Fleisch [56], creep is dominated by dislocation climb, with the climb rate controlled by lattice diffusion. Since dislocation density can vary greatly with irradiation, a difference in creep behaviour between irradiated and unirradiated samples would be expected, in agreement with observations. For low temperatures and low stresses, creep is dominated by grain boundary sliding, which is controlled by grain boundary diffusion. For this latter mechanism, the creep rate is not affected by irradiation. The transition temperature between these mechanisms is predicted to be about 300°C at 100 MPa, 400°C at 50 MPa, and 500°C at 20 MPa.

The interpretations of the temperature and stress dependencies of creep rupture times are more complicated than for creep rate. The rupture time depends on both the creep rate and the grain boundary fracture resistance. Intergranular fracture mechanisms dominate creep rupture times at dry storage temperatures. At stresses above ~150 MPa, fracture occurs by triple-point cracking. At stresses below ~150 MPa, fracture occurs by grain boundary cavitation.

For the CANDU fuel cladding, the storage temperatures and stresses are predicted to be less than 200°C and 100 MPa, respectively. This would place the fuel sheath within the region where creep fracture is controlled by cavitation diffusion. Using the corresponding expression for time-to-fracture given by Chin and Gilbert [57], the mechanistic approach predicts creep rupture times greater than 10^6 years.

A review of these results in the light of data that has become available during the last five years is being conducted, in order to estimate a defensible time limit for extended dry storage of CANDU fuel. However, the results are considered sufficient to support the safety assessments done for interim storage periods of up to 50 years.

7.2.7. RBMK and WWER-440/1000

For RBMK and WWER-440/1000 fuel, the assessment of the cladding degradation under dry storage conditions has to focus on the creep behaviour of the Zr1Nb cladding used in all reactor types. A storage programme was carried out to demonstrate the performance of Zr1Nb clad WWER-440 fuel in the temperature range of 330–380°C [4]. The post-test investigation showed no alteration of the fuel rods, especially no diameter increase beyond the detection limit. These results had been expected, and are in accordance with test predictions. It was concluded that, since the WWER spent fuel with a burnup of 36 GW·d/t HM exhibited no detectable hoop strain at dry storage between 380 and 400°C, this experiment proved that WWER spent FAs behave like other fuels with comparable end-of-life conditions,

7.3. TRANSITION MODES BETWEEN DIFFERENT STORAGE CONDITIONS

7.3.1. Transition from wet to dry storage

In most cases, wet storage of spent nuclear fuel is followed by dry storage. The transition from wet to dry may occur either relatively early (e.g. 3–10 years) if the back end strategy includes dry interim storage, or much later if the spent fuel management strategy includes extended wet storage (e.g. Sweden). Since the mechanical properties of the cladding and structural materials are expected to change very little during the wet storage period, the time of transition will not have a significant influence on fuel performance during the transition process.

The drying process of spent fuel is a very important step during preparation to the dry storage. Removal of all residual water from the storage container and the fuel requires a vacuum drying process, this results in an increase in the spent fuel temperature, which will usually exceed the equilibrium storage temperature, due to the absence of heat transport by the cover gas. For most of the fuel, the stresses and strain, related to the changes in temperature during drying, have negligible effect on the fuel assembly integrity. However, the temperatures reached during vacuum drying could be quite elevated. If the vacuum drying period is relatively short, the temperature increase is limited to a range between 10°C and 30°C. However, since the temperature dependence of the Zircaloy creep is of the Arrhenius type, even small temperature increases could result in enhanced creep. Tertiary creep resulting in cladding failure should not occur, since the vacuum drying periods are much too short and the associated temperature increase over the licensed maximum storage temperature is too small to induce enough deformation. Another aspect resulting from the increased temperatures during vacuum drying is the recovery of the irradiation hardening from the cladding material. The recovery starts at about 350°C, and may be considerable for temperatures above 400°C. However, in most of the cases, the recovery has no influence on the estimated dry storage performance. If calculation of creep in dry storage makes use of the irradiation hardening, the portion of recovery, occurring during vacuum drying, should be considered in the analysis.

Fuel handling procedures are normally designed and performed in accordance with the mechanical properties of fuel assemblies. Therefore, based on the above considerations, under normal operating conditions, these operations are such that possible failure modes are unlikely.

In summary, if the handling procedures (designed in accordance with the fuel assembly mechanical design) are strictly followed, no mechanisms exist that might cause the failure of operationally intact fuel elements during a transition from wet-to-dry.

The transition performance of fuel assemblies with *cladding defects*, originating from reactor operation, is however, heavily influenced by the process of water removal from fuel rod internal void volumes. Drying of spent fuel with cladding defects needs a careful adaptation of the drying process to the water release mechanism, to ensure that all water is reliably extracted. At the increased dry storage temperature, residual water remaining in the fuel rod internal void volume could conceivably react with the UO_2 to form U_3O_8 and $\text{H}_2\cdot\text{U}_3\text{O}_8$. This in turn could impose mechanical stress on the cladding, due to the resulting volumetric increase with respect to UO_2 . Therefore, careful drying is recommended when operationally defected fuel is entering the dry storage.

7.3.2. Dry-to-wet transition

The most critical part of the dry to wet transition process is the flooding of the fuel with water. The temperature drop can be sufficiently fast (quenching) to result in high thermal stresses in the cooling zone. However, simple calculations indicate that the stresses remain well below the yield material strength. In addition, the available operating experience shows that the cooling-down process has no detrimental effect on fuel assembly integrity. During dry storage demonstration tests [56], casks were flooded and the cask temperatures were carefully recorded. The total cooling time varied from 1 to 5 hours. The fuel rod quenching at the waterfront could be clearly observed. Flooding of the cask during these tests did not affect the fuel assembly integrity. Intentionally failed fuel was also placed in the test casks. Although a significant increase of the $^{134/137}\text{Cs}$ activity was observed, there was no further deterioration of the identified cladding defects.

It was concluded from these dry-to-wet transition tests that no mechanisms existed that might cause the failure of intact fuel elements.

8. BEHAVIOUR OF STORAGE FACILITY COMPONENTS

This subject is also the topic for the IAEA CRP on Ageing of Materials in Spent Nuclear Fuel Facilities which will be reporting in 2003. In the current text the behaviour of these materials will be summarised for completeness and any experiences reported within the SPAR CRP will be included.

8.1. BEHAVIOUR OF METALLIC COMPONENTS IN WET AND DRY STORAGE

The durability of spent nuclear fuel facility components in wet storage was dealt with in some detail in Ref. [58]. In the case of the internal environment of dry storage systems, excluding air filled systems, the only conditions, which exist for corrosion, are from residual moisture and pool chemical additives remaining after loading operations. All external surfaces and interfaces, however, are still susceptible to attack.

In the case of wet storage all materials corrode in the storage environment, the variance is in the rate of corrosion of the different materials. For example, unprotected mild steel has a significant corrosion rate in water. The second point to remember is that as storage duration increases up to 50–100 years, the general corrosion rate for some materials used in storage systems can become a life-limiting factor. Thirdly, additives, which have been used to minimise other corrosion mechanisms or for system operability purposes, may have detrimental effects on the long-term use of some storage components.

8.1.1. Stainless steel

Stainless Steel components are highly resistant to corrosive attack due to the formation of protective oxide films. For example, corrosion as a consequence of pitting or galvanic attack has been shown to be negligible [59], and the rates of general (or uniform) corrosion only become significant over extended time periods (100s of years). To put the later in context the measured corrosion rate of stainless steel in radiolysed water is $0.3 \mu\text{m}/\text{year}$ and $<0.1 \mu\text{m}/\text{year}$ in a caustic environment.

The most significant failure mechanisms affecting stainless steel, under the right conditions, are stress corrosion cracking and microbial failure. Pitting, galvanic or crevice corrosion are considered to be negligible.

8.1.1.1. Stress corrosion cracking

For a sensitised stainless steel, when the chromium content in the grain boundary has fallen below 12%, subjected to stress is brought into contact with aggressive ions, such as chloride, then stress corrosion cracking will result. Sensitisation of stainless steel can result by one of two mechanisms:

Thermal sensitisation

In a certain temperature range, the chromium in unstabilised stainless steels can migrate from grain boundaries and combine with any free carbon in the metal to form particles of chromium carbides. Thermal annealing processes are sufficient to re-establish a homogeneous distribution of chromium and thus remove thermal sensitisation.

Radiation induced sensitisation

Changes in the concentrations of chromium at metal grain boundaries can occur due to high-energy (fast) neutrons (and does not involve the formation of chrome rich carbides). Because of the irradiation profile along a fuel stringer or fuel assembly, the effect peaks in certain zones. Unlike thermal sensitisation, the sensitised microstructure remains as an artefact of reactor operation after the fuel is removed from the reactor.

8.1.1.2. Microbial attack

Over time, microbial colonies may become attached to pipes etc. establishing local environments that favour the growth of species, which would not otherwise survive. Established colonies are also highly resistant to fluid forces and most chemical treatments, making removal difficult. The microbial colonies effectively shield the metal surface from water treatment chemicals, rendering corrosion inhibitors ineffective.

It has been demonstrated that welding of austenitic stainless steel increases their susceptibility to microbial induced corrosion (MIC) compared to smooth pipe surfaces. In fact, the weld and the associated heat-affected zones become the primary area of MIC attack.

The impact of MIC has generally been ignored as it is hard to detect and this mechanism is usually masked by general corrosion. The only indicator of its influence is where components under periodic replacement programmes have to be replaced at slightly greater frequency than originally anticipated.

8.1.2. Aluminium alloys

The performance of aluminium alloys in wet storage has a wide range of corrosion behaviours, from rapid penetration to minimal corrosion over decades. In general, the stability of the material is very much dependent on retention of the protective oxide barrier. For example, uniform or general corrosion of aluminium is low in the pH range 4–8.5, but at higher or lower pHs becomes significant because of dissolution of the protective film.

The most significant degradation mechanism is from pitting attack; however, other mechanisms such as crevice, galvanic and microbial have also been reported, particularly in association with pool water chemistries with conductivities $>3 \mu\text{S}/\text{cm}$.

8.1.2.1. Pitting corrosion

Pitting corrosion is most prominent in high chloride environments. Pits tend to develop at defects in the passive film where chloride ions not only promote local depassivation of the protective oxide film, but also stimulate metal dissolution and hence promote growth of the resulting cavity or pit.

The attack is generally limited to small areas where local chemistries develop in the pits. In some cases the chemistries are isolated from the bulk water by crusts that form over the pits. Improving the water chemistry in such cases may only slow the mechanism rather than totally inhibit the attack.

A specific experience of pitting corrosion has been experienced with the outer high-grade aluminium cask seals.

The 1000 series alloys are high purity aluminium alloys that in most environments provide a high level of corrosion resistance, due to the rapid formation of a thin, tight, and protective aluminium oxide film. However, in some environments, such as those containing boric acid, with chlorides present, the alloys can approach or even exceed their pitting potentials. In this situation chlorides can destroy this passive protective oxide film, leading to pitting. Aluminium in this condition, which is then placed in close contact with a metal that is more cathodic, will undoubtedly experience corrosion as a result of the galvanic cell that is formed once pitting has occurred.

8.1.2.2. Galvanic attack

Aluminium alloys lie at the active end of the galvanic series. Galvanic corrosion of aluminium alloys has been observed in the fuel storage environments, when coupled to stainless steels and even when different aluminium alloys are coupled in waters with high conductivities at $>100 \mu\text{S}/\text{cm}$.

In highly pure waters ($1\text{--}3 \mu\text{S}/\text{cm}$), galvanic effects have not appeared to degrade the service capability of Aluminium alloy components; except in dissimilar metal services, that facilitate development of low pH and concentration impurities.

8.1.2.3. Crevice corrosion

Crevice corrosion is a localised form of attack, occurring in closely fitted surfaces that are sufficiently offset to allow water entry. The suggested mechanism of attack is where chloride ions are drawn into the crevice, as metal dissolution proceeds and acid conditions develop.

As with galvanic attack, mitigation is through maintaining water conductivities in the 1–3 $\mu\text{S}/\text{cm}$ region.

8.1.3. Carbon steels

Untreated carbon steels have relatively poor corrosion resistance in oxygenated waters at low temperatures, and therefore do not find general application. To overcome this effect carbon steels are normally treated with zinc and/or coated with epoxy.

The main degradation mechanism to be considered, therefore, is pitting corrosion, where the protective coating has been damaged during operation or the original material preparation was poor. Where such defects occur, pitting corrosion will proceed in the presence of aggressive ions, such as chloride. Once again, the maintenance of good water chemistry conditions will minimise this effect.

8.1.4. Cast iron

Similar to untreated carbon steels, cast iron has relatively poor corrosion resistance. Its application is mainly in the manufacture of transport/storage casks and cask handling equipment. The material behaviour, however, is taken into consideration in the design of storage components, such as dry casks, where cask body thickness is derived from the shielding requirements; proposed duty, assuming a uniform corrosion rate; and a safety margin (e.g. 100%). To mitigate against uniform corrosion, cast irons are usually treated with several coatings of epoxy.

Nevertheless, it should be mentioned that long-term corrosion tests with ductile cast iron have been made under extreme conditions in salt water and showed an equilibrium corrosion rate of about 1/10 mm/year. This is the result of the good adhesion of the oxide layer, in cause of the high content of carbon in the cast iron.

8.2. BEHAVIOUR OF REINFORCED CONCRETE

8.2.1. General properties

Concrete has been used for both wet and dry storage facilities. Storage conditions may differ significantly in wet and dry storage (as an example, concrete temperature of storage components varies from 30 to 80°C). As a consequence, in the framework of spent fuel long-term storage, requirements for concrete durability shall be defined in order to avoid storage facility lifetime limitation.

There are many possible causes of distress in reinforced concrete structures; they include (i) physical conditions such as movement of the foundations, structural overloading, accidental impact damage, and (ii) chemical/environmental effects, described below.

8.2.1.1. Carbonation of concrete

Carbonation, or more correctly the neutralisation of concrete, is the process in which carbon dioxide from the atmosphere (on average 0.03% by volume) diffuses into the concrete and chemically reacts with the products of cement hydration. The reduced alkalinity (pH down to about 8.0) that results from this effect, leads to a loss in the ability of the concrete to passivate

the steel reinforcement so that corrosion can then occur. In areas of high pollution, neutralisation also occurs, as a result of the presence of sulphur dioxide in the atmosphere.

8.2.1.2. Frost attack

It is important that dense concrete (i.e. high standards of compaction with low porosity) are achieved so that the amount of free water in the concrete is kept to a minimum, making it less vulnerable to the action of “freeze-thaw” cycles that can progressively burst-off surface concrete.

8.2.1.3. Chloride ingress

The most common material, which can destroy the protective passivation of steel in concrete, is the chloride ion. Chlorides may be present in concrete from the time the concrete was originally mixed (i.e. is encast), or they may migrate into the concrete in service (i.e. is due to ingress).

The process of concrete deterioration due to chloride ingress is presented in the figure 11 below.

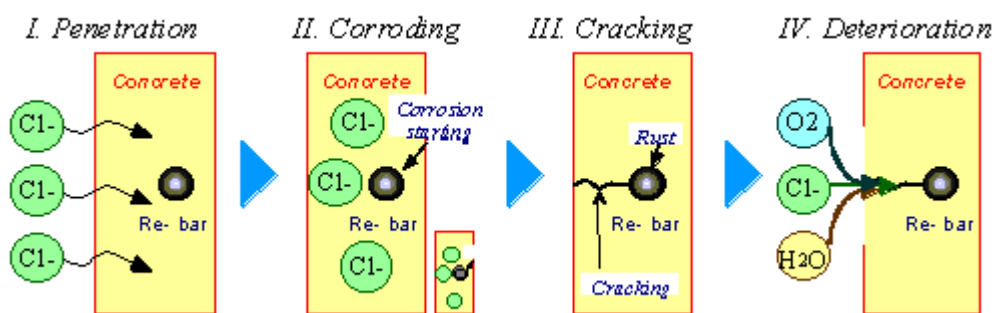


FIG. 11. Process of concrete deterioration due to chloride ingress.

Chloride ingress leads the cracking to the reinforced concrete components, because the corrosion and volume expansion of the steel bars occur.

8.2.1.4. Specific chemical based deterioration mechanisms

There are two types of deterioration mechanism that can result as a consequence of impurities in the concrete:

- Alkali-aggregate reaction (sometimes called “alkali-silica reaction”),
- Delayed ettringite formation.

Notably, these two chemically based defects can be avoided by the use of limestone aggregates. It is imperative that concrete structures are regularly inspected, so that warning signs can be detected and the causes of defects identified.

8.2.2. Concrete used for wet storage

Under wet storage conditions, concrete is permanently saturated. As a result, it is normally protected from corrosion by lack of oxygen. Carbonation of concrete beyond the water surface may take place in storage pools. In addition, respiration of the pores caused by alternate wetting and drying (e.g. at the level of the air-water interface at the top of pools) or heating and cooling may also speed carbonation. Therefore, examination of the general conditions of the concrete components, such as cracking occurrence, pool water chemistry, coastal environment and humidity considerations, will be useful to evaluate deterioration of their performance. The following paragraphs outline techniques, though not presently proven and established, which have the potential to become useful evaluation techniques.

8.2.2.1. Linear polarisation resistance measurement (LPRM)

The LPRM technique lies in its ability to measure corrosion resistance, coupled with the more usual measurement of electro-corrosion potential by the half-cell method. Below a certain threshold value, the rate of corrosion is not considered to be a problem. Consequently, by periodic measurement of the rate of corrosion, it is possible to decide when action needs to be taken, because the rate of corrosion has started to take off.

8.2.2.2. Modal testing

The NDT method of “Modal Testing” essentially involves the application of a standard vibration signal (or spectrum), being physically applied to a fixed initiating point on the structure (involving relatively low energy input) with consequent local response vibration effects, being measured at a number of other remote fixed points on the structure. Exceptionally sensitive accelerometer transducers that are capable of measuring imperceptibly small time dependent oscillatory responses achieve this response measurement.

This technique has advanced in the last five years, owing to developments in PCs with much increased speed of computation, improvements in instrumentation, and the commercial availability of very sensitive low-noise transducers that give rise to inherently low electronic noise.

The application of this technique to existing pools as a means of monitoring ongoing global structural stability, particularly in the long term, appears to be a valid and reasonable approach.

8.2.2.3. Acoustic emission (AE) method

The AE method is typically utilised in material research, to study deformation and fracture processes, such as dislocation and crack initiation, and in non-destructive testing to evaluate structural integrity, monitor pressure vessels for incipient failure, etc. It is considered that the AE method, unlike the NDT method, would entail a considerable number of sensor devices. Consequently, while a dense arrangement of these sensors on the outside wall faces can be used to monitor the perimeter of the pool, it is unlikely that they will be able to monitor the middle (inaccessible) parts of the base slab (i.e. from sensors at the accessible bottom part of the perimeter pool wall).

8.2.3. Concrete used for dry storage

8.2.3.1. Canadian studies

The dry storage containers (DSC's) used at Ontario Power generation (OPG) for interim dry storage of used CANDU fuel at the reactor sites are designed for a service life of 50 years. A study was carried out in 1998, with the purpose of providing assurance that the DSCs will meet their requirements over the 50-year design life and to assess the effects of outdoor storage. More recently, within the context of development of a long-term management plan for all the Canadian spent nuclear fuel, a more extensive study has been initiated to assess the expected service life of the three different dry storage systems used in this country. The main results and conclusions of the 1998 study [60, 61] are described below.

Approach

Two teams of experts were assembled, one advisory team on concrete issues and another team of advisors on metallurgical issues. The following activities were undertaken to assess the effect of potential degradation mechanisms that could impact on the DSC's 50-year design life when stored outdoors:

1. Identify and assess durability issues on both concrete and the metallic components.
2. Review existing documentation about the integrity of the DSC concrete.
3. Identify which issues have been adequately addressed.
4. Document outstanding issues, their potential effects and possible mitigating actions.

Concrete

Temperature gradient

Temperature gradients across the concrete will affect free water distribution. The use of a silica fume blended concrete (CSA type 10SF) and a low water-content ($\sim 135 \text{ kg/m}^3$) to minimize free water was suggested as a low cost mitigation measure.

An analysis of the amount and movement of free water in the concrete was recommended in order to assess the extent of potential freeze-thaw damage.

Freeze-thaw cycles and cracking model

A description of the elements relevant of cracking due to freeze-thaw cycles is summarized below.

For freeze-thaw damage to occur in a DSC with air-entrained, low water/cement ratio reinforced concrete, the following conditions must exist:

- There must be pre-existing cracks,
- Free water must fill the cracks, and
- Concrete temperatures must be below the water freezing point.

Pre-existing cracks

Cracking of the outer (colder) side of the concrete is predicted to occur if the DSC wall is subjected to a high temperature gradient. The maximum temperature gradient would occur

when a DSC is first loaded with fuel that has the minimum cooling time (10 years). The nature of the cracking is not known but it is expected to be a number of shallow, narrow cracks. These cracks will partially close as the temperature gradient diminishes over time. In the case of micro-cracks, self-healing may occur when excess (free) moisture reacts with the non-hydrated cement in the crack region.

Free water

Free water will initially be evenly distributed throughout the concrete. After loading the container with fuel, water migration from the warm region to the cold region will take place. However, even if capillaries become fully saturated, freeze-thaw damage is expected to be minimal, or not to occur, if the concrete has an appropriate distribution of voids produced by the air-entrainment process.

When the DSCs are exposed to below-freezing temperatures, the free water would saturate the outside (cold) layer of concrete, condense on the inside of the outer shell and fill the gap between the steel and concrete as well as any pre-existing cracks. Water would run down the gap and saturate the lower regions of the concrete mass.

Freezing

Freezing of concrete will occur to a certain depth determined by:

- The ambient temperature conditions,
- The heat source (i.e. the age of the fuel),
- Freezing point of free water. The free water will likely have a lower freezing point (e.g. -3°C) than fresh water, due to its dissolved solids content,
- The insulating effect of the ground. The base of the DSC is in contact with the base slab/ground, which is expected to provide an insulating effect. As a result a large portion of the base concrete is not expected to freeze.

Further work is recommended to assess potential damage from freeze-thaw cycles via direct examination of concrete samples. This assessment should correlate in-situ moisture patterns with the real thermal history of the sampled containers. If necessary, the test containers should be subjected to thermal gradients to monitor moisture migration.

Metallic components

The potential for corrosion of the DSC bottom, in areas where the external protective coat might develop defects, was identified as a concern. It was suggested that this phenomenon should be addressed via the existing inspection and maintenance program and by corrosion prevention measures.

Corrosion Prevention

The DSCs are protected by a dual layer of epoxy paint. The application of an additional layer of a sealing epoxy coating to the bottom plate, and using thicker bottom plate were suggested.

It was recognized, that a risk of galvanic corrosion exists, due to the accumulation of water in the bottom part of the DSC, particularly in the weld areas. This risk could be minimized by proper weld design and welding procedures to provide adequate protection against corrosion, resulting from the microstructural gradient across the heat affected zone. It was also

recommended that the possibility of galvanic corrosion at the joints where stainless steel (e.g. vent and drain ducts) is joined to a carbon-steel plate be investigated in more detail.

Conclusions and recommendations

The following modifications of the DSC design or storage mode were made to reduce the overall risk of degradation:

- Consider indoor storage,
- Coat the DSC bottom plate with a protective sealant, such as coal tar epoxy,
- Consider manufacturing the DSC bottom plate from stainless steel.

The following monitoring activities were recommended:

Take core samples from the test container(s) to correlate in-situ moisture patterns with the actual thermal history of the containers.

Monitor winter ambient air temperatures in the Pickering used fuel dry storage facility, to assess the potential severity of freeze/thaw cycles.

The primary recommendation made in the report was for indoor storage. Indoor storage is expected to extend the life of the DSC surface coating, minimizing the need for replacement and should prevent stagnant water corrosion of the bottom plate. It is also expected to reduce the potential for freeze-thaw damage of the concrete by reducing the severity of temperature cycles.

8.2.3.2. German studies

The German steel/concrete-cask type CONSTOR[®] consist of an outer and inner steel-layer (liner). The space between the two layers is filled with heavy concrete, for shielding. The space is completely closed by welding, so that the environment has no contact to the space between the two layers.

Because of the high alkalinity (pH 12 to 12,6) in the closed volume, no corrosion is expected to occur. The visual inspection of a reopened cask wall showed that no significant corrosion effects exist.

8.2.3.3. Japanese studies

In 1997, a new study programme of verification tests for interim storage of “recycle fuel resources” (FY1997-2003) was started by the Central Research Institute of Electric Power Industry (CRIEPI), which is mainly related to concrete modular storage technology, such as a horizontal concrete silo and a concrete cask.

As the concrete modular structures for dry storage of spent fuel may be used under high temperature and severe radiological conditions, reinforced concrete components important to safety must be designed, fabricated, erected and tested to withstand the effects of natural phenomena and accident condition. For example, in the Japanese design code for concrete reactor vessels, allowable concrete design temperatures are specified as shown in Table XVI. This design principle is introduced from ASME Sec. III, Div. 2. Higher temperatures than the ones given in this table may be allowed in the concrete if evidence is provided which verifies that the increased temperature does not cause deterioration of the concrete.

Table XVI. Allowable concrete design temperatures

State	Temperature limits, °C	
	Local area	Local area
Normal operation	65	90 ¹⁾
Accident	175	350 ²⁾
Note: 1) such as around a penetration 2) steam jets in the event of a pipe failure		

To propose safety standards for dry storage concrete material and structures, the following studies are now in progress [49].

1. Long-term durability of concrete material under high temperature

It can be noted that temperature considerably affects the damage, due to chloride ingress, carbonation, etc. especially if the storage site is located near the seashore. To obtain the basic material properties at high temperature and to evaluate long-term durability of concrete materials, the following tests are now in progress:

- a. Chloride ion diffusion test and carbonation accelerated test in concrete at high temperature.
- b. Corrosion tests of steel bar in concrete at high temperature by pre-mixed salt solution.

2. Dynamic strength for concrete material under high temperature and in case of accidents

In the Japanese design code, there is a notice, that material characteristics must be provided considering the reduction in strength. To make clear the influence of the temperature on concrete material, chemical investigations and dynamic material tests at high temperature are now in progress.

3. Characteristics of temperature stresses, shielding ability and activation of concrete structures

The temperature stresses are one of the primary factors governing the design of reinforced concrete hollow cylindrical structures subjected to temperature gradients such as a concrete cask body. To investigate the conditions in which cracks, due to temperature gradients, initiate in reinforced concrete structures, heat transfer tests with several scaled concrete structures are now going on.

In addition, after cracking due to temperature gradients, shielding ability will decrease depending on cracking width. To contribute to shielding design criteria, which give an allowable crack size, shielding analyses with a slit model simulating a concrete crack are being conducted. Moreover, the effect of activation for concrete structures is also under investigation (See Figs 12–13).

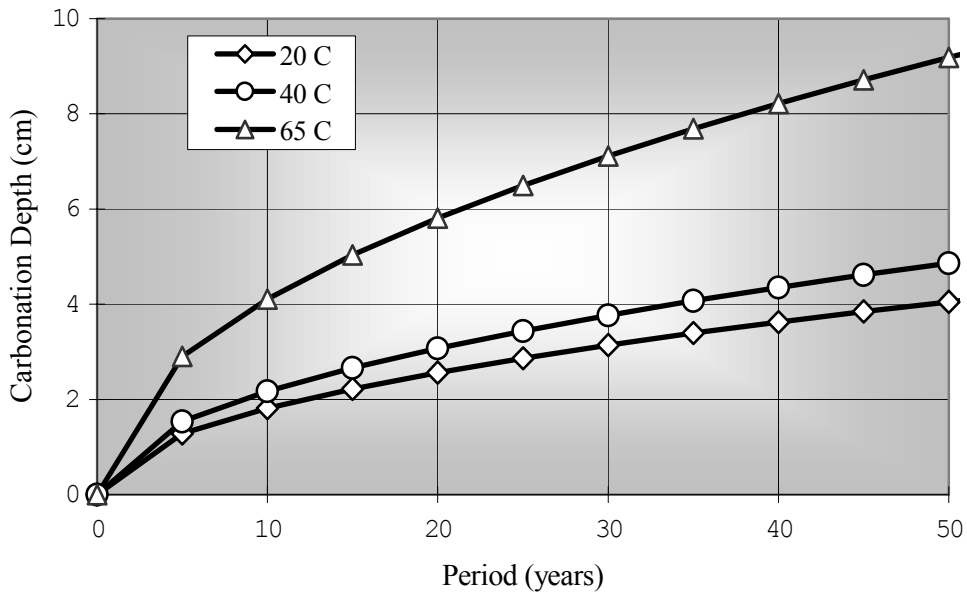


FIG. 12. Influence of temperature of carbonation depth and diffusion coefficients of chloride ion.

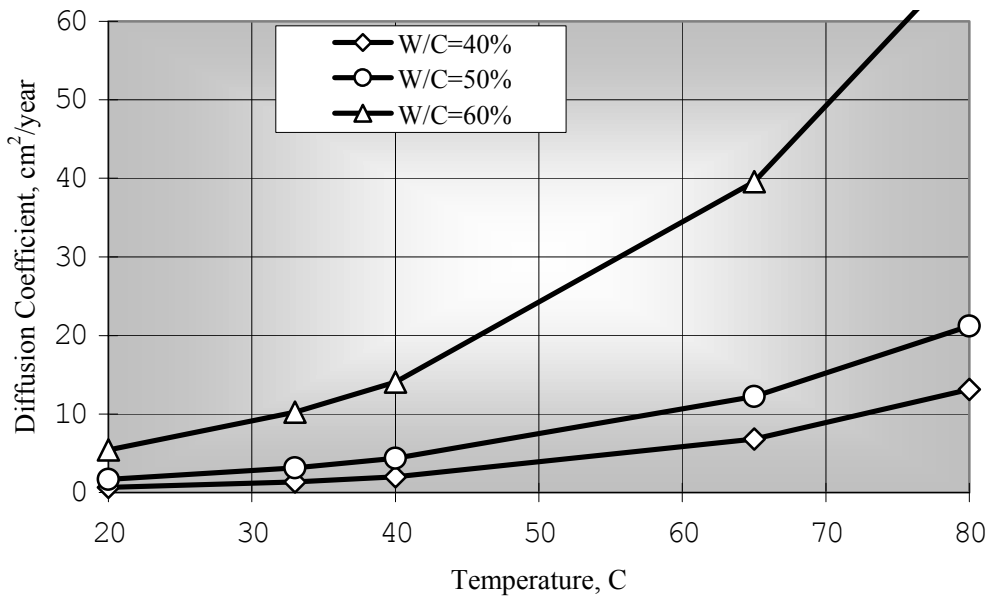


FIG. 13. Influence of temperature of carbonation depth and diffusion coefficients of chloride ion in concrete.

8.2.3.4. US studies

The NRC, EPRI, the DOE Office of Civilian Radioactive Waste Management (DOE-RW) and Environmental Management (DOE-EM) are participating in a research project to determine the long-term integrity of dry cask storage systems and spent fuel under dry storage conditions. The Castor-V/21 cask at INEEL, which was loaded in 1985 with fuel from Virginia Power, is being used in the program. Examinations of the cask and cask components were made in 1999 and 2000 and some fuel removed for detailed hot cell examination.

Inspections were performed on the cask exterior, primary lid bolts, primary cask lid seals, cask interior, and fuel assemblies to assess if degradation occurred, and where possible, to determine the mechanism for any degradation that was detected. The cask exterior was in very good condition with only minor small areas where some peeling of the paint had occurred. All bolts were in good condition with no evidence of degradation.

The two concentric o-rings (one metal and one elastomer) that constitute the primary lid seal were removed for inspection in 2000. The seals and grooved sealing surfaces were inspected. Both O-rings were in excellent condition with no evidence of cracks, breaks, delaminations, or oxidation. The elastomer ring was still flexible with no evidence of embrittlement. The metal ring also showed no signs of degradation with the sealing surface still ductile. The sealing surfaces in the cask and the lid remained shiny with no signs of oxidation. Slight discoloration was noted near the bolting circle where it appeared that some excess anti-seize compound might have run down the bolt-holes prior to the lid being torqued down.

The inspection of the cask and basket interior was somewhat hampered by the tight clearances of the design and the lack of accessibility to all regions of the hardware. It could be determined, however, that the interior cask surfaces were in good condition and that the galvanically applied nickel coating was still intact with only some scuff marks that appeared to be related to initial installation of the basket. The basket condition appeared comparable to the one shown in the videotapes made in 1985. Other than those welds, which were known to have failed from differential thermal expansion during the original fuel loading, no additional weld failures were observed and the previous failed welds appeared unchanged from the observations made in 1985.

In summary, the cask and cask components were all in excellent condition with no evidence of degradation. The visual check of the fuel also appeared to reflect no changes, but conclusive results will only be known after the detailed hot cell examinations are complete in 2002.

9. ISSUES RELATED TO LONG-TERM STORAGE

Whatever the back-end policy is for countries relying on nuclear technology for the generation of electricity, spent fuel inventories as well as interim storage durations are increasing. To this day (2000), the amount of spent fuel generated worldwide is 225 000 t, of which 80 000 t have been reprocessed. The balance, 140 000 t, is presently in wet or dry storage. Since the rate at which spent nuclear fuel is being reprocessed is less than the rate at which spent nuclear fuel is being generated, the inventory of spent fuel in storage will continue to increase over the foreseeable future.

In this Section, “extended” storage means an extension of the originally planned storage time. It is not defined specifically in terms of number of years, but typically encompasses a range of times from 50 up to 300 years.

9.1. EXTENDED STORAGE

9.1.1. Extended storage trends

Extending the period of interim storage is a general trend in all SPAR countries, regardless of the policy adopted for the closing the back-end of the nuclear fuel cycle. This fact mainly

results from the delays encountered by most final disposal programmes, or the “wait and see” policy adopted by some countries, especially those with a relatively small nuclear programme.

The increased need for extending storage times is driving several research projects in different SPAR countries.

9.1.2. Review of extended storage projects in the various countries

9.1.2.1. Canada

Canada has developed extensive experience in both wet and dry storage of CANDU fuel. Fuel has been stored in water pools at CANDU reactor sites for approximately 40 years, and in dry storage facilities for a large part of the past decade. Currently, Canada has 38 825 t HM of spent fuel in storage, of which 4239 t HM are in dry storage.

Canada has not selected yet a strategy for long-term management of its spent fuel. This decision will be made under the framework of nuclear waste legislation, which is currently under development and will not be enacted until 2003. Partly because of this situation and also because of the transition process that Ontario Power Generation is undergoing, research programs have been re-prioritised. As a result, the program for monitoring storage parameters in dry storage containers at the Pickering facility has been postponed, and will not be initiated until 2003. Therefore, monitoring results will not be available for some time.

9.1.2.2. France

The main objective of the third axis of the 1991 French Law is to investigate the very long-term storage of radioactive wastes before deep disposal. Studies have to cover the vitrified wastes, the technological wastes, and the spent fuel assemblies that will not be reprocessed.

At this time, the French Atomic Energy Commission (CEA) is studying technical issues concerning very long-term storage of wastes (“ETLD” project). The two types of storage, dry and wet, are studied through five concepts.

The first ETL D concept is a modular vault storage facility consisting of the following steps:

- Spent fuel decay in at-reactor storage pool for 5 years, followed by,
- Transfer of spent fuel within casks for storage in a wet, centralized away-from-reactor (AFR) facility for 25 years, followed by,
- Transfer of spent fuel to a dry AFR storage facility for conditioning within canisters and storage during about 300 years.

9.1.2.3. Japan

A domestic commercial reprocessing plant, with a capacity of 800 t HM/y, has been under construction. The aim is to be operational by 2005. However, the rate of spent fuel arisings in Japan will be more than this reprocessing capacity. In 2010, it is estimated that AFR interim storage facilities of spent fuel, with a capacity of about 7700 t HM, will have to be constructed. In 1999, a revised law, which specifies operational and regulatory conditions for interim storage facilities away from reactor sites, passed the Diet, and it was enforced in June 2000. Presently, an advisory committee of the Japanese Government’s Ministry of Economy, Trade and Industry (METI) is deliberating AFR storage. A private entity, which will be licensed by the Government, will be able to construct an AFR interim storage facility.

In Japan, there is no doubt that technical answers will have to be given to the licensing authorities. To evaluate the behaviour of MOX spent fuel, five rods irradiated in a commercial BWR to about 20 GW·d/t HM and stored for 20 years, were examined at the SCK/CEN. Three rods were stored under wet conditions, and two rods were stored in a capsule under dry conditions (air), after cutting into short segments. The pre-storage characterisation was performed on rods irradiated in the same fuel assembly in symmetrical positions. The measurements showed no significant release of fission gas and helium and no marked difference of the microstructure of fuel and cladding. Post-irradiation examination (PIE) of PWR-UO₂ fuel irradiated in a commercial PWR to 58 GW·d/t HM, and stored for 20 years under air, was carried out at the AEAT of the UK [62]. The visual inspection and the oxide layer thickness inside/outside the cladding showed no particular feature that could be attributed to 20 years of storage. EPMA measurement of Cs and Xe in pellet also showed that there was no significant migration and fission gas release after 20 years of storage. The trend in mechanical properties, such as Young's modulus, 0.2% yield stress, and ultimate tensile strength, with fast neutron fluence was similar to that reported in the literature.

9.1.2.4. Hungary

The licence of the plant's storage pools, as well as the licence for the MVDS, is issued for a period of 10 years, which can be renewed. The design lifetime of the plant is assumed to be 30 years, but lifetime extension activities are also planned. Since a period of 70 years is foreseen by the existing decommissioning plan for a safe storage of the plant components before decommissioning works commence, it is expected that spent fuel will have to be stored for a maximum of 100 years at the site. The design lifetime of the MVDS is 50 years; therefore, a number of technical activities are expected to be conducted, to support the future 10-year licence renewal processes.

9.1.2.5. Republic of Korea

The Korean policy for the back end of the nuclear fuel cycle is a "wait and see" strategy, which does not presently provide a specific approach beyond a long-term interim storage of spent fuel. With regard to the latter, the Korean Government set up a revised plan in September 1998, to complete an AFR interim storage facility by 2016.

In the near future, the choice of an ultimate spent fuel management option, direct disposal or reprocessing, is not foreseen. Therefore, there will be an increasing demand for long-term interim storage of spent fuel.

According to the September, 1998 Plan, the spent fuel will be stored at reactor sites until 2016, using several options for expansion in storage capacity. The expansion options include the installation of high-density storage racks, trans-shipment of spent fuel between neighbouring units, and the addition of dry spent fuel storage facilities at reactor sites. Therefore, flexibility in spent fuel management will be provided through this step-wise, at-reactor storage expansion scheme. The capacity of interim storage facilities will be 2000 t HM at first, and then expanded up to 20 000 t HM.

9.1.2.6. Russian Federation

Presently, Minatom of the Russian Federation is actively expanding research on maximum permissible storage times, and determination of optimum modes for dry storage of spent fuel assemblies (SFA) over 50 years.

The research addresses storing both intact and damaged SFA. To provide safe storage conditions, damaged SFA, which are candidates to be reprocessed, will be placed in sealed cans. Spent fuel assemblies, which are candidates for final disposal, have to be made monolithic, if they lost geometrical shape or are damaged up to bare fuel. A prerequisite in determining the optimum modes of dry storage is the requirement for SFA integrity, to allow retrieval for examination, and treatment or preparation for final disposal. Because many methods have been suggested for storing intact and failed spent nuclear fuel (SNF) under dry conditions, a procedure for monitoring the integrity of RBMK-1000 SFA after wet storage has been worked out. Use of this procedure is anticipated prior to the preparation of the SFA for storage. The preparation will include cutting the SFA (~10 m in length) into two bundles of fuel rods and the end fitting. Such cutting will be performed in a hot cell that is intended to be an integral part of the facility for preparing and storing SNF in metal and concrete casks (MBK). At the present time, these facilities are being constructed at the Leningrad and Kursk NPP, while at the Smolensk NPP, the planning, leading to construction is underway. At each NPP with RBMK-1000, spent fuel will be stored in MBK pending shipment to the federal SNF storage facility.

Dual-purpose metal and concrete casks for transportation and dry storage of RBMK-1000 SNF (TUK-104/109) have been designed. Their licensing is near completion.

The decision has been taken to construct, by 2005–2007, a federal facility for dry storage of SNF from WWER-1000, RBMK, and other reactor types with a capacity up to 39 000 t HM. The design of the federal SNF storage facility is nearing completion.

In July 2001, the President of the Russian Federation approved a package of four laws — new and amended — *On Leasing (of Nuclear Fuel), On Use of Atomic Energy, On Environment Protection, and On Radiation Remediation*. The adoption of these laws will allow not only to establish a reliable basis for development of the entire complex for closing the nuclear fuel cycle, but also to resolve environmental problems including those of extended storage of SNF.

9.1.2.7. Spain

In Spain, licensing of spent fuel storage at the reactor pool is linked to the reactor licence, and, therefore, part of the reactor life extension programmes. Additional storage will be provided by dry technologies. Regarding dry storage, the initial licence time calls for 20 years and could be further extended for similar time periods. The time the fuel is expected to be in interim storage, prior to its disposition in the repository, will be in average over 50 years.

9.1.2.8. UK

Currently, all MAGNOX and AGR spent fuel arisings are reprocessed. The future trend will be toward long-term storage of AGR fuel, once the reprocessing capability is no longer available. Storage for up to 80 years is envisioned, but this is dependent upon the availability of a HLW repository. In the case of MAGNOX fuel, no long-term storage is planned.

9.1.2.9. USA

In the USA, the reference spent fuel strategy is to permanently sequester the fuel in a geological repository. This strategy, in principle, precludes the need for very long surface storage, i.e. greater than 100 years. However, because of delays in the scheduled startup of the repository, it is expected that some fuel will be in dry storage for periods as long as

50 years. This storage may take place at the reactors' interim spent fuel storage installations, or at a centralised interim storage facility, such as the private fuel storage facility, or at the fuel staging area at the repository.

Interim spent fuel storage installations and storage cask designs are licensed for an initial duration of up to 20 years. To allow for continued storage beyond 20 years, data will be needed to verify and document that there has not been any detrimental degradation of the stored spent fuel and storage system components over the first 20 years of storage. In 1985, a joint program sponsored by DOE, EPRI and VEPCO led to the shipment of 69 assemblies of spent Surry fuel to INEEL for establishing a technical basis for storing commercial spent fuel in a dry, inert environment. By 1999, this Surry fuel had been in dry storage at INEEL for close to 15 years. A multi-year project to examine the fuel and selected cask components was initiated in 1999, and is scheduled for completion in 2002, to provide the confirmatory data to be used by licensees submitting an application for continuing dry storage beyond 20 years. The project is a cooperative effort among NRC, EPRI, and DOE.

9.1.3. Data collection and documentation

Extending the storage period may require additional investigations related to both spent fuel and storage systems. Knowledge on fuel evolution during very long-term storage (i.e. greater than 100 years) is needed, to predict potential effects resulting from active fuel degradation mechanisms. In addition, the storage system behaviour is also a key point for extended storage implementation.

Additional data related to material behaviour will be needed for spent fuel storage time extension. Several areas have been identified by various organisations (EPRI, ASTM, CEA, etc.). They include:

- High burnup and MOX fuels:
 - Effects of higher fission gas inventory, increased cladding oxidation and hydrogen pickup, fuel-cladding interaction layers,
 - Evaluation of active cladding degradation mechanisms.
- Dry storage system:
 - Evaluation of aging of the storage system components, with emphasis on deterioration of polymeric materials (neutron shields, seals, etc.).

The independent safeguard systems (IAEA/EURATOM) primarily rely upon surveillance equipment, i.e. appropriately placing fixed cameras linked to recorders, which facilitates a check between the observed and plant data. Where bulk transfers in containers are practised, plant data can be further supported by safeguard seals on the containers, together with supplementary radiometric scanners, to confirm the presence and age of the fuel.

It is important to keep logbooks documenting any difficulties in handling specific spent fuel assemblies, for example, so that this will be known the next time these fuel assemblies will have to be handled.

It is important to maintain the continuity of knowledge for the FAs during the whole fuel cycle, and have a system for registration and documentation, which uses a media for the information, which can be read and used in the future (more than 100 years).

9.2. HANDLING OF SPENT FUEL AFTER LONG-TERM STORAGE

Interim storage may last from several decades to more than a hundred years. Presently, most envisioned operation durations are between 20 and 100 years. In some cases, the design of the underground repository requires minimum storage durations that allow for decreases in the decay heat, so that the maximum temperatures licensed for the spent fuel assembly, canister, and the host geological environment are not exceeded.

Furthermore, potential degradation of the fuel assembly as a function of time, temperature, radiation, oxidation, change of environment, etc. cannot compromise retrievability of the spent fuel from interim storage under normal and off-normal conditions, or recovery after accidents.

In most cases when dry storage is considered, interim storage containers are not foreseen to be used for final disposal. Requirements for the latter strongly depend on the host rock type and the credit the canister has in the overall safety assessment of the repository. Therefore, spent fuel may need to be re-conditioned and re-packaged prior to its emplacement in the final disposal facility. Pilot and/or industrial facilities have already been constructed for this purpose in Germany and Sweden, and these will be described later.

In general terms, this type of facility only demands that the spent fuel assemblies are capable of being handled mechanically. This requirement is well established and expressed in the US Code of Federal Regulations [63], requiring that dry storage systems must be designed to allow retrievability of spent fuel for further processing or disposal. Similarly, in Germany, for dry storage systems, the RSK requires that systematic fuel degradation be excluded to ensure the overall safety of spent fuel storage and disposal.

So far, research programmes have supported the retrievability of intact fuel after up to 20 years of storage [20]. Cask demonstration programmes with fuel having cladding defects have also shown that this fuel can be successfully stored and retrieved without problems.

9.3. ENCAPSULATION/PACKAGING OF SPENT FUEL FOR FINAL DISPOSAL

9.3.1. Facilities for spent fuel final packaging

9.3.1.1. Germany: The pilot conditioning facility

Objectives

Since 1979 investigations have been made in Germany on the direct disposal of spent fuel elements from LWR's [64]. In 1985, the Federal Government in Germany decided to build a demonstration plant for spent fuel final conditioning. This was to promote and develop technology as part of a greater programme, leading to the final disposal of spent fuel in a salt dome at Gorleben in lower Saxony. The required processes and components were developed step-by-step until 1989. Presently, the pilot conditioning facility is practically completely erected and the facility is waiting for the commissioning licence. The scope of tasks to be performed by the pilot conditioning plant includes:

1. Treatment and packaging of fuel elements and radioactive wastes in a manner suitable for direct disposal.
2. Processing of all PWR and BWR fuel assemblies of different designs and manufacturers.
3. Loading of casks suitable for final disposal and further interim storage in the time period between the packaging process and the final disposal.

4. Conditioning of radioactive wastes and loading the conditioned wastes into casks suitable for final disposal and further interim storage in the time period between the packaging process and the final disposal.
5. Repackaging of packages not appropriate for final disposal for different reasons.
6. Repair and maintenance work on transport and storage casks.

The operational capacity of the conditioning plant is 35 t HM/year. However, the component and process design would be also valid for facilities with larger capacity.

Safety requirements

The packaging facility and the conditioning process comply with all relevant German regulations for nuclear installations. Prior to the erection, a basic assessment was made. This resulted in a positive statement with regard to the features of the installation with respect to civil, environmental and nuclear regulations valid in Germany. Among others, the safety requirements include:

1. Subcriticality for all spent fuel configurations that might occur during spent fuel conditioning assuming processing of fresh fuel with typical fuel enrichments foreseeable at the time of licensing.
2. Adequate shielding of the plant and physical protection for the operators assuming characteristic and foreseeable fuel burnup at the time of licensing.
3. Ability to handle defective fuel rods throughout the complete conditioning process.
4. Compliance with safeguard requirements.

As a consequence, the conditioning process is operated remotely, and to a large extent also automated.

The POLLUX cask for the final disposal of spent fuel

The POLLUX cask is designed to fulfil all requirements simultaneously for transport and storage as well as for final disposal in a gallery-type salt repository. The cask is able to receive fuel rods after their removal from the skeleton of spent UO₂ PWR assemblies, together with the compacted structural parts of those FAs. Alternatively, the cask can be loaded with the fuel rods from 7 spent UO₂ PWR-FAs and 3 spent MOX-FAs. Altogether the POLLUX payload is 5.5 t HM.

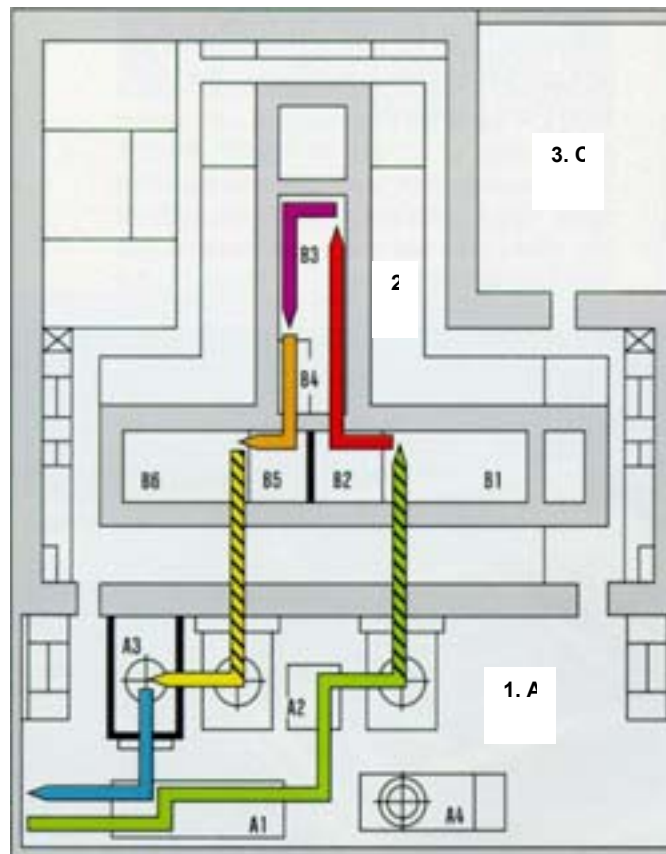
The functional process

The functional process steps for the fuel conditioning can be summarised as follows (see also Figure 13):

1. Arrival of the transport cask to the conditioning plant,
2. Transportation of the cask through the cask handling area to the basement under the unloading cell,
3. Dry unload the spent fuel in the hot cell,
4. Intermediate storage of the FA in the buffer store,
5. Rod consolidation and loading of the fuel rods into canisters,
6. Intermediate storage of the canisters in the buffer store,
7. Loading the canisters into the POLLUX cask,
8. Closing the cask by the primary lid,

9. Filling the POLLUX cask with He and checking for tightness,
10. Transportation of the POLLUX cask out of the hot cell wing to the welding gantry at the cask hall,
11. Closing the POLLUX cask by welding a secondary lid,
12. Closing the shielding overpack by screwing in the shielding lid,
13. Transportation of the completed POLLUX cask to the interim cask storage facility.

Depending on the type of final disposal packages, LWR fuel rods may be packaged as a whole or as chopped into 1-m long segments. The two possibilities to condition the spent fuel are shown in Figure 14.




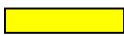





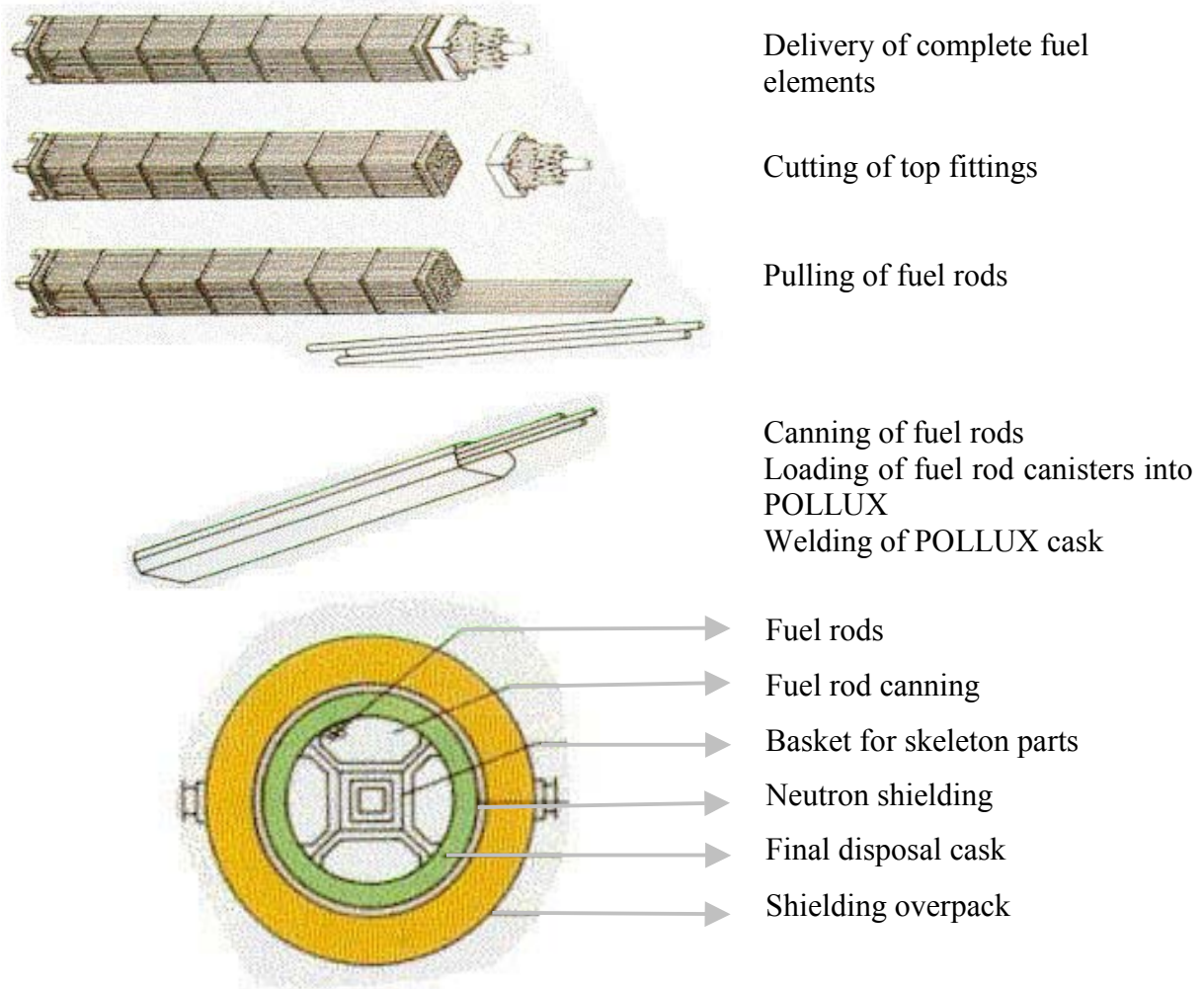
A	Cask Hall	B1	TC-unloading
B	Hot-cell area	B2	FA-intermediate storage (Buffer storage 1)
C	Utilities and personnel area	B3	Disassembly of FA
A1	Lock	B4	Loading of canister and basket
A2	TC-preparation and dispatching	B5	Intermediate storage of canister and basket (Buffer storage 2)
A3	Welding of POLLUX, leak test and screwing of shielding lid	B6	Loading of POLLUX and tightening lid
A4	Internal decontamination of TC		
	Transport cask (TC)		POLLUX screwed tight
	Fuel assembly (FA)		POLLUX welded and shielding lid screwed tight
	Fuel rod and components		Passage in basement
	Canister and basket		

FIG. 13. PKA: Packaging of consolidated LWR-fuel into POLLUX casks.



POLLUX-cask:

Diameter:	1.5 m	Length:	5.5 m
Weight:	65 000 kg	Content:	8 PWR fuel elements
Surface dose rate:	< 0.2 mSv/h		

FIG. 14. PKA: Conditioning systems.

Key process steps in the pilot conditioning plant

One key process step is **rod consolidation**. Upon removal of the FA nozzles, the assembly table moves to a gripping device. The gripping device grips a complete row of fuel rods and pulls them out of the FA-structure starting with the upper row, taking in the next sequence the row below and so on, until the skeleton is completely emptied. If the row of fuel rods just processed is completely removed from the skeleton, they fall into a trough. The trough is filled with the fuel rods of a little more than 2 PWR-FA or 8 BWR-FA ($16 \times 16-20$, or $8 \times 8-1$, Siemens type, respectively). In order to reach the highest degree of filling, vibration is used to densify the fuel rod package. The loaded trough is then closed by a lid and transported to the pushing device. The trough has the same geometry as the one in the POLLUX canister into which the fuel rods can be inserted. Four canisters are loaded into the POLLUX cask.

Another key process is the **compaction of the non-fuel bearing components** from the FA. Upon completion of the rod removal process, the fuel element structure is passed on to the

hydraulic compactor. The hydraulic compactor performs the compaction of hardware such as head-end and bottom-end nozzles, spacers, guide tubes, and fuel channels to a minimum volume. Through a feeding unit, the components to be processed are inserted by axial charging into the press channel. After insertion of the counter plunger, a first stroke of 900 mm is exerted. Altogether six strokes are made, and therefore five extension plungers are inserted. The compaction force has a maximum of 5000 kN. The compaction degree is 8:1. The compacted non-fuel bearing components are loaded in the central volume of the POLLUX cask.

9.3.1.2. Sweden: The encapsulation plant

The Swedish nuclear program, as it is planned today, will require about 40 000 BWR and 5000 PWR fuel assemblies to be encapsulated. A small amount of them are damaged, and special arrangements will have to be implemented before encapsulation. In addition, there are also some MOX fuel assemblies, and some odd fuel coming from research reactors. Besides the fuel, there will also be high-level waste, in the form of core components, which has to be taken care of.

The spent fuel is to be encapsulated in disposal canisters. Core components will be placed in some kind of moulds.

Cooling time before encapsulation is about 30 years. The residual heat in the fuel depends on many factors, but is calculated to be between 70 and 800 W per fuel assembly at the time of encapsulation; the average is 140 W for BWR and 420 W for PWR fuel assemblies.

In the current design, the SKB reference Disposal Canister consists of an outer copper casing for corrosion resistance, and an inner cast insert for mechanical strength. The final disposal is planned to be situated about 500 m down in the bedrock, with the canisters embedded in bentonite clay (See Figure15).

The spent fuel is stored in water filled pools at the CLAB interim storage facility. The Encapsulation Plant is planned to be built as an extension to CLAB (Fig. 16). This location provides possibilities to extend several existing service systems and functions into the Encapsulation Plant. As a first stage, only spent fuel will be encapsulated, but preparations are made for the later addition of equipment for treating core components.

9.3.1.3. USA

The United States Department of Energy plans to begin acceptance of spent nuclear fuel and high-level radioactive waste in 2010 for disposal in a monitored geological repository. These materials will be placed in disposal containers, or waste packages, designed to accommodate the variability in waste forms, sizes and nuclear characteristics. Waste packages have been designed for PWR (21 PWR) and BWR (44 BWR) assemblies, high-level wastes generated from defence activities (DHLW), and naval spent nuclear fuel (Naval SNF). Waste packages have been designed to take advantage of locating the repository host horizon above the water table in an unsaturated zone, which consists of relatively dry rock. As long as the waste packages remain intact, the waste will be contained completely within the packages and prevented from contact with the host rock, air, or groundwater. The waste package design has evolved and currently employs corrosion resistant Alloy-22 over stainless steel. This, combined with a titanium drip shield, will achieve a life for the waste package that exceeds regulatory requirements. (See Figure 17)



FIG. 15. Disposal canister.

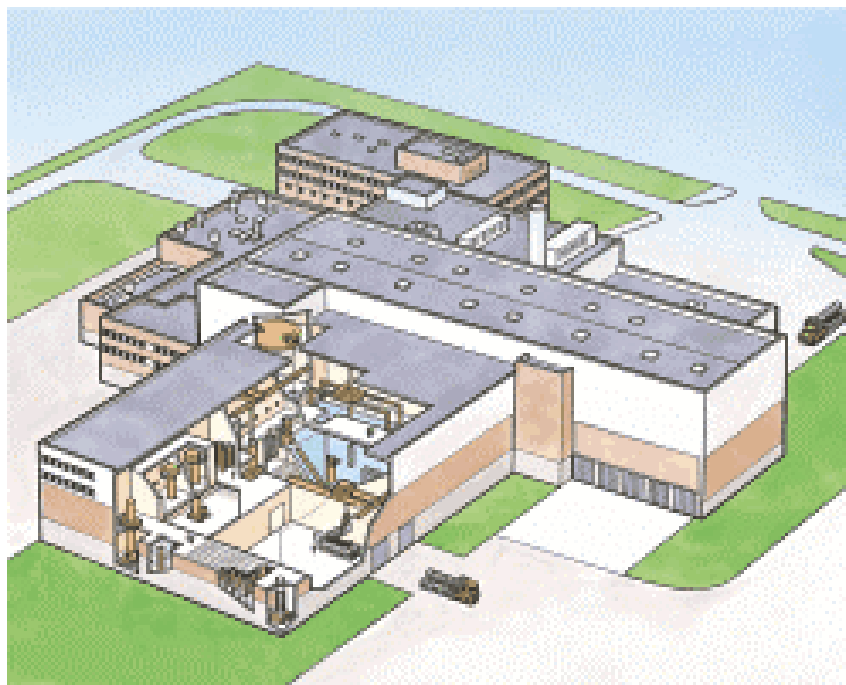


FIG. 16. Encapsulation plant connected with the CLAB facility.

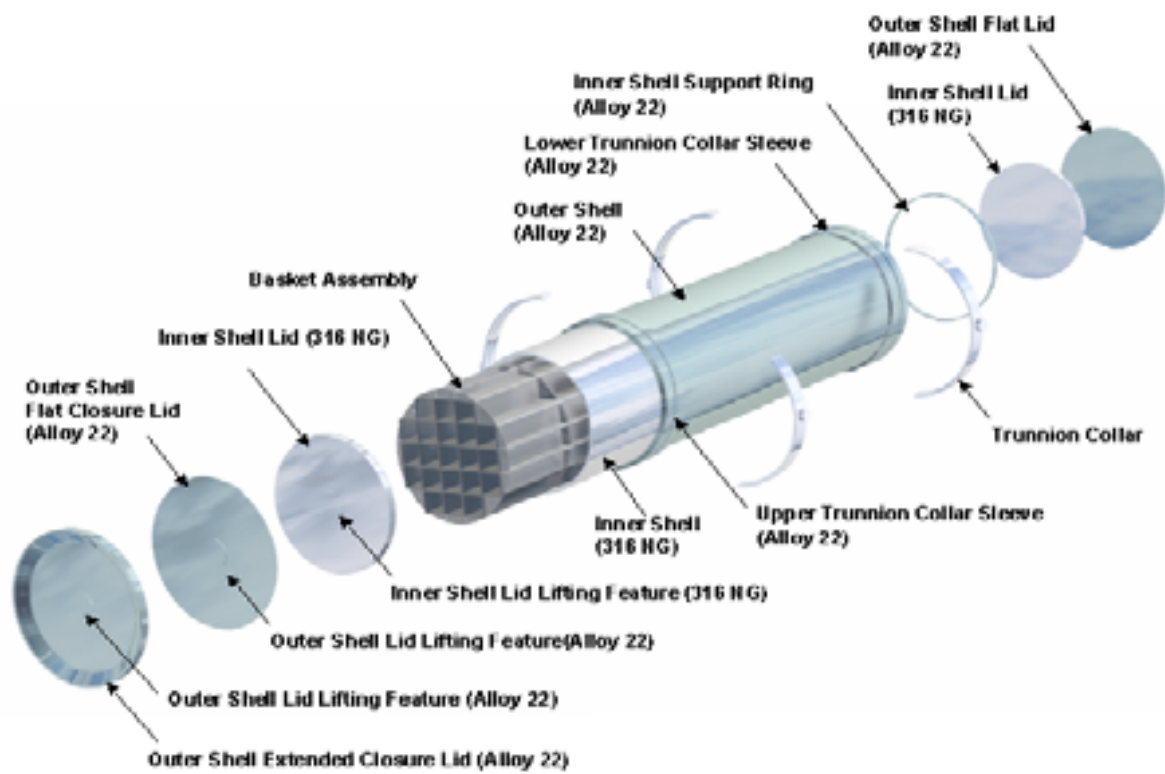
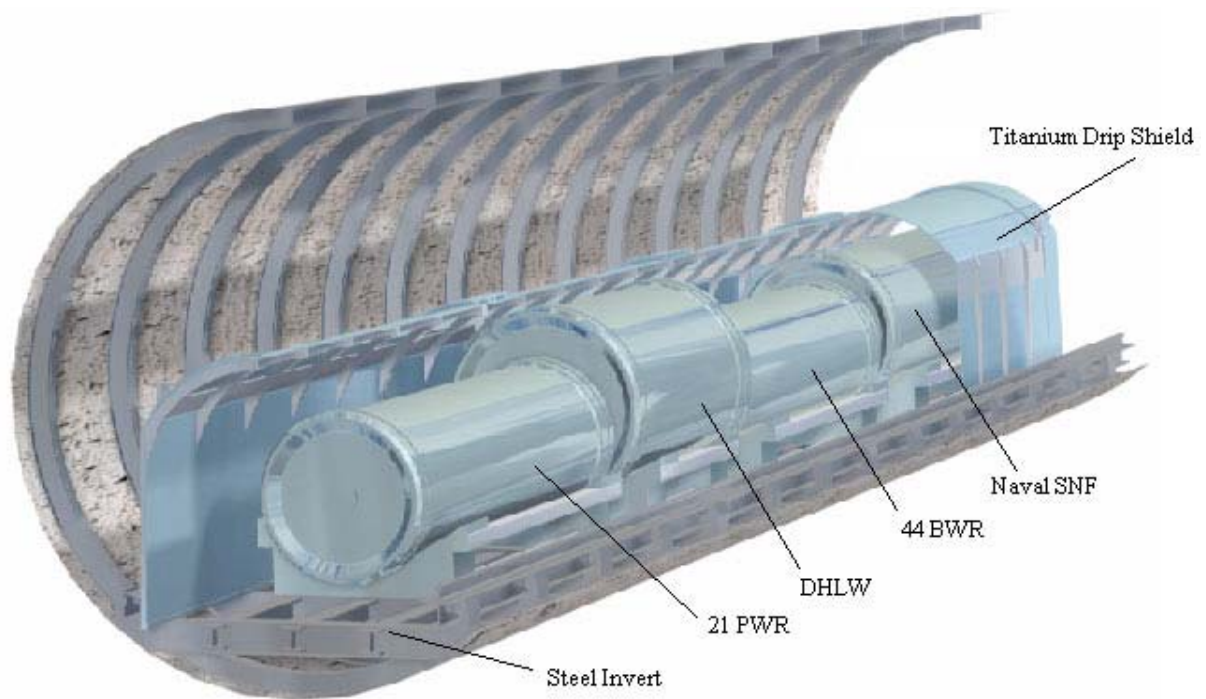


FIG. 17. Proposed waste packages in the US disposal system.

9.3.2. Studies on techniques to support the final disposal of spent fuel

9.3.2.1. Germany: Early encapsulation

The encapsulation technology [65] provides the possibility to encapsulate both spent PWR-FA and spent BWR-FA at the end of the wet storage period. The encapsulation process takes place in the spent fuel storage pools at the reactor site. It is based on well-approved service technologies in all individual process steps:

1. Among other considerations, there is the advantage that the final packaging of the spent fuel occurs in using reliably performing processes
2. The engineers, who are familiar with the design, fabrication and operation performance of the fuel will also perform the packaging process
3. The reliability of the encapsulation process is assured, by designing each individual process step reversible, repeatable and repairable.

After an appropriate decay time in the wet storage pond, the spent fuel assemblies will be individually encapsulated, directly in the fuel pool. The canister that contains 1 PWR-FA or 2 BWR-FA is internally dried, inerted, and then seal-welded. The capsule will be the first barrier to prevent the release of gaseous fission products. If the fuel is already encapsulated, this barrier function is moved from the cladding to the capsule for all subsequent processes in the back end of the fuel cycle. From this point on, decoupling between use in the reactor and the disposal of the fuel assemblies is accomplished:

1. The canister is a new, engineered barrier, specially designed for all requirements in the back end of the fuel cycle, e.g. the canister provides a complete mechanical stability for handling, storage, and transport purposes.
2. The new canister now takes on the barrier function for the fuel rod cladding tubes, since the cladding tubes have been considerably altered during its in-service operation by corrosion, hydriding, irradiation induced effects, etc.
3. The canister serves at the same time as a "waste package" for the spent fuel assembly, which can be used until placing the spent fuel into the final repository.
4. Canistering the spent fuel individually allows for the shortest interim storage time period prior to final disposal, without violation of the allowable hosting rock temperature, since the linear heat rate of the package is much smaller than for multiple FA-containers. This might be essential for UO₂-FA and MOX-FA having achieved a burnup in excess of 50–60 GW·d/t HM.

The equipment

The encapsulation process occurs remotely and will have the capacity of several spent FA per day. Early encapsulation will use available and approved technologies for transportation and storage casks and final disposal techniques. Only the baskets in the spent FA-casks will need some slight modifications. Figure 18 exhibits the back end of the fuel cycle when early encapsulation is selected for the final packaging of the spent fuel already in the spent fuel pool at reactor sites. The transportable encapsulation device consists of:

1. The encapsulation station with two individual process modules for working in parallel.
2. One joint service module that contains all necessary tooling and the welding device.
3. Two intermediate FA storage positions for receiving the spent FA from the storage racks, and after completion of the encapsulation process, for passing the encapsulated FA for either further storage in the pool, or for being immediately loaded into a waiting transport and storage cask.

4. A first transportation cover, to lower the empty capsule in a dry manner, into the encapsulation module.
5. A second transportation cover, with an integrated FA drying device, to transfer the FA from the receiving position to the encapsulation module and to dry the spent FA.
6. A third transportation cover, with an integrated He-leak-detector system, to remove the encapsulated FA from the encapsulation module, to provide necessary leak testing and to forward the encapsulated FA to the second intermediate storage position.

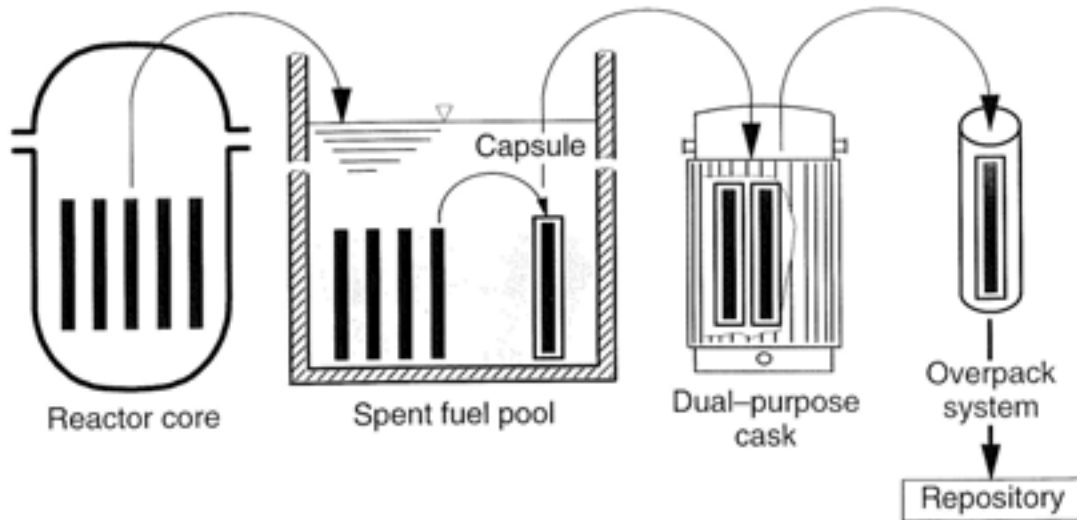


FIG.18. Principal steps in early encapsulation.

The Process

Figure 19 describes the complete encapsulation process. The overhead crane in the containment provides the necessary support for handling heavy components. Transportation of the FA is carried out either by the fuel manipulator crane, or by the auxiliary hoist available at the fuel manipulator crane. The following characteristic process steps describe the encapsulation process:

1. On the floor aside of the spent fuel storage pool, the empty capsule is raised into a vertical position and filled with argon.
2. The argon-filled capsule is transported, with its cover lid, in a dry manner to the encapsulation module, connected to the locking device, and lowered to the encapsulation module. The cover lid of the capsule is now removed into a waiting position.
3. In parallel to the process steps 1 through 3, a FA is taken with the FA handling machine from the storage racks and brought into the first intermediate storage position.
4. The FA is subsequently removed from the first intermediate storage position by the second transportation cover, docked on top of the encapsulation module that contains the empty capsule, and is dewatered.
5. After drying the FA to the specified value, by heating it with electrical heaters and by subsequent vacuum, the FA is ready to be inserted into the encapsulation module.
6. The FA is lowered into the capsule. The cover lid of the capsule is taken back from its waiting position and positioned on top of the capsule now containing the FA.
7. The capsule is then closed by an automated TIG-welding process. During the welding process, a valve — contained in the cover lid of the capsule — is kept open and allows a pressure equalisation between the capsule and the encapsulation module, thus avoiding a material blow-out at the very last moment when the weld tightens the capsule closure.

8. After completion of the main welding, the still open valve in the capsule cover lid is connected to a gas supply system, providing the right capsule internal gas pressure by feeding argon and helium. The helium is later on needed to perform the He-leak test. Finally, this gas valve is closed mechanically and welded.
9. The encapsulated FA is now tied up into the third transportation cover with the integrated He-leak testing system. Having passed the leak test successfully, the transportation cover is disconnected from the encapsulation module and brought to the second intermediate storage position, from where it might be transferred to the waiting storage position, or to the storage and transportation cask.

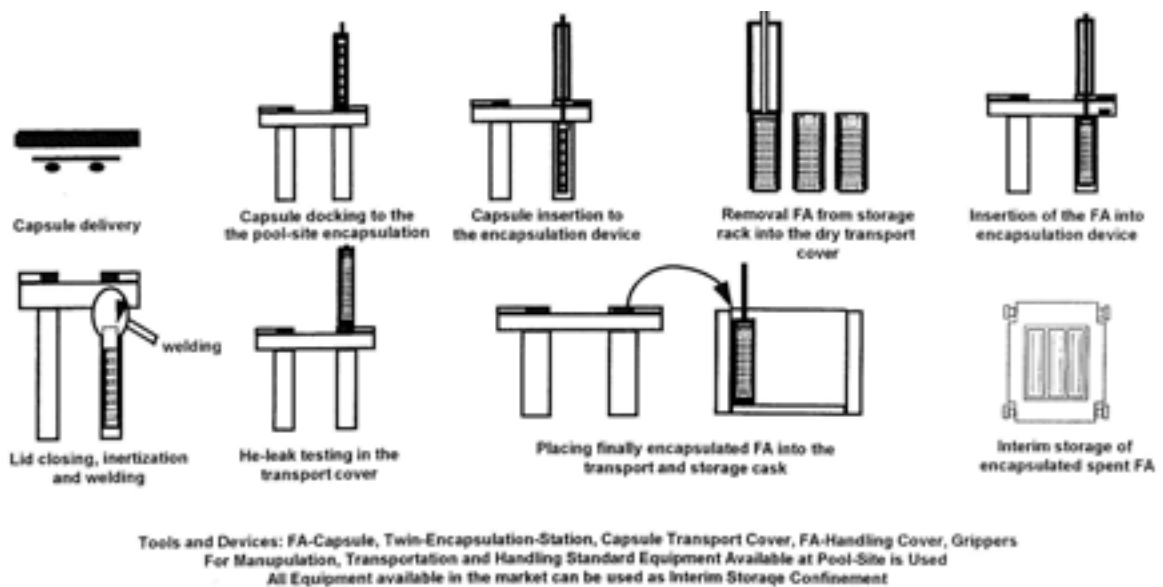


FIG. 19. Early encapsulation process.

The encapsulation process will be remotely performed in an automated manner. The “hands-off” mode in normal operation allows minimising radiation exposure to the operating staff. The process can be operated in the forward direction, to encapsulate the spent FA, and in the backward direction to decapsulate the spent FA. Decapsulation is possible from each intermediate process step, and from the final process step, to ensure that the spent FA can be removed from the encapsulation station independently of what has happened.

Each electrically driven machine is designed redundantly or can be remotely replaced by a new device. As a final measure, each electrically driven motion can also be operated manually using long handling tools.

Key process steps

The following steps will be discussed in greater detail, to illustrate how the available know-how is integrated in the spent FA encapsulation process:

1. From the two possible *welding processes* — laser welding and TIG welding — the TIG-welding was selected because of the larger experience under remote service conditions. TIG-welding can be performed with and without material supply. Experience shows that the material supply during remote welding might be troublesome; therefore, it was decided to weld without any additional material supply during welding. Argon will be

used as the welding atmosphere. Normally some hydrogen is added to argon, for better focusing the welding arc. Since there are safety-related issues with hydrogen addition, welding will be performed in pure argon. Gas cooled — instead of water-cooled — welding heads were selected to avoid problems with water in case of malfunction. Since the operation temperature of such welding heads is somewhat higher, increased wear of the W-cathode is likely.

2. With respect to long interim storage periods the FA *drying process* during the encapsulation is of great interest. After water rinsing is completed, there are still water films adhering to all surfaces of the FA, which will be easily evaporated with help from the decay heat. The removal of residual water present in the lower ends of the guide tube does not occur so easily, because only the heat capacity of the FA-structure can supply the heat of evaporation. Without any additional heat supply from external sources, this water may freeze during vacuum drying. Therefore, the concept foresees that — as a first step — heat is supplied by a hot gas (He) to the FA structure, from a heating system, integrated into the drying device. Vacuum drying starts if the cold spot in the structure is heated to $>90^{\circ}\text{C}$. A second heating/drying cycle assures that all water is reliably removed. This drying procedure was developed experimentally first. In a second step, this procedure was modelled theoretically. Finally the design of the drying equipment was performed, by using this experimentally verified design code.

9.4. POTENTIAL LOSS OF INERT CONDITIONS IN DRY STORAGE OF SPENT FUEL

9.4.1. Potential temperatures for loss of inert conditions for various fuel types and storage facilities

The following table (Table XVII) gives a survey of the dry storage technologies in the countries contributing to the SPAR-programme.

The information exhibited in the table above shows that spent nuclear fuel from LWR, WWER, CANDU, and HTR is stored under inert conditions. The licensed temperature is up to 380°C in the case of LWR and WWER spent fuel, and 250°C for CANDU-spent fuel. Practically the storage hot spot temperatures are much less. Most of the spent fuel is stored under temperatures much less than the licensed maximum hot spot temperatures.

MAGNOX fuel needs not to be assessed for the loss of inert condition, because the storage atmosphere for spent MAGNOX fuel is already not inert in normal operation. The AGR spent fuel in UK is stored exclusively wet. The spent HTR fuel is presently stored at nearby ambient temperature. Therefore, the following consideration is focused on the spent fuel from LWR, WWER and CANDU-reactors and temperature levels of less than 250°C . The materials to be considered are zirconium-based alloys, stainless steel, and UO_2 .

9.4.2. Air ingress

Limited access of air can be assumed if the seals of the dry storage system fail. When the SNF is free of defects, only the cladding is exposed to corrosion by air, and the fuel will not experience any alteration. When the SNF contains pinholes or larger defects and the fuel temperature is less than 175°C , fuel oxidation does not occur to any practical extent and fission products are not released; when the fuel temperature exceeds 175°C , fuel oxidation may occur and fission product release from the defective fuel may have to be considered.

Table XVII. Dry storage parameters in the SPAR countries

Country	Storage technology	Atmosphere	Kind of SNF	Tmax. /°C
Canada	Concrete canisters	Air, He	CANDU	250
	Modular dry storage	He	CANDU	250
Germany	Nodular cast iron casks	He	LWR, WWER, HTR	370
Hungary	Modular dry storage	N ₂	WWER	350
Japan	Cask	He	LWR	390
Republic of Korea	Concrete canister	Air	CANDU	159
Russian Federation	Concrete cask	N ₂	RMBK	300
	Metal concrete construction	He	WWER	330
Spain	Metal casks	He	LWR	380
UK	Modular dry storage	Air, CO ₂	MAGNOX	150
USA	Forged metal casks	He	LWR	380
	Nodular cast iron	He	LWR	380
	Concrete casks	He	LWR	350
	Modular concrete stores	He	LWR, HTR	350

9.4.2.1. Oxidation of zirconium-based alloys

Oxidation of zirconium-based alloys is a thermally induced process. During in-service operation, the surface of the cladding is already oxidized in a controlled manner, because the boiling or pressurised water, surrounding the fuel rods is oxidising the cladding. From the viewpoint of materials science, the zirconium-based alloy claddings are passivated, while in dry storage. Oxidation at temperatures below 250°C is therefore of no concern, because of the following reasons:

1. Cladding oxidation during reactor operation is very limited, since the zirconium-based alloys are specially developed to minimise corrosion
2. Corrosion is negligible given that the storage temperature is less than the in-reactor operation temperature
3. The cladding is fully passivated from the in-pile operation condition.

9.4.2.2. Oxidation of UO₂

Oxidation of UO₂ fuel in the presence of air or other oxidising atmospheres at elevated temperatures may lead to the formation of U₃O₈; this produces a considerable volume increase. U₃O₈ is the only U-O-phase with an O/U-ratio >2, which is characterised by a smaller density than UO₂, the oxide inserted as fuel in the fuel rods. This means that the oxidized fuel will swell only if the oxidation process results in the formation of U₃O₈. The volume increase of the oxidized fuel can result in a longitudinal split of the cladding.

Oxidation of irradiated UO₂ proceeds in a way different from oxidation of unirradiated UO₂ [66, 67]. U-O-phase diagrams, based on unirradiated phases, are not representative for irradiated U-O-phases. If unirradiated UO₂ is oxidized in air, a tetragonal U₃O₇ phase forms and continues to oxidize to U₃O₈. If irradiated UO₂ is oxidized in air at temperatures above 150°C, a cubic U₄O₉ phase forms. Further oxidation of that phase has only been observed at

temperatures above 250°C; and this oxidation appears to proceed directly to U₃O₈. As the burnup is increased, the fission products present in the UO₂ matrix stabilise the U₄O₉ phase [68].

9.4.2.3. Investigation of a spent fuel with an operational defect in the presence of air

To study the behaviour of fuel rods with operational defects, under dry storage conditions at 400°C, a test device for full size LWR-fuel rods was built and integrated into a hot cell. In a 16-day test, there was limited air access to the spent fuel rod at a temperature of 400°C. Only the fuel in the neighbourhood of the operational defect was oxidized. All ⁸⁵Kr from the oxidized fuel was released. Post-test investigations showed that no additional fission gas was contained in the oxidized fuel. The cladding was split in the area, where the fuel was oxidized to U₃O₈.

9.4.3. Water ingress

Water ingress raises, in theory, the potential for corrosion of the cladding, the FA components, and any liner material. As far as the FA is concerned, it is designed to operate at in-service reactor temperatures for several years. Therefore, for the case of unexpected water ingress into the dry storage confinement volume, the additional risk of corrosion can be regarded as negligible.

10. MONITORING TECHNOLOGIES AND TECHNIQUES

The purpose of monitoring is to determine the condition of the spent fuel, to verify or to validate assumptions related to the behaviour of components or materials in a storage system. It is undertaken to support one or more activities such as a new licence application, a condition of the existing operating licence, a statutory requirement, good operating practice or part of a continuous improvement programme.

Monitoring technologies have been developed to:

- Confirm fuel cladding integrity
- Confirm fuel component/geometric integrity
- Confirm the containment system integrity
- Confirm derived nuclear data
- Confirm shielding adequacy
- Support research and development activities
- Monitor environmental (radiological and ecological) conditions of the storage facility, and its surroundings

Where practicable non-intrusive/destructive techniques are preferred.

10.1. TECHNIQUES EMPLOYED

Techniques employed range from basic visual inspections (through the biological shielding wet or dry (cave)) to sophisticated radiometric systems (such as a burnup monitor).

Examples of commonly used techniques are briefly outlined here and summarised in Table XVIII and XIX.

Table XVIII. Summary of monitoring techniques and applications

Technique	Function	Containment	Shielding	Criticality	Operability	R&D Data Confirmation	Other Safety Functions
Pressure Monitoring		√				√	
Fission Gas Sampling		√					
Temperature Radiation Detection Systems			√		√	√	√
Liquid Sampling		√			√		√
Visual Inspections				√	√		√
Gas Sampling (non-active)						√	
In-pool Inspection Systems						√	√
Burnup				√		√	√
Ultrasonic Test		√			√		

10.1.1. Visual inspection (including closed circuit TV)

The primary purpose of visual inspection is for safety purposes (overall integrity of a fuel assembly and/or storage system) [69, 70]. The technique, however, is also utilised for fuel examination purposes, such as PWR top nozzle guide tube connection inspections.

10.1.2. Liquid sampling

This is a commonly used technique for monitoring individual fuel assembly integrity in wet storage through the analysis of liquor samples for soluble fission products, such as ^{137}Cs (sipping tests). The technique is also used for monitoring surface and underground water activity to demonstrate containment, or retrospectively, in combination with an activity release model, to assess bulk fuel integrity performance in wet storage.

10.1.3. Fission gas sampling

This is another method, similar to liquid sampling, for establishing fuel cladding integrity. This is achieved by sampling, for example a cask cavity, for the presence of ^{85}Kr , using a gas analyser. The technique has also been deployed within cask storage buildings, or at the site perimeter, for similar purposes.

Other uses of fission gas sampling are primarily for fuel performance testing (i.e. puncture test) in hot cell examinations.

10.1.4. Pressure monitoring

Pressure monitoring is usually carried out to support the demonstration of containment integrity in closed systems, such as casks, dry storage tubes, sealed cans or canisters. In the case of casks, several types of pressure tests can be undertaken, such as a hydrostatic head (usually underwater) and/or pressure drop measurement of lid seal inter-space. Pressure drop is also used to demonstrate that cavities or fuel are dry after loading.

Table XIX. Summary of current and future monitoring techniques and applications in each country

TECHNIQUE/FUNCTION	Canada		France		Germany		Hungary		Japan		Republic of Korea		Russian Federation		Spain		Sweden		UK*		USA	
	W	D	W	D	W	D	W	D	W	D	W	D	W	D	W	D	W	D	W	D	W	D
Pressure Monitoring	---	F	---	C	---	C	---	C	---	C	---		---	F	---	C	---		---		---	C
Fission Gas Sampling		F			C		C		C				C	F	C		C			C	C	
Temperature	C	C	C	C	C		C	C	C	C	C		C		C	C	C		C	C	C	
Radiation Detection Systems	C	C	C	C	C	C	C	C	C	C	C	C	C	F	C	C	F		C	C	C	C
Liquid Sampling	C	---	C	--	C	---	C	---	C	--	C	---	C	---	C	--	C	---	C	--	C	--
Visual Inspections	C		C	C	C		C		C		C	C	C	F	C	C	C		C	C	C	C
Gas Sampling (non active)		F		C			C		C					F			C		C			C
In-pool Inspection Systems	C	---	C	--	C	---	C	---	C	--	C	---		---	C	--	C	---	C	--	C	
Burnup			C	F					P	F	F	F	F			F	F		P			F
Ultrasonic Test			C						C		C				C				---			C

Note: W (Wet), D (Dry), C: Current, F: Future, P: Prior to reprocessing only,

* UK has plan to introduce corrosion sensor to wet storage in future

10.1.5. Gas sampling

Gas sampling (not for fission gas products) applications include helium leak testing for sealing integrity checks (cask systems) and to support development activities; for example, monitoring the control of radiolysis in cask or container interspaces.

10.1.6. Radiation detection instrumentation

Standard detection systems in every day use, mostly for demonstrating containment or shielding, include hand held and fixed monitors:

- Gamma monitors,
- Beta-in-air monitors,
- Neutron monitors.

10.1.7. Temperature measurement

This is a very basic technique, used in both wet and dry storage, for detection of loss of coolant. In ventilated cask storage monitoring, the air inlet/outlet temperature difference is used to verify that fuel cladding temperature is maintained well below allowable limits, and for confirming the heat removal capability of the system.

10.1.8. In-Pool fuel integrity inspection systems

There are a variety of techniques and systems available for monitoring the integrity of individual fuel assemblies in-pool. These are detailed in [71, 72].

10.1.9. Burnup monitors

This subject is covered in detail elsewhere [73, 74, 75, 76, 77, 78, 79, 80 and 81]. The primary function of a burnup monitor is to support criticality safety. To date, this approach has been mainly connected with reprocessing activities (partial credit for burnup in France's case, or for setting trip systems in the UK).

It is likely that burnup monitors will have a more important role to play in the future for enhancing fuel storage capacity, especially for highly enriched, high burnup and MOX spent fuels.

10.1.10. Ultrasonic test

This technique can be used in storage pools at reactor sites for detection of failed fuel rods. For monitoring of stored spent fuels, this technique is also applicable for periodical surveys of fuel integrity under water environment.

10.1.11. Corrosion sensors

There are number of electrochemical based techniques available for condition monitoring of materials deployed throughout the storage facility, and for monitoring changes in storage condition. Some of the techniques finding application include zero resistance ammetry (ZRA) [82], electrochemical noise and field signature method (FSM).

10.2. EXAMPLES OF MONITORING

10.2.1. Korean CANDU silo monitoring programme

1. Complete quarterly physical inspection of the spent fuel silo area (foundations, security fence, surveillance equipment, and silos). Particular attention is paid to any evidence of silo deterioration, e.g. excessive cracking, spallation of small pieces of concrete, etc.
2. Quarterly gamma monitoring of the silos with a hand-held detector. Two measurements are taken from representative sample points on the silos: one on contact at a point 1.9 m above the ground level, and the second at a distance of 1.9 m from the silo, perpendicular to the contact measurement point.
3. Continuous gamma monitoring of the area by thermo-luminescent dosimeters located at representative positions on the fence boundary.
4. On an as-needed basis, verification that the surface water around the site does not have an increased level of radioactive content attributable to the fuel radiological content.

As experience is gained over the years, the monitoring/inspection intervals may be modified.

Presently, the periods are: one year for silo surface inspection, three months for silo leak test and site ground water analysis, and one week for site surface water analysis. No problems have been encountered so far in connection with the system operation.

10.2.2. Ontario Power Generation Used Fuel Dry Storage (UFDS) Monitoring Program

The current Canadian strategy for managing used nuclear fuel includes dry storage for several decades prior to final disposal (geological isolation or alternative future technology). There is uncertainty regarding potential degradation of the stored fuel, which may have an impact on safety and future handling operations. A UFDS monitoring program will be implemented at all of the Canadian dry storage facilities. The program will be implemented first at the Pickering used fuel dry storage facility, where the used fuel from the eight Pickering reactors is to be stored. The program includes modelling of the fuel degradation processes under specific dry storage conditions, and instrumentation of four dry storage containers (DSCs), with the purpose of monitoring over the long term several container cavity parameters. The capabilities of the DSC monitoring system include: cavity gas sampling, temperature monitoring, pressure monitoring, and ^{85}Kr monitoring. Fuel examinations are being considered as part of a long-term program for verification of fuel condition.

The Pickering UFDS monitoring program includes design of a sophisticated system to monitor temperature at various points of the storage containers (including at the inner steel liner) as well as to monitor pressure and gas composition inside the container. The system will allow the extraction of gas samples for off-line analysis, as well as on-line monitoring of the ^{85}Kr concentration in the cavity. This measurement will be used as an indicator of the occurrence of cladding failures during dry storage.

The design of the monitoring system and minor design changes required for the test DSCs were being done during 2001. The system will be built, installed and commissioned at the Pickering site during 2002, following approval by the regulatory body (Canadian Nuclear Safety Commission). The monitoring methods to be used in this program are described in further detail in [80].

10.2.3. Inspection of leaking fuel in Sweden

Leaking assemblies are inspected visually to identify the position of the leak. Usually the leaking rod is removed and replaced by a dummy, or one of similar burnup, to allow further operation of the fuel.

If visual inspection is inadequate, Eddy Current testing of each rod is performed. Of course, this operation requires the dismantling of the assembly.

After the repair, the assembly is sipped by the box sipping method, to verify that it does not leak any longer. In this situation, caesium isotopes (or ^{85}Kr) are the only nuclides available for analysis.

10.2.4. Sipping of transport flasks, Sweden

10.2.4.1. Prior to shipment to the interim storage facility (CLAB)

When the flask is loaded, it is allowed to stand for a few hours. Water samples withdrawn from the pool water and the flask water, before and after this period, are analysed with respect to caesium isotopes to establish if there are any leakers.

10.2.4.2. Unloading at CLAB

When the transport casks, with the spent nuclear fuel, arrive at CLAB, the vacuum is broken and the assemblies are cooled down. If there are any leakers present, it is revealed by beta sensitive detectors measuring ^{85}Kr in the gas, from the cask cooldown procedure. If any shipment has indications of leaks, it is completely sipped in a box sipping Equipment.

10.3. REGULATORY REQUIREMENTS AND FUTURE DEMANDS

The monitoring requirements are either part of the operating/site licences, or a function in respect of the plant safety, and the ability to demonstrate safe operations at any point in time.

Annex

COUNTRY REPORTS ON RESEARCH PROJECTS WITHIN THE SPAR CRP

TITLE: Behaviour of power and research reactor fuel in wet and dry storage

COUNTRY: Canada

CHIEF SCIENTIFIC INVESTIGATOR: J. Villagran

AGREEMENT NUMBER: 9661 COMPANY: Ontario Hydro

BACKGROUND:

Canada has developed extensive experience in both wet and dry storage of CANDU fuel. Fuel has been stored in water pools at CANDU reactor sites for approximately 40 years, and in dry storage facilities for a large part of the past decade. Currently, Canada has 38 825 t U of spent fuel in storage, of which 4239 t U are in dry storage.

Canada has not selected a strategy for long-term management of its spent fuel yet. That decision will be made under the framework of a new nuclear waste legislation, which is expected to be enacted the next year. Because of this situation and also because of the transition process that Ontario Power Generation is undergoing, research programs have been re-prioritized.

OBJECTIVES:

The Canadian utilities and AECL are conducting development work in extended storage systems as well as research on fuel behaviour under storage conditions. Both activities have as ultimate objective to establish a technical basis for assuring the safety of long-term fuel storage.

RESEARCH APPROACH:

The studies currently under way are:

- Assessment of the service life of existing dry storage systems (LEX).
- Direct disposal in a deep geological repository (DGR) in the Canadian Shield.
- Reactor-site extended storage (RES).
- Centralized extended storage (CES).
- Spent fuel transportation systems (TRA).
- Modelling of fuel degradation mechanisms under dry storage conditions (IFS-01).

Other studies expected to start within the next two years include:

- Deep geological repository in alternative geological media (e.g. clays or shale).
- Strategy for partitioning and transmutation.
- Assessment of long-term fuel evolution in dry storage.
- Long-term monitoring of storage parameters in dry storage containers.

A plan for long-term monitoring of four dry storage containers at the Pickering facility was presented at the initial SPAR meeting in Washington, in 1998. However, as a result of the changes in work priorities at OPG this program has been postponed and it will not be initiated until 2002 or 2003. Therefore, monitoring results will not be available for some time.

TITLE: Dry storage performance assessment of spent LWR-FA with increasing burnup

COUNTRY: Germany **CHIEF SCIENTIFIC INVESTIGATOR: M. Peehs, W. Goll**
AGREEMENT NUMBER: 9663 **COMPANY: Siemens AG.**

BACKGROUND:

Back end of fuel cycle costs represent about a half of the total fuel cycle costs. To reduce the costs for electricity generation the burnup of the spent LWR-FA is increasing. The decrease of the spent fuel arising involves a direct decrease of back end of fuel cycle costs. Direct disposal of spent fuel is a reliable strategy to close the fuel cycle. Direct disposal is preceded by long-term dry storage. Therefore, dry storage performance assessment of spent LWR-FA with increased burnup supports selecting of most suitable storage conditions.

OBJECTIVE:

- Review of the current and future burnup of spent fuel in Germany.
- Generate a typical set of EOL data describing a spent LWR-FA.
- Review typical FR defect mechanisms, which might occur under the aspect of increasing burnup.
- Perform dry storage performance predictions.

RESEARCH APPROACH:

To assess the extended storage performance of spent LWR-fuel, the available experience can be collated into 3 storage modes:

- **Mode I:** fast decrease rate of temperature between maximum dry storage temperature and 300°C.
- **Mode II:** medium decrease rate of the fuel rod dry storage temperature between 300°C and 200°C.
- **Mode III:** slow to negligible decrease rate of fuel rod dry storage temperature for temperatures > 200°C.

Mode I is typical for early interim storage and fuel rod creep is expected to occur. **Mode II** dry storage is characterised by the fact that all creep deformations of the spent fuel cladding can already be regarded as terminated as well as the corrosive attack of the cladding. **Mode III** covers extremely long-term storage, which is encountered presumably for nearly all, dry storage extensions to be considered.

The results from the investigations teach that cladding creep from inner gas overpressure is the rate determining degradation and failure mechanism for setting maximum allowable storage temperature limits in dry inert storage. It is desirable to avoid any degradation. This can be accomplished by confining the creep degradation mechanism to its primary and early secondary stages. This approach is in correspondence with standard creep engineering practices. This methodology relies on the availability of:

- A database of cladding creep.
- A correlation that allows predicting post-pile creep from creep of unirradiated material.
- A database of the potential total strain of a fuel rod cladding under dry storage conditions.
- A numerical model to predict creep strain during dry storage.

RESULTS:

As an example a CASTOR V/19 is considered loaded with 19 LWR FA with UO₂ as nuclear fuel having a maximum rod burnup of 55 GW·d/t HM or 15 LWR FA with UO₂ as nuclear fuel and 4 LWR FA with UO₂/PuO₂ as nuclear fuel both having a maximum rod burnup of 55 GW·d/t HM. Additionally we look at 2 types of fuel rod claddings to assess the influence of different cladding material.

Case	1	2	3	4	units
Cladding	fast creeping clad		less fast creeping clad		
Burnup (rod)	55	55	55	55	GW·d/t HM
Fuel type	UO ₂	UO ₂ /PuO ₂	UO ₂	UO ₂ /PuO ₂	
Pond storage	5.6	6.5	5.6	6.5	years
Decay heat/cask start	32.3	37.5	32.3	37.5	kW at start
FR hot spot temp.	348	357	348	357	°C

In calculating EOL-conditions for reactor operation, the different in-service behaviour of both cladding types has been considered. Fast creeping cladding contacts due to the creep down under the external coolant pressure the oxide fuel earlier than strong cladding. As a result of the earlier closed fuel cladding gap the fuel temperature and consequently the fission gas release is less than for the stronger cladding. Therefore at the beginning of dry storage the internal fuel rod pressure in the cases 1 and 2 is less than for cases 3 and 4. The hot spot temperature of a fuel rod in the CASTOR V cask was between 348°C (MOX-FA) and 358°C (U-FA) at the beginning of storage. The highest hoop strain predicted — all occurring in the mode I — is 0.77%. This result proves that dry storage is safe for LWR fuel of such burnup also for time periods exceeding the presently considered time period of 40 years by far.

TITLE: Assessment of spent WWER-440 fuel performance under long-term storage conditions

COUNTRY: **Hungary**
CONTRACT NUMBER: **9664**

CHIEF SCIENTIFIC INVESTIGATOR: **F. Takáts**
COMPANY: **TS Enercon Kft.**

BACKGROUND:

Paks Nuclear Power Plant is the only NPP in Hungary. It has four WWER-440 type reactor units. The fresh fuel is imported from the Russian Federation so far. The spent fuel assemblies were shipped back to the Russian Federation until 1997 after about 6 years cooling at the plant. A dry storage facility (MVDS type) has been constructed and is operational since then. By 1 January 2001, there were 1980 assemblies in dry storage.

OBJECTIVE:

- Wet AR storage of spent fuel from the NPP Paks:
 - Measurements of conditions for spent fuel storage in the at-reactor (AR) storage pools of Paks NPP (physical and chemical characteristics of pool water, corrosion product data)
 - Measurements and visual control of storage pool component characteristics
 - Evaluation of storage characteristics and conditions with respect to long-term stability (corrosion of fuel cladding, construction materials).
- Dry AFR storage at Paks NPP:
 - Calculation and measurement of spent fuel conditions during the transfer from the storage pool to the modular vault dry storage (MVDS) on the site
 - Calculation and measurement of spent fuel conditions during the preparation of fuel for dry storage (drying process), such as crud release, activity build-up
 - Measurement of spent fuel conditions during the long-term dry storage, activity data in the storage tubes and amount of crud
 - Benchmarking the COBRA-SFS computer model to predict spent fuel dry storage parameters, adaptation of the code to WWER fuel and MVDS geometry.

RESEARCH APPROACH:

Aiming to evaluate physical and chemical conditions in the storage pools the following parameters are monitored: temperature ($^{\circ}\text{C}$), pH, optical transparency (%), chemical composition (Cl, H_3BO_3), isotope-specific activity (Bq/kg).

The overall limitations ($t < 70^{\circ}\text{C}$ in case of refuelling operations, $< 60^{\circ}\text{C}$, in case of storage; $\text{H}_3\text{BO}_3 > 12 \text{ g/kg}$) are continuously controlled. Chemical conditions are measured and corrected if necessary minimum daily during refuelling operations, and weekly during storage, except the boron concentration monitored continuously.

As routine measurements the following activity data are measured:

- Corrosion products (^{51}Cr , ^{54}Mn , ^{59}Fe , ^{58}Co , ^{60}Co , $^{110\text{m}}\text{Ag}$)
- Volatiles (^{131}I , ^{132}I , ^{133}I , ^{134}I , ^{135}I)
- Non-volatile components (^{137}Cs , ^{134}Cs , ^{138}Cs , ^{139}Ba , ^{140}Ba , ^{91}Sr , ^{92}Sr , ^{85}Rb , ^{88}Rb , ^{89}Rb)
- Other components (^{187}W , ^{56}Mn , ^{99}Mo , ^{122}Sb , ^{124}Sb , ^{24}Na , ^{42}K , ^{95}Nb , ^{95}Zr , ^{97}Zr , ^{140}La).

The measurements of activity are carried out — as a rule — once in month, during refuelling measurements they are done more frequently.

RESULTS

The conclusions drawn are similar to those of the earlier research timeframe, namely:

- On the basis of the measured chemical characteristics, the water quality remains in stable condition. The pH values vary between 4.5 –6.2, the lowest value of optical transparency is 95.4 %. The activity levels can be managed safely using the pool water purification system.
- The main sources of activity in the storage pools are surface and water contaminants. Corrosion during storage period is negligible.
- Dry storage experience is good; temperature of the stored fuel is low, as confirmed by the benchmarked calculations. There were some storage tubes reopened and no unexpected observations were made.

Preparations are made for dry storage of spent fuel with surface deposits.

TITLE: Integrity of spent fuel during storage and situation of burnup credit development

COUNTRY: Japan

CHIEF SCIENTIFIC INVESTIGATOR: T. Matsumura

AGREEMENT NUMBER: 9665 COMPANY: CRIEPI

BACKGROUND:

Advanced spent fuel storage technology is necessary for high burnup UO₂ and mixed oxide (MOX) spent fuel elements coming from nuclear power plants in the near future. There is few data related to spent fuel behaviour, especially during dry storage. It is therefore necessary to obtain the data of the fuel integrity during storage by PIE. An introduction of burnup credit to the storage facilities is also important to reduce storage cost and cope with the increasing spent fuel.

OBJECTIVES:

- To review questions related to the storage of high burnup UO₂ and MOX spent fuel.
- To perform a systematic review of criteria for the use of the burnup credit.

RESEARCH APPROCH:

- The post irradiation examinations on the twenty-years-stored spent fuels were carried out to evaluate fuel integrity during storage.
- “The Nuclear Criticality Safety Handbook of Japan” was published in 1988 by JAERI (government) and following the 2nd edition of the “Handbook” was published in 1999.
- The effect of reactivity biases on cask reactivity was evaluated for spent fuel transport and storage casks.

RESEARCH RESULTS:

- The results of the post irradiation examinations for twenty-years-stored spent BWR-MOX fuel (20 MW·d/kg U, wet and dry condition) and PWR-UO₂ fuel (58 MW·d/kg U, dry condition) showed no additional gas release and no remarked difference in oxide thickness, hydride distribution and mechanical properties in cladding, and fuel pellet after storage.
- “The Nuclear Criticality Safety Handbook of Japan” was used for criticality safety design, in which burnup credit was considered, in the Rokkasho Reprocessing Plant.
- The introduction of burnup credit has a large merit in criticality safety analysis of casks even if the reactivity biases owing to burnup/void fraction profile, operational histories and so on are considered. Additionally, the concept of equivalently uniform burnup was developed to generalize and simplify the reactivity bias evaluation in burnup credit.

TITLE: Long-term storage behavior of defective fuels

COUNTRY: Republic of Korea **CHIEF SCIENTIFIC INVESTIGATOR: S.W. Park**
AGREEMENT NUMBER: 9666 **COMPANY: KAERI**

BACKGROUND:

Long-term storage safety of spent nuclear fuel is very important for continuing nuclear power generation. The behavior of defected fuel, especially, has been a key factor and its oxidation behavior has been considered as the most important one for dry storage. For wet storage, PIE techniques will be helpful for detection of rod characteristic changes due to defects. A storage basket for defective fuel assembly will be useful to avoid the pool water contamination.

OBJECTIVES:

- Investigation of oxidation behavior of fuels under dry storage condition
- Surveillance on behavior of defective fuels under pool storage condition
- Fabrication of a basket for defective fuel assembly

RESEARCH APPROACH:

- The oxidation kinetics of unirradiated- UO_2 was investigated and the morphology changes and U_3O_8 formation time were observed during oxidation.
- To understand the behavior of defective PWR spent fuel during pool storage, two rods having 12-year storage duration were examined by PIE techniques.
- The filter design for a storage basket of defective PWR fuel assembly was calculated by the COBRA-SFS and ORIGEN-II computer codes.

RESEARCH RESULTS:

- Oxidation tests on unirradiated- UO_2 were performed at 150–300°C. At this temperature range the activation energy and powdering time can be expressed as follows:
 - Activation energy: 145 kJ/mole.
 - Powdering time: $\log t \text{ (h)} = 12.89 + 7650/T \text{ (K)}$.
- There were no indications of any dimensional changes and eddy current signal changes at the defected positions. The yellow material, which was found previously on another defective rod, was not identified in these rods.
- The calculation for a storage basket filter shows that 200 g of Zeolite-A and 400 g of Chabazite would be necessary for filtration of the ions from defective fuels.

TITLE: Assessment of spent fuel during wet storage

COUNTRY: Russian Federation CHIEF SCIENTIFIC INVESTIGATOR: **T. Makarchuk,
N. Tikhonov,
N. Kalyazin**

AGREEMENT NUMBER: 10895 COMPANY: **ICC “Nuclide”, VNIPIET, LNPP**

BACKGROUND:

In the Russian Federation the wet storage technology will most likely retain its dominating role for the next several decades. In view of this perspective the determination of potential fuel degradation mechanisms in relevance with storage becomes actual. The results may be used in assessments of permissible time limits for safe extended long-term storage. The fuel cladding of Russian power reactor fuel is fabricated from Zr1Nb alloys; dioxide uranium is used for the fuel composition.

OBJECTIVE:

- The determination of potential fuel degradation mechanisms during long term storage of spent fuel
- Assessment of defective fuel condition at water pool storage.

RESEARCH APPROACH:

Studies of RBMK-1000 and WWER-440 spent fuel condition were carried out after 8 to 15-year storage in the water pool of Leningrad and Novo-Voronezh NPPs.

The assessment of defective fuel behaviour was made on the basis of water radiochemistry results, which were measured on the storage facility of SF of Leningrad NPP.

RESULTS:

As it is evident from the foregoing analysis the spent fuel assembly is exposed to multiple impacts, related to fuel fabrication technologies, fuel operation and storage in water pools. Since the fuel cladding functions as the primary barrier against radioactivity release to the environment, the cladding integrity is the main guarantee for long-term storage.

Physic-chemical processes taking place in the cladding material and eventually causing its degradation are related to fuel fabrication defects, power output, fuel operation condition and corrosion processes which occur in the reactor and continue in storage pool.

According to the data obtained the mechanism of fuel rod damage is the interaction between the rod cladding and debris. The corrosion behaviour of fuel cladding material was characterised by the thickness of uniform oxide film, nodular corrosion depth and still “alive” cladding section determined by optical metallography. The uniform corrosion rate of wet storage varies within 3–5 μ /year depending on the burnup depth and storage time. For the major portion of FAs the corrosion film thickness on the cladding surface was 10–40 μ . However, some FAs of RBMK-1000 developed significant nodules and one FA demonstrated fretting corrosion in the vicinity of the spacer grids.

The investigations of the most extensively damaged cladding show surface nodules of 40 μ depth, and under the spacer grids even nodules of 150 μ m depth were found. With burnup increase to 19.3 MW·d/kg U these nodule depths also grew to 130 and 380 μ , respectively.

Some cladding specimens demonstrated local thinning of 400 μ depth, in the vicinity of the spacer grids, which was probably due to fretting corrosion. These nodule depths and fretting wear may significantly decrease the cladding mechanical strength. Hydrogen uptake by the fuel cladding material was insignificant ($(3-15) \times 10^{-3}$ % mass) and relates to cladding wear. The tensile tests on 3 mm high annular samples were fulfilled with a tensile-testing machine at 20°C and 350°C and the cross-head speed of 3 mm/min. The strength properties were calculated with nominal cladding thickness 0.9 mm. The testing results show rather high strength and plastic properties of investigated claddings in spite of their significant wear.

The studies at the Leningrad NPP show that the long-term storing of RBMK-1000 spent fuel assemblies in cans with the stagnant water (the water quality in these are significantly worse than in the pool) may produce effects upon the fuel condition. The results of the study confirm that the water activity of loaded cans, dynamics of activity change and radionuclides composition may serve reliable criteria for assessing cladding integrity condition and may be used for fuel integrity monitoring purposes during long-term storage of this fuel type. The results of the study were used for justification of the above criteria before cutting RBMK assemblies into individual fuel bundles in hot cell and subsequent overloading into a metal-concrete cask for extended dry storage.

TITLE: Dry storage performance assessment of spent LWR fuel assemblies

COUNTRY: Spain

CHIEF SCIENTIFIC INVESTIGATOR: J.A. Gago

AGREEMENT NUMBER: 9668 COMPANY: ENRESA

BACKGROUND:

Current back-end policy in Spain accounts for direct disposal of spent fuel at a geological repository, after a few decades of interim storage, using both wet and dry technologies. To support the use of the latter one, especially for high burnup fuel, a research programme was initiated to assess the behaviour of zircaloy fuel cladding, which had been stored in a dry oxidising atmosphere for more than 20 years.

OBJECTIVE:

- Review of spent fuel data relevant to its future storage in Spain
- Perform non-destructive and destructive examinations on high burnup fuel rods
- Review typical defect mechanisms, which might occur with increasing burnup.

RESEARCH APPROACH:

Six high burnup rods irradiated to burnups in the range of 60 MW·d/kg U and subsequently dry stored in air for more than 25 years, were subjected to a non-destructive programme consisting in:

- Visual examination
- Cladding oxide thickness measurement
- Eddy Current testing
- Total Gamma scan.

Two of those rods were subjected to a destructive programme comprising:

- Fission gas release measurements on two rods
- Optical metallography (including a brief consideration of fuel microstructures)
- EPMA to establish radial fission product profiles and the composition of bonding phases at the fuel/cladding interface
- Cladding mechanical tests (tube burst tests and tensile testing).

RESULTS:

The following are the main results obtained from the non-destructive examination programme:

- All rods covered in a thin base oxide layer
- Rods covered in a layer of nodular corrosion (increased around the grid band region)
- Axial scratches filled with oxide (caused during rod insertion into the assembly)
- Average growth corrected at 293°K of around 0.4%
- Five rods were free of any signal indicative of cladding defects
- One rod (57.2 MW·d/kg HM) showed an indication
- All rods show some primary and secondary ridging, more important in the high burnup ones
- Results from the cladding oxide thickness measurements were accordance with previous measurements, revealing no major changes in the cladding condition following 25 years of dry storage.

The following conclusions could be derived from the destructive examination programme:

- Fission gas release measurements made on two rods showed gas analysis results as expected.
- Cladding inner and outer surface oxidation was measured by optical metallography, as well as the hydride distribution within the cladding. A brief examination of the fuel microstructure was also performed, revealing structures similar to those in previously examined high burnup water reactor fuel, including a ‘rim’ region of 100–140 μm thickness.
- In the EPMA studies, the two samples studied gave very similar composition results. Most fission products (Pu, Nd, Zr, Mo, Ru) had a flat radial profile with an enhanced concentration close to the rim associated with the high burnup in this region.
- The fuel/cladding bonding layer was composed primarily of zirconium and oxygen, with traces of fission products. The oxygen content of this region was high relative to the fuel. The fuel at the edge of the bonding layers showed enhanced levels of fission products associated with the higher burnup rim region. Cladding analysis showed the composition to be primarily zirconium with small amounts of fission products compared to the bonding layer.
- Of the three samples subject to tube burst testing, two gave very similar burst pressures (around 137 Mpa), while the third was lower at 115 MPa. Two further samples were also subject to tensile testing, showing generally similar results.

The results obtained in the research programme show in overall a good performance of the fuel cladding in an air atmosphere at relatively low temperatures.

TITLE: Interim wet storage of advanced gas reactor fuel up to 80 years

COUNTRY: UK

CHIEF SCIENTIFIC INVESTIGATOR: P. Standring

AGREEMENT NUMBER: 10285 COMPANY: SFEG BNFL

BACKGROUND:

Spent nuclear fuel has been interim stored in wet storage ponds, prior to reprocessing on the Sellafield site since the 1950s. Given the availability of oxide reprocessing facilities some of this fuel will be up to 35 years cooled prior to reprocessing. Such examples present us with the opportunity to assess the impact of long-term wet storage on fuel assembly condition when retrieved for reprocessing. In addition to the retrieval of long stored spent oxide fuel, materials development programmes have been initiated to provide supporting data to safety case development for the wet storage of AGR fuel for up to 80 years.

OBJECTIVES:

- To report on the condition of long wet stored fuel retrieved for reprocessing
- To investigate the transpassive dissolution mechanism of 20/25 stainless steel cladding in caustic dosed water to pH 11.5
- To determine the rate of corrosion 20/25 stainless steel cladding in caustic dosed water to pH 11.5
- Development of a corrosion sensor to support long-term wet AGR fuel storage.

RESEARCH APPROACH:

The condition of long stored LWR fuel is routinely monitored during fuel handling operations in the LWR storage pond and Thorp Head End Feed Pond using CCTV and video recording.

The materials studies replicating the original AGR cladding corrosion inhibitor development work and determination of AGR cladding corrosion rate in caustic to pH 11.5 were undertaken on unirradiated cladding specimens using a mixture of electrochemical and traditional techniques. The supplementary corrosion sensor development programme has been initiated using a zero resistance ammetry technique.

RESEARCH RESULTS:

- The condition of long-stored LWR fuel has generally been found to be good when retrieved for reprocessing. There is no evidence to suggest that there has been any fuel assembly degradation as a result of the wet storage regime employed. This observation is supported through the repeated examination over many years of a fuel assembly with known top nozzle/guide tube connection corrosion; which showed no deterioration in condition.
- The transpassive dissolution of the protective chromium-rich oxide film on 20/25 stainless steel cladding reported to occur in caustic dosed deionised water to pH 11.5, was not observed in the latest investigations. The current investigations conclude that the original observations were due to a solution oxidation reaction associated with hydrogen peroxide and not metal loss as originally reported.
- The rate of corrosion of 20/25 stainless steel cladding in caustic dosed deionised water to pH 11.5 has been determined by the traditional method of long-term immersion testing and the newer Field Signature Method. A corrosion rate of <0.1 $\mu\text{m/a}$ has been concluded to date.

- The scoping trial of a corrosion sensor for 20/25 stainless steel based on zero resistance ammetry has been performed. The prototype sensor was found to replicate the behaviour of reactor-sensitised AGR braces in that:
 - Corrosion initiated in 3 mg/l Cl⁻, observed sharp increase in current;
 - Increasing the pH to 11.5 retarded the corrosion, but did not inhibit it fully;
 - Increasing the pH to 12 improved the inhibition.

TITLE: US spent fuel research and experience

COUNTRY: USA

**CHIEF SCIENTIFIC INVESTIGATORS: A. Machiels,
L. Stewart**

AGREEMENT NUMBER: 9669 COMPANY: EPRI and USDOE

BACKGROUND:

The USA has developed broad experience in both the wet and dry storage of LWR fuels. Some 15 sites now have commercial dry storage facilities in operation. The current focus of R&D in the USA is to increase the efficiency and flexibility of the storage packages and to improve on the time and cost of loading the various storage systems. In light of the delays in the startup of the final DOE repository system, it is also of great importance to develop the technical basis for assuring the safety of long-term dry storage.

OBJECTIVE:

The objectives of the US research program included the following:

- Stainless steel cladding performance in dry storage
- Application of burnup credit to storage and transport casks
- Multipurpose Canister System Synopsis report
- Dry transfer demonstration program
- Data and studies necessary to support long term dry storage
- Wet and dry storage experience and data in the USA.

RESEARCH APPROACH:

Many research projects have been initiated under the sponsorship of EPRI to address the objectives listed above. In several cases, funding and program direction have been shared with the USDOE and in the case of the Dry Transfer Demonstration program, DOE took the lead role. The research was contracted out to vendors, national laboratories and utilities and the results published in EPRI and/or DOE reports.

RESEARCH RESULTS:

- A detailed engineering evaluation of the potential failure mechanism for stainless steel clad fuel in dry storage was performed and the results published in TR-106440, April 1996. It was concluded that the limits currently being applied to zirconium clad spent fuel are also appropriate and adequate for stainless steel clad fuel.
- The work toward applying burnup credit in storage and transportation was slowed by the pacing function of interacting with the NRC regulators. While some progress has been made, an agreement on the necessary licensing approach to obtain meaningful burnup credit has not been achieved. As a result, no defining report has been issued.
- The concept of using multipurpose canisters in the USA has been an illusive target. Dual-purpose canisters for storage and transportation are now widely in use, but true multipurpose canisters that would include the canister being used for disposal has not been practical to develop in the current commercial structure of the US nuclear industry. EPRI has documented the evolution of the multipurpose canister in the report TR-108736, September 1997. Additionally, the report TR-106962, September 1997 presents the technical basis and design synopsis for the most advanced multipurpose system developed by the US DOE.

- The Dry Transfer Demonstration Program, a joint effort of the US DOE and EPRI, has been completed and the results documented in TR-105570, December 1995.
- Developing the technical basis for long-term dry storage is an ongoing effort in the USA. In April 1998, an EPRI report, TR-108757, documented the data that was needed to support the licensing of long term storage. Subsequently, a program was initiated using the fuel in storage at INEEL to gain the majority of this data. An interim report documenting the work done to date under this new program was published in an EPRI report 1000157 in June 2000.

The US experience in both wet and dry storage continues to be good. Several incidents have occurred in the commercial implementation of dry storage, which have resulted in some changes in operational procedures and some minor design changes. None of the incidents has resulted in injuries or unusual exposures to workers or the public. Dry storage technology has matured considerably and is now part of routine operation at many nuclear plants in the USA.

ABBREVIATIONS

AFR	away from reactor storage
AR	at reactor storage
AGR	advanced gas cooled reactor
BWR	boiling water reactor
CANDU	Canadian deuterium–uranium reactor
CC	concrete canister
CLAB	Central Interim Storage Facility for Spent Nuclear Fuel (Sweden)
Crud	A deposit on fuel assembly surface (in SPAR context)
DSC	dry storage container (Canada)
FA	fuel assembly
FR	fuel rod
HLW	high level waste
LWR	light water reactor
MAGNOX	Magnesium no Oxidation (Magnesium Alloy Cladding, UK)
MEB	multi-element bottle (UK)
MOX	mixed oxide fuel
MVDS	modular vault dry storage
NGS	nuclear generating station
PWR	pressurized water reactor
RBMK	Russian type of channel reactor, water cooled, graphite-moderated (reaktor bolshoy moschnosti kipyaschiy)
SF (A)	spent fuel (assembly)
SNF	spent nuclear fuel
SS	stainless steel
t HM	tons of heavy metal
THORP	thermal oxide fuel reprocessing plant (Sellafield, UK)
WWER	Russian type of PWR (vodo-vodyanoi energeticheski reaktor)
Zr1Nb	zirconium — niobium alloy fuel cladding (WWER)
Zry	Zircaloy

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