IAEA-TECDOC-944

Further analysis of extended storage of spent fuel

Final report of a Co-ordinated Research Programme on the Behaviour of Spent Fuel Assemblies during Extended Storage (BEFAST-III) 1991–1996

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The originating Section of this publication in the IAEA was:

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

FURTHER ANALYSIS OF EXTENDED STORAGE OF SPENT FUEL
IAEA, VIENNA, 1997
IAEA-TECDOC-944
ISSN 1011–4289

Printed by the IAEA in Austria
May 1997

FOREWORD

Considerable quantities of spent fuel continue to be produced and to accumulate in a number of countries Although some new reprocessing facilities have been constructed, many countries are investigating the option of extended spent fuel storage prior to reprocessing or fuel disposal Wet storage continues to predominate as an established technology However, dry storage is becoming increasingly used with many countries considering dry storage for the longer term

This Technical Document is the final report of the IAEA Co-ordinated Research Programme on the Behaviour of Spent Fuel Assemblies During Extended Storage (BEFAST-III, 1991-1996) It contains analyses of wet and dry spent fuel storage technologies obtained from 16 organizations representing 13 countries (Canada, Finland, France, Germany, Hungary, the Republic of Korea, Japan, the Russian Federation, Slovakia, Spain, Sweden, the United Kingdom and the USA) which participated in the co-ordinated research programme as participants or observers

The report contains information presented during the three Research Co-ordination meetings and also data which were submitted by the participants in response to requests by the Scientific Secretary

The IAEA wishes to thank the BEFAST-III Chairman, E Vitikainen (Technical Research Centre of Finland), as well as all the programme participants, observers and the consultants The Scientific Secretary was F Takáts of the Division of Nuclear Power and the Fuel Cycle

EDITORIAL NOTE

In preparing this publication for press, staff of the IAEA have made up the pages from the original manuscripts as submitted by the authors The views expressed do not necessarily reflect those of the governments of the nominating Member States or of the nominating organizations

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1. INTRODUCTION AND OBJECTIVES

This publication is the final report of the IAEA BEFAST-III co-ordinated research programme (CRP) — a continuation of the previous BEFAST-I and -II. The programme had 15 participants from Canada (2), Finland, France, Germany, Hungary, Japan, the Republic of Korea, the Russian Federation (2), Slovakia, Spain, the UK (2) and the USA. There was also an observer from Sweden.

The primary task for each country within the programme was to report on national spent fuel experience and contribute data to build a comprehensive international database. The second objective was to exchange current operating experience in spent fuel storage.

2. HISTORY OF THE BEFAST CO-ORDINATED RESEARCH PROGRAMMES

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 programme (1981–1986) involved 12 organizations from 11 countries: Austria, Canada, Czechoslovakia (CSSR), Finland, Federal Republic of Germany (FRG), German Democratic Republic (GDR), Hungary, Japan, Sweden, USA, and USSR. A subsequent programme, 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, Republic of Korea, Japan, UK, USA and USSR.

During these two co-ordinated research programmes (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, 2].

On completion of the second CRP, the future activities in the field of spent fuel storage were appraised and a new CRP – BEFAST–III – was suggested. The objectives of the CRP were to collect and exchange spent fuel storage experience of participating countries to build a comprehensive international database and to carry out research work which would evaluate the storage of spent fuel for extremely long periods of time (more than 50 years). The scope of BEFAST-III included both wet and dry storage for all types of spent power reactor fuel.

Three research co-ordination meetings (RCMs) were held during the course of the third CRP: the first in October 1992 in Toronto, Canada; the second in April 1994 in Manchester, UK; and the third in October 1995 in Paris, France.

Major topics for both wet and dry storage during all three CRPs are summarized in Table I. It records detailed objectives and the shift in emphasis during the various phases of these programmes between 1981 and 1995.

TABLE I BEFAST RESEARCH SUBJECTS

Long-term behaviour	B I	B-II	B-III	Surveillance	B-I	B-II	B-III	Facilities & Operation	B I	B II	B III
• Material aspect (cladding & components)	X			• Monitoring, - environment - components - fuel assemblies - workers dose rate			X	• Capacity enhancement - high density racks - reracking - double tiering - doped coolant - rod consolidation		X	
• Degradation mechanisms and models	X	X		• Fuel conditions - operational - fabrication - technology - defective fuel rods and assemblies		X	X	• Changing modes wet-dry		X	X
• Validation - experimental - experience		X	X	• Different reactor types		X	X	• Handling of heavily damaged fuel		X	X
								• System performance	X	X	X

3. CURRENT AND FUTURE SPENT FUEL STORAGE ASPECTS, WET AND DRY STORAGE

3.1. WET STORAGE

Wet storage continues to be the primary method for storing most LWR fuel. This technology has proven to be extremely well suited in meeting the shielding and cooling requirements of spent nuclear fuel.

For zirconium alloy clad fuel, data exists for continuous pool storage of greater than 30 years. This data indicates cladding corrosion to be extremely low (virtually not measurable) and therefore corrosion of the cladding is not viewed to limit storage below 100 years. Pool storage for MAGNOX fuel is good under controlled conditions, but is generally restricted to interim storage lasting only a few years. AGR fuel (niobium stabilized stainless steel cladding) has been successfully stored in water pools for up to 17 years. Care must be taken, however, to avoid the risk of cladding corrosion through sensitization of the stainless steel. This risk has been minimised in the UK by the use of the corrosion inhibitor sodium hydroxide since 1986 [3–6].

3.1.1. Spent fuel behaviour and storage experience

3.1.1.1. General performance

Virtually all light and heavy water reactors discharge spent fuel into a water filled pool. Only water offers a practical means to provide the demanding cooling and shielding requirements of freshly discharged fuel. For LWR reactors that use zirconium alloy clad fuel, pool storage times appear technically unlimited. The only limitation related to pool storage may be the capacity of the pools, or economic considerations that may dictate the closure of the pool at the time of a reactor shutdown. In these cases, dry storage technologies may be implemented or the fuel can be moved to a centralized pool storage facility that may be located off-site.

Vast quantities of zirconium alloy clad fuel now reside in pool storage. As of 1 January 1996 in the USA alone, more than 31 200 Mg U resides in wet storage. France now has 16 700 Mg U in both at-reactor (AR) and away-from-reactor (AFR) storage pools. Spent fuel currently stored has been discharged by reactors of the French utility EDF or comes from foreign countries holding French reprocessing contracts with COGEMA. Germany, the Republic of Korea and the UK in Sellafield have, respectively 3000, more than 2000, and 6200 Mg U. Spain and Sweden have around 1500 Mg U and 2200 Mg U of fuel in wet storage. Additionally, Canada has some 21 000 Mg U of Zircaloy clad CANDU fuel in wet storage, while Russia has about 6500 Mg U of RBMK-1000 fuel and 3300 Mg U of WWER fuel in storage pools (both reactor types use Zr-1%Nb clad fuel). Japan has 4600 Mg U of Zircaloy clad fuel from PWR, BWR and ATR reactors in wet storage. In all cases, where water chemistry has been adequately controlled, the performance of the zirconium alloy clad fuel has been excellent.

The most important degradation mechanism in water storage is cladding corrosion. Corrosion data collected over the past 40 years, including that associated with fuel in continuous storage for 33 years, indicates that the zirconium alloy oxidation rate at normal pool conditions is near zero. This result agrees with theoretical predictions based on Arrhenius-type aqueous corrosion. Corrosion of Inconel and stainless steel fuel assembly components in storage has also been shown to be negligible [7–8]. However, more attention has to be given to fuel structure component materials of different compositions in order to avoid spent fuel end fitting corrosion.

Other mechanisms that may affect fuel assembly integrity during wet storage are less important. Internal fuel pressure is relatively low at pool temperature and cladding creep is negligible. Both the micro and macro distribution of hydrogen in the cladding is also stable at pool storage conditions [7, 9]. In summary, there are neither observed nor theoretical mechanisms that have been identified which would limit the time zirconium alloy clad fuel could remain in pool storage, assuming pool chemistry

is properly controlled Examinations of RMBK spent fuel stored in the open canisters have shown that increased concentrations of chloride (up to 1 mg/kg) and hydrogen peroxide (up to 3 mg/kg) initiate corrosion of Zr-1%Nb cladding in the stainless steel spacers' location This effect can be minimized by the addition of $Ca^{+}CaCO_3$ mixture as a corrosion inhibitor

With 18 reactors currently using MOX fuel in Europe, the MOX industry is gaining increasingly more experience In interim storage, MOX spent fuel behaviour is similar to that of UO_2 spent fuel MOX fuel end-of-life conditions are similar to UO_2-fuel regarding cladding corrosion and internal rod pressure Spent MOX fuel, however, has a higher decay heat and radiation level Therefore, some extended wet storage is recommended The major licensing criteria such as allowable total strain is the same for UO_2 and MOX fuel Higher decay heat enhances the temperature related strain Consequently, longer wet storage is helpful

3 1 1 2 Country specific examples

The following paragraphs provide representative examples of the application of wet storage by major fuel types

Canada

Canada has been storing fuel under water since the early 1960s Examination of that fuel has confirmed its excellent storage behaviour

Eastern European countries

Behaviour in storage of the Zr-1%Nb clad WWER and RBMK fuel has also been excellent Over a 25-year period, there have been no known fuel assembly failures under normal storage conditions Corrosion studies have been conducted on Zr-1%Nb alloy cladding specimens from RMBK and WWER spent fuel assemblies that experienced a different burnup and cooling times Results of this work showed that burnup has a more significant influence on the cladding plasticity than storage duration

For RMBK fuel, the maximum oxide-layer growth rates of 3 to 5 μm/year has been observed in the spacer contact area of the spent fuel stored in open canisters with uncontrolled chemistry for the first 5 years of storage

France

In France, AR spent fuel storage is limited to a short period because the back-end policy is based on prompt reprocessing After a maximum two years 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 TO dry cask unloading facility and the C, D and E storage pools The four stainless steel lined storage pools are interconnected and offer a total nominal capacity of more than 14 400 Mg U

Spent fuel transportation casks are unloaded in one of the two cask unloading facilities Fuel possibly damaged during transport is identified at the unloading units during the cask unloading process and defective fuel is stored in tight bottles Spent fuel stored at La Hague has been irradiated in both French and overseas customers' NPPs (Belgium, Germany, Japan, The Netherlands, Switzerland) The fuel stored is all zirconium alloy clad, with excellent behaviour under wet storage conditions

Spent fuel 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 Storage density of 3 Mg U/m^2, including all the basket transfer areas for pool operation, is achieved

As of 1995, storage records:

- More than 9300 Mg U was stored in the La Hague AFR storage complex;
- Spent fuel with high burnup (up to 50 GW·d/t U) is currently unloaded and stored;
- MOX fuels are stored (approximately 28 Mg U);
- Average personnel doses in the cask unloading facilities is very low (respectively 0.5 mSv/year in the wet unloading facility and 0.3 mSv/year in the dry cask unloading facility);
- Average personnel doses for the operations linked to spent fuel storage (spent fuel handling, water treatment, etc.) are equal to zero, thanks to remote handling operations (for basket transfer) and to specially designed equipment such as immersed water treatment units.

Sweden

In Sweden, the back end policy is based on final disposal of the spent fuel generated in the nuclear power plants. At the reactor, spent fuel storage is limited to a short period, 9 months for BWR and 12 months for PWR, prior to transportation to centralized 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.

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

The current storage capacity is 5000 Mg U. As of 1 January 1996, the storage records are as follows:

- 2200 Mg U, with a maximum burnup of 48 GW·d/t U is stored;
- The average collective dose is about 100 mmanSv/year;
- After 10 and a half years of successful operation the emission of radioactive nuclides to the environment is negligible.

United Kingdom

The wet storage of fuel discharged from MAGNOX and AGR Power Stations can be conveniently divided into two phases. The first covers short term storage (AR), and the second the further interim storage away-from-reactor (AFR) at Sellafield, prior to reprocessing.

With the exception of Wylfa which has dry stores, all other MAGNOX and AGR stations currently operated by Nuclear Electric plc and Scottish Nuclear Limited use cooling pools fabricated from reinforced concrete. Each pool is provided with water circulation, cooling and filtration. For MAGNOX pools, high circulation rates throughout the volume of the pool are desirable to avoid local temperature variations and ensure thorough mixing of freshly treated water returning from the chemical plant.

The storage of AGR fuel in station pools continues to be good. No problems have been experienced with interim storage of MAGNOX fuel in station pools. Irradiated MAGNOX and AGR fuel is stored at the power stations in open topped skips (rectangular transportable containers containing fuel assembly storage tubes) and transported to Sellafield in shielded flasks.

There are four main AFR storage pools at Sellafield. These pools store fuel from the first and second generation of UK reactors, MAGNOX and AGR, and also irradiated fuel from LWR reactors, mostly from BNFL's overseas customers. The pools are unlined reinforced concrete structures. Some pool walls are painted concrete.

Sellafield currently accepts irradiated fuel from 11 MAGNOX power stations in the UK and elsewhere, and MAGNOX fuel will continue to arrive at Sellafield at least until the year 2006. AGR

and MAGNOX fuel is received in the fuel handling plant (FHP), the fuel is placed into containers The design of the container lids allows for triple stacking After a minimum of 180 days cooling the AGR elements are dismantled, the fuel being consolidated into cans MAGNOX fuel is stored up to 18 months, being decanned prior to reprocessing

Fuel integrity/performance under extended storage periods has been monitored

Because of the recognized risk of stainless steel fuel pin cladding corrosion through sensitization of non-stabilized SS, the condition of AGR fuel elements has been subjected to an extensive monitoring programme to determine the effects of increasing storage time (longest stored fuel approximately 19 years) and increasing fuel burnup (initial fuel approximately 5 GW·d/t stringer mean irradiation (SMI), now at 22 GW d/t SMI, with a planned increase up to 30 GW d/t SMI) Fuel elements were removed from the pool and subjected to detailed destructive examination From this work not only has a thorough understanding of the pool storage behaviour of AGR fuel been obtained, but a corrosion inhibitor to prevent AGR fuel pin corrosion has been identified All the main AGR and MAGNOX fuel storage pools have now used corrosion inhibitors for the last nine years

LWR fuel is transported from the reactor pools to Sellafield in multi-element bottles (MEBs) contained within heavily shielded, high integrity, transports flasks MEBs are stainless steel tubes with a welded base, removable lid, internal grid for criticality control/holding fuel and integrated pipes for flushing/venting/ullaging Criticality control is maintained by either "Boral" (boron carbide aluminium matrix), or boronated stainless steel plates

MEBs were introduced to overcome problems associated with crud migration within wet transport casks and fuel transfer operations for interim storage, but have afforded several benefits These include substantial reduction in operator dose uptake, faster cask turnarounds, no fuel handling operation (thus reducing the probability of dropping fuel elements), reduction in pool water activity levels, minimised volume of contaminated water to be handled when the fuel is reprocessed and better control of water chemistry around the fuel assemblies

3 1 1 3 Defective fuel inspection/detection methods

Inspection of spent fuel for cladding defects is generally dictated more by in-reactor fuel performance than by requirements for storage primarily to identify the number and cause for fuel failures [10] However, this information may be useful to provide assistance to develop a spent fuel pool-storage strategy or to identify candidate assemblies for encapsulation in those pools that require defected fuel containerization

In-pool sipping is often used to detect cladding defects in spent fuel bundles Isotopes used as indicators include ^{131}I and ^{85}Kr Because of radioactive decay, sipping techniques become less effective as a function of time They need either specially adapted test procedures or eventually may become impractical

In recent years, utilities have begun to use an ultrasonic testing technology that detects defective fuel by sensing the presence of internal water that has entered the fuel rod through the defect This technology can locate the specific rod where a failure has occurred The technology is effective in screening fuel for defects in cases where procedures prohibit placing known defective fuel into dry storage

Key issues still remain in developing practical strategies for adequately defining defective fuel and then implementing appropriate actions if necessary From a storage standpoint, particularly wet storage, it is not clear that the definition of defective fuel used as the criterion to require special handling or encapsulation procedures should be the same as the defective fuel definition used in reactor performance Currently, different countries have taken different approaches for handling defective fuel which reflect different levels of conservatism One factor to be considered is the relative risk of defective fuel compared to the normal risk of bare intact fuel that is heavily contaminated by external

surface crud. Often the crud (from BWR fuels) causes more severe handling and contamination problems than defects in fuel cladding.

When defects in the fuel become large or when there has been obvious mechanical damage to a fuel assembly, then special treatment is clearly in order. Encapsulation is appropriate where the cladding defect size is such as to permit loss of particulates from the bundle. For AFR storage the presence of defective fuel is recognized through water chemistry analysis results (namely fission product presence). If required, defective fuel identification may be done by isolating the fuel and following the water activity increase. Also, if mechanical damage occurs that makes normal fuel pickup and handling impractical, or where the changed geometry of the bundle makes storage in normal rack or basket spaces difficult, then special treatment for the bundle is called for.

In the **United Kingdom**, mobile equipment for endoscopy has been developed and is now used at the five AGRs operated by Nuclear Electric (see Ref. [11]) and by SNL. This technique allows internal inspection of fuel elements. Out-of-pool monitoring on fuel transported to hot cells at NE's Berkeley Technology Centre has continued and includes the destructive examination of elements from stringers discharged at lead fuel irradiations.

3.1.1.4. Behaviour of defective fuel

Fuel with known defects has also been stored in water-filled pools with little difficulty. Even so, some countries elect a conservative approach of storing known defective fuel in tight secondary enclosures or "bottles" to ensure minimum release of activity into the pool water. One country, the UK, has elected to containerise all LWR fuel destined for the THORP reprocessing plant in MEBs as a way to minimize the problems created by external surface crud and defective fuel in the transportation casks and in the storage pools. This regime offers the advantage that bulk pool water can be discharged to the sea, after monitoring and conditioning, without the need for expensive in-pool ion exchange systems.

Several R&D programmes have been conducted to establish the behaviour of defective fuel rods in wet storage.

Canada has conducted the most comprehensive wet storage fuel integrity surveillance program. Beginning in 1977, both intact and deliberately defective CANDU test fuel was identified and characterized. A subsequent reexamination was made in 1988. This fuel, after some 27 years in storage, showed no measurable oxidation of the cladding surface and no changes were observed in the condition of the fuel. Examination of the UO_2 in the immediate area of the intentional defect showed a very thin surface layer of oxidized UO_2 in the form of UO_3 hydrate, but there was no evidence of a diametral increase in the element due to UO_2 oxidation in defective fuel that was stored wet for up to 21 years. This evidence suggests that pool storage of defected bundles can continue for at least 50 years with no significant degradation. This surveillance programme in Canada is scheduled to continue for the next 50 years with the next reexamination to occur in 2000. At the time of the reexamination the undefected fuel will have been stored for up to 30 years while the intentionally defective fuel will have been stored for up to 33 years.

In **Germany,** a surveillance programme with intact and operational defective fuel rods has been conducted for 18 years. No changes were observed in the nature of the defects over this period of time. The defects included cracks, hydride failures, boils and wear spots [12, 13].

The Republic of Korea has also conducted a surveillance programme on fuel with known defects and has seen no changes after 5 years of storage.

Russia reported no changes in RMBK spent fuel water activity over a 25-year period with standard cleanup system operation. This indicates that there is no defect propagation mechanism working in the storage phase. For WWER, fuel pool activity variations are associated with handling operations and the number of defective fuel assemblies stored.

In **Sweden**, at CLAB, two shipments of leaking fuel have been received The first consisted of five leaking BWR-assemblies from Ringhals 1, the second shipment had three leaking PWR assemblies from Ringhals 3 The assemblies were transported in special bottles During unloading at CLAB, the assemblies were removed from their bottles and sipped No significant ^{134}Cs and ^{137}Cs leakage was observed The assemblies were therefore treated as normal fuel and were placed in ordinary fuel storage canisters Experience indicates that the ^{134}Cs and ^{137}Cs concentrations increase slightly in the storage pools (containing 10–11 000 assemblies) After approximately three weeks, the concentration in the pool returns to normal as a result of the normal continuous cleaning

3 1 1 5 Special design related issue

As spent fuel storage pools begin to approach their design capacities, certain factors require surveillance In some cases (US reactors) pool-cooling limitations are being approached and bulk pool temperatures are increasing substantially This problem is exacerbated when it is necessary to conduct a full core discharge and an entire core of short-cooled fuel is off-loaded to the pool In those reactors, pool temperatures as high as 65 °C have been observed Such temperatures, while in the short term do not endanger fuel integrity, do produce a more aggressive corrosion environment and also produce very restrictive personnel working conditions in the fuel storage area

3.1.2. Pool water

When storing spent fuel in pools, the role of water is to keep the cladding temperature low, to maintain the integrity of fuel cladding, and to allow spent fuel visual inspection while ensuring biological protection for the operators The experience reported so far indicates that

- Spent fuel storage performance for all types of fuel is satisfactorily,
- Operational fuel defects are not propagating

To continue the above performance, water quality has to be maintained at a high purity level to prevent significant corrosion In addition, proliferation of microbiological species (such as algae growth) has to be controlled to prevent the loss of water clarity

It is clear that water quality will be specific to the type of fuel stored in the pool and the conditioning mode selected for that fuel (bare fuel or containerized) Depending on the type of fuel stored in the installation (CANDU, AGR, MAGNOX, PWR, BWR, RBMK or WWER fuel type), operating conditions for the pool water will be different Variables that may be affected include water chemistry, water treatment, water quality control and water temperature

3 1 2 1 Water chemistry

The role of water chemistry is key for underwater storage For each storage installation, optimized chemical composition has to be specified in order to prevent the corrosion of cladding or fuel structure material In the case of defective fuel storage, water chemistry is providing adequate storage performance if selected properly in accordance with the specific requirements

- For **CANDU fuels**, which use Zircaloy for cladding material, water chemistry specification is based on a long experience gained during more than 30 years in Canada and 12 years in the Republic of Korea Spent fuel is stored in at-reactor pools in demineralized water with typical specifications 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)

TABLE II TYPICAL EXAMPLES OF WATER CHEMISTRY PARAMETERS IN STORAGE POOLS

Type of fuel	CANDU	AGR		MAGNOX		RBMK	
Pool type	AR pool	AR pool	AFR pool[a]	AR pool	AFR pool	AR pool	AFR pool
Fuel container (if any)			Flooded containers		Ullaged containers		Water filled cans
Cooling	Demineralized water	Borated water	Demineralized water			Demineralized water	Demineralized water
pH	6 9	7	13	11 5	11.4[b]	5.5 - 8	6-7 5
Conductivity	200 µS/m		-	< 200 µS/m	-	300 µS/m	-
Chloride	0 5 mg/kg	0 5 ppm	< 0 5 - 1 2 ppm	< 0 1 ppm	0 4 ppm	0.1 mg/kg	0 05 - 0.1 ppm
Sulphate	0 02 ppm	0 5 ppm	0 9 ppm	< 0.4 ppm	0.6 ppm	-	-
Fluoride	0 06 ppm	-	-	< 0 5 ppm	-	0 1 mg/kg	
Na^+ and Ca^{++}	0 1 ppm	-	-	-	-	-	-
^{137}Cs	0 003 MBq/m^3	10 MBq/m^3	29 4 MBq/m^3 [c]	0 - 60 MBq/m^3	20 MBq/m^3	-	
Water activity	9 MBq/m^3	40 MBq/m^3	29 5 MBq/m^3	0 - 60 MBq/m^3	20 MBq/m^3	5 MBq/m^3	

[a] Purged pools, no in-pool ion exchange systems. Ion exchange takes place downstream of the storage pool in a separate facility, prior to sea discharge.

[b] pH inside ullaged containers, not in pool

[c] Pool contains small quantities of fuel with perforated cladding

TABLE II TYPICAL EXAMPLES OF WATER CHEMISTRY PARAMETERS IN STORAGE POOLS

Type of fuel	WWER		PWR and LWR				
Pool type	AR pool	AFR pool	AR pool	AFR pool			
				General case	CLAB pool	La Hague pool	THORP pool [d] MEB container [e]
Cooling	Borated water	Demineralized water	Boratedwater (PWR) Demin water (BWR)	Demineralized water	Demineralized water	Demineralized water	Demineralized water [f]
pH	4 3 - 6 5	5 5 - 7	4 5 - 5 5	4 - 6	5 - 6	5 - 5 4	6 2 - 7 5
Conductivity	1000 µS/m	200 µS/m		100 - 350 µS/m	< 100 µS/m	130 µS/m	
Chloride	< 0 1 ppm	< 0 1 ppm	0 15 ppm	< 0 1 ppm	< 0 005 ppm	< 0 005 ppm	0 4 ppm
Sulphate	< 0 1 ppm	< 0 1 ppm	0 15 ppm	< 0 1 ppm	< 0 005 ppm	0 01 ppm	0 6 ppm
Fluoride	< 0 1 ppm	< 0 1 ppm	0 15 ppm (Korea)	< 0 1 ppm	< 0 005 ppm	0 05 ppm	
Na^+ and Ca^{++}	< 0 5 ppm	< 0 1 ppm		< 0 5 ppm	< 0 001 ppm	< 0 03 ppm	
^{137}Cs	10 MBq/m^3	1 MBq/m^3			0 02 MBq/m^3	-	4 1 MBq/m^3
Water activity	40 MBq/m^3	40 MBq/m^3	20 MBq/m^3	5 - 20 MBq/m^3	10 MBq/m^3	12 MBq/m^3	4 2 MBq/m^3

[d] Purged pools, no in-pool ion exchange systems Ion exchange takes place downstream of the storage pool in a separate facility, prior to sea discharge.

[e] Pool also contains AGR fuel in ullaged containers

[f] MEBs can contain either deionized or borated water

- For **AGR fuels** that are clad with stainless steel, pool water chemistry has been controlled in order to prevent fuel corrosion. The risk of AGR fuel pin cladding corrosion through sensitization has been recognized and investigations have been carried out to identify a corrosion inhibitor (sodium hydroxide). As a result, AGR fuel storage pools at Sellafield have now used corrosion inhibitors for at least 9 years.

  All AGR stations use cooling pools where spent fuel is stored in borated water. The water chemistry at AGR stations is dictated by criticality control. Boron is added in form of boric acid. Sodium hydroxide is also added to return the pH of pool water to 7. The 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 <0.5 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, are established when the fuel is placed in containers for storage in the pool (pH <7, Cl^- <0.2 ppm). Containers are sealed. For experimental purposes, individual containers can be 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.

- **MAGNOX fuels** use magnesium alloy for cladding. During pool storage, the magnesium hydroxide corrosion products dissolve and raise the pH. MAGNOX fuel storage performance is closely related to the pH value of the pool water and to the chloride and sulphate impurities. Fuels have been satisfactorily stored for up to about 5 years with a pH value of 11.5 by dosing with sodium hydroxide and keeping $[Cl^- + SO_4^{2-}] < 1$ mg/kg.

  Following a recent review of MAGNOX fuel at reactor storage, chloride levels have been reduced to 0.1 ppm and alkalinity is now specified in relation to OH^- concentration rather than pH. Raising the pH value to 13 with low anion levels has provided better storage performance. Based on extensive research programmes, a water chemistry of pH 13 and $[Cl^- + SO_4^{2-}] < 0.5$ ppm has been chosen for the MAGNOX storage environment. When received to the AFR storage pool in Sellafield, spent fuel is placed in sealed containers. The required water chemistry for MAGNOX fuels at the AFR storage pool is established when the fuel is placed in the container (pH 13, $Cl^- < 0.5$ ppm).

- For AR **RBMK** storage, elements are suspended from the pool floor beams. The fuel has zirconium alloy cladding (Zr-Nb) and the required water chemistry quality has been defined in order 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 that it changes with storage time. The simultaneous effects of the absence of water mixing, filtration and exchange, lead to an increase in water corrosivity, and the amount of radiolysis products and halogens. As a result, corrosion becomes more intensive and diversified. Steel (X18H10T) shows a tendency to pitting corrosion. For suppressing water corrosivity alkali additives can be used.

- **Light water reactor fuels** (PWR or BWR) have zirconium alloy 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 the UK). The water purity is controlled to suppress conditions that might lead to a corrosion environment for spent fuel clad and structures. Depending on the countries that operate LWRs, 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). The pH requirement for BWR stations is 5.3–7.5 in Japan. Chloride and fluoride concentrations are limited to 0.5 ppm, while each operating value is generally maintained below 0.05 ppm.

  The at-reactor pools for PWR reactors are filled with borated water, while demineralized water is used for BWR reactors as well as for AFR wet storage installations serving both LWR reactor types.

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.

- **WWER fuels** use Zr-1%Nb alloy cladding, they are stored in borated water in the at-reactor pools. Concentration of boric acid is maintained within 12–16 g/kg; concentration of halides (Cl^- and F^-) is below 0.15 mg/kg; pH varies in the 4.3–6.5 range. AFR pools, used at a later storage stage, are filled with demineralized water. Concentration of halides is kept below 0.15 mg/kg; pH values are within the 5.5–7.0 range.

3.1.2.2. Water treatment

Water treatment is necessary because activity is transferred from the fuel cladding to the storage water (through corrosion products settled on the cladding or fission products transferred to the pool water through clad defects). Water activity has to be maintained at a reasonably low level to limit operator dose rates as well as to reduce the activity accumulation in tubes and pipes used to transfer pool water outside the basin. In addition, special attention is required for water quality to avoid microbiological species proliferation that could reduce water clarity

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

Mechanical treatment is generally performed by filters (pre-coated, sand, or mechanical filters), while chemical treatment is realized with ionic exchangers (cationic and anionic resin types are used). When saturated, ionic exchange resins may be regenerated. Mixed bed can also be used, in which case, there is no regeneration phase for the resin bed and it is disposed of after resin saturation. Some pools are equipped with an immersed water treatment unit (Nymphea ion exchanger in France). This helps to avoid installing radioactive parts of the water treatment network outside the pool, thus providing an improvement in operating conditions (lower personal exposure).

Depending on the type and condition of the fuel stored in the pools, water activity content may vary. If intact fuel is stored, corrosion products (such as Co, Ag, Mg) will represent the main part of the water activity. In the case of defective fuel storage, fission products (Cs, I, Sr) will constitute the major part of the activity to be removed. In the case of AR storage, some activity may be introduced in the water pool during refuelling operations.

In any case, the volume of water to be treated each day depends on the water activity level to be maintained on one hand and on the activity transferred to water on the other hand. For spent fuel stored in sealed containers (such as the Multi Element Bottle used in THORP), water treatment volume is limited. However, the relatively small volume of active liquor contained inside the MEB, which is flushed out before opening, is sent for treatment at the site's Enhanced Actinide Removal Plant.

For AR and AFR storage pools, water activity is generally maintained below 100 MBq/m^3. The water activity level maintained in the pool is a key issue to the dose received by the operators. Experience shows that the water activity level is responsible for the half of the doses received by the operators in storage facilities. The lower the activity, the less is the radiation dose of the operators.

In some cases, mostly in summer, microbiological or algae growth has occurred. Significant biological fouling (biofouling) was experienced in a spent fuel storage pool heat exchanger at one of the nuclear power stations in Canada. The fouling was attributed to a wide variety of bacteria which included pathogenic coliforms. Sulphate reducing bacteria (corrosion bacteria) were either absent or present in low numbers. The action taken to control the bacteria consisted of thorough cleaning of the heat exchanger with a brush, collection of the biological material and incineration of the protective clothing worn by the workers. Appropriate biocides (hydrogen peroxide) at concentrations up to 1000 ppm were added (to the pool water) to control biofouling.

Biofouling and algae's growth was also detected in another spent fuel storage pool in Canada, which resulted in frequent plugging of the pool purification filters. The bacteria observed on the filters were actinomycetes and polysaccharides. Unicellular and non-motile algae cells, both gram-positive (Bacillus type) and gram-negative (Pseudonomas type) bacteria, were also observed on the paper filters. The walls and the floor of the storage pool were brushed and hydrogen peroxide at the concentration of 150 ppm was added to the pool water to kill the biofilm.

In the **Slovak Republic**, biological species growth in pool water was recognised since 1990. The phenomena resulted in biomass flocculation at the pool surface and a partial decrease in water clarity. Biomass growth has taken place mostly in low flow conditions (inspection and emergency pool with limited water circulation). In the biomass, psychrophylic and masophylic bacteria and micromycetes were identified as the main part of the biological mass. Only negligible amounts of sulphur bacteria have been found. Application of a standard ion-exchange resin-based cleanup system has not been successful in all cases of biomass removal needs. As biomass growth has decreased, only mechanical removal from the water surface has been carried out. Laboratory tests were conducted to test coal and micro-porous membrane filtering and hydrogen peroxide application up to 1000 mg/kg to manage this problem. Microfiltration has been identified as the most effective method, it will be implemented in the future as high flow submersible filter unit.

The use of chlorinating agents to combat algae is not recommended with sensitised AGR cladding.

3.1.2.3. Water quality control

The objective of water quality control is to maintain the conditions that minimize the corrosion of spent fuel and pool components and the concentration of radioisotopes in the storage water and guarantee the clarity of storage pool water. Analyses of chemical elements are performed on a routine basis.

The following parameters are monitored:

- pH
- Conductivity
- Water turbidity
- Chemical composition (i.e. Cl^-, F , SO_4^{2})
- Isotopic specific activity.

Generally, the complete water activity content is checked weekly, while gross beta/gamma measurements may be performed daily in some installations. The complete chemical analysis of the water is generally performed on a monthly basis. In some installations, chloride, fluoride and sulphate levels are measured weekly (La Hague in France, Sellafield in UK).

In **Sweden**, at CLAB, it was found that the activity release to the storage pool water is more than 95% ionic. In the Safety Analysis Report the opposite was assumed, predicting 90% to be in particulate form. The activity released is 90-95% ^{60}Co, the remainder being mainly ^{54}Mn; less than 1% is Caesium. The activity concentration at 2200 metric tonnes of UO_2 is low (7 MBq/m^3) but higher than expected. The reason for this is the above mentioned high proportion of ionic release. Particles would have settled on the pool bottom and would not have been observed in the water samples taken from the pools. Instead the activity now remains in the water volume.

The following list summarizes the common parameters used to control water quality:

Parameter	**Frequency**
pH	Weekly
Cl^-, F^-, SO_4^{2-}	Weekly
NO^3, PO_3^{4-}	Weekly
Beta- gamma control	Weekly
Complete chemical analysis	Weekly

3 1 2 4 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, minimizing bacterial or microbial growth, and lowering the humidity level in the storage area

For AFR storage installations, where fuel cooling times are 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 demonstrated (La Hague in France) the lower the storage water temperature, the less activity is transferred from the fuels to the water At 30°C the activity release leads to 15 MBq/m^3 for water activity, while 6 MBq/m^3 at 17°C As a result, the current policy adopted at the La Hague AFR storage complex is to maintain the lowest temperature in the storage pools

In **France** pool water is cooled with the patented Nymphea in-pool heat exchanger units Normal temperature is below 40°C and, should the heat exchanger pumps stop, heat exchange is maintained by thermosyphon effect

In **Sweden** measurements of the influence of water temperature on the activity release were performed in 1988 when the cooling was reduced so the temperatures rose from 28°C to 36°C The activity concentration reached a new equilibrium level, 2 1 times higher

The lesson to be learnt from this is that it is an advantage to keep pool water temperature as low as reasonably possible, thereby delaying the release and taking advantage of the decay of the activity while it still is fixed to the fuel surface The load on the waste treatment systems becomes correspondingly lower (see Ref [14])

In the **UK**, for MAGNOX pools, high circulation rates throughout the pool are desirable to avoid local temperature variations The temperature in the fuel handling pools is specifically limited to below 15°C to minimise the leaching of activity from Magnox fuel

3.1.3. Pool components and materials

Stainless steel is the predominant pool component material used for spent fuel storage structures, heat exchangers, piping and fuel handling equipment Corrosion studies conducted in Japan, The Republic of Korea and the UK indicated that the corrosion of stainless steel is very small (10^{-5} mm/year) [15]

3 1 3 1 Pool lining

Pool storage installations are massive structures made of reinforced concrete Two main methods may be considered for lining the spent fuel storage pools stainless steel pool lining or coating the concrete with an epoxy resin

Most LWR AR storage pools are stainless steel lined, while both stainless steel and epoxy liners are used for CANDU AR storage All AFR storage installations are stainless steel lined (France, Finland, Japan, Russia, Slovak Republic, Sweden), with the exception of UK AFR storage pools which use epoxy coated concrete

In case of concrete pools coated with epoxy, 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 GGy radiation dose. With the extension of the operating life, this dose limit could be exceeded at some point after the original design life of the pool is reached. Since 1988, some radiation-induced deterioration has been observed at the Pickering station in Canada, 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. 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.

For stainless steel lining, no corrosion phenomena have ever been observed, even after biomass growth, as mentioned in the report from the Slovak Republic.

3.1.3.2. Pool leakage detection

The design of the pool storage facility has to ensure that pool water leakage is kept to an absolute minimum in terms of both magnitude and frequency of occurrence. The design also generally ensures that any small leakages can be detected.

Since the beginning of the use of pool storage, only a few minor pool liner leaks have been reported. These leaks have all been at weld joints. Small leaks in Hungary, the Slovak Republic, and the USA were reported. However, in preparation for a possible leakage caused by a design basis accident, an underwater welding system has been developed in Japan in order to repair the defective stainless steel pool lining without evacuating the pool. This equipment was successfully tested in a full scale mock up test.

In **Canada**, techniques have been developed for underwater repair of defects in both stainless steel and epoxy liners. Using an underwater-curing epoxy, a leak that developed at the stainless steel-epoxy interface has been successfully repaired. This technique will be used to rehabilitate the Pickering epoxy liner where localized radiation-induced surface deterioration has occurred.

In **France** the stainless steel inner pool lining of the storage pools, formed by the welding of stainless steel plates, is 100% X-rayed during construction. Permanent monitoring of the welded seams and X-ray inspection capability is provided to locate and detect any leak during operation.

In **Hungary** the lining of the spent fuel pools and of the adjacent pit No. 1 (used for transport container loading) leaked. Originally there were rectangular welds, which developed cracks as a result of cyclic forces. A part of the pool linings was replaced in the past. It is assumed that the leaks come from the old, non-reconstructed, part of the lining. In 1992, the leakage at Unit 1 was about 20–600 l/h, and at Unit 2: 7–8 l/h. The lining at Units 3 & 4 had no leaks.

In **Sweden** the leakage from the pools containing about 14 000 cubic metres of water is between 5 and 10 litres per day. This value has been unchanged during10 years of operation of the plant.

3.1.3.3. Neutron absorber material

There are two types of neutron absorber material used in the storage pools:

- Liquid,
- Solid.

In PWR type reactors (including WWERs) there is a liquid absorber (boron) in the primary circuit, and as a result, also in the AR pool. The BWR, CANDU and RBMK reactors use only pure water; their AR pools contain no liquid absorber.

In many storage installations solid neutron absorber material has to be used to ensure the subcriticality of the storage array. Various materials have been developed such as:

- Boraflex,
- Boral,
- Boronated aluminum alloy,
- Boronated stainless steel alloy,
- Cadminox

Boraflex, which contains 25 to 40% B_4C is a silicone rubber binder that was introduced in the late 1970s The material is effective in these sections (typically 1-2 8 mm) and is delivered precut to the required size The Boraflex is then clad in SS in various plate and square tube configurations and is vented Boraflex is initially quite flexible but it hardens to nearly ceramic-like condition under irradiation and undergoes shrinkage (see Refs [16-18])

In the USA there are some 40 to 50 plants that use Boraflex as a neutron poison in their high density storage racks This material undergoes degradation as a function of cumulative gamma dose and exposure to pool water With the degradation, the B_4C material contained in the polymer may be released, thus potentially affecting the criticality safety of the pool While blackness testing methods are available that indicate either the presence or absence of the neutron absorber, no current methods exist to quantitatively measure the in-rack concentration of B_4C EPRI has undertaken an R&D programme to develop such a technology

Boral consists of boron carbide particles in an aluminum matrix cladding with pure aluminum and is widely used as a neutron absorber It comes in sheets containing three layers the central one contains a cermet of approximately 50% B_4C in 1100 type aluminum, and the outer two layers are the cladding (usually aluminum) When fresh aluminum surfaces are placed in water, hydrogen gas is given off for about the first week, and then gradually stops (the oxide film that builds up on the aluminum surface thereafter prevents further reaction and the surface is passivated) Aluminum has the potential for galvanic corrosion in the presence of SS, and this potential is greater in PWR pools with dissolved boric acid at a pH ~5, as compared to BWR pools with a neutral pH of 7 Both these problems must be addressed in the design of a Boral neutron absorber rack Blistering of Boral has created problems in Boral equipped racks in the USA in the 1980s, and once in France in 1988

Boral inserts allow the closer packing of assemblies within MEB In most MEBs, the Boral is exposed, which results in the removal of oxygen produced by radiolysis of the water in the MEB and produces an ullage gas composed mainly of hydrogen There is, however, a small number of MEBs where the Boral is clad in stainless steel In these cases the oxygen removal is less efficient and higher ratios of oxygen to hydrogen can result in the ullage space Consequently, the latter MEBs are flooded on arrival at the AFR and allowed to vent to the bulk pool water No evidence of pool-induced degradation of LWR fuel has been found at Sellafield

The internal baskets of some transport and storage casks are equipped with *boronated aluminium alloy* This material, used to guarantee the criticality safety, is also more efficient against thermal loads

Boronated stainless steel is a type of 304 SS with addition of up to 1 9% natural boron to act as a neutron absorber It has very low corrosion rates and high stability under irradiation However, the addition of boron makes the material considerably harder and more brittle The addition of boron and the need to ensure its uniform dispersion within the SS makes the material normally expensive The maximum 1 9% loading of boron, dictated by material workability considerations, may be overcome by using enriched boron-10, although the cost of the material in this case undergoes a significant increase because of the cost of enriched B-10 From the viewpoint of long term material integrity, boronated SS is the preferred absorber material although from the material and fabrication cost perspectives it may be the most expensive

Cadminox is the most recently developed neutron absorber material and uses cadmium in a leak-proof envelope instead of boron The rolled cadmium sheet is mate of natural cadmium with metallic and organic impurities Its use is limited to some French NPPs and a very small number of other plants

In **France** boronated steel is used in the fuel storage baskets for spent fuel (a boronated steel sleeve is introduced in the compartments of the PWR basket).

In **Germany** compact storage racks have been equipped with neutron absorbers made of boronated stainless steel since the early 1970s [19] after an extensive R&D programme. Base material is DIN 1.4550 or 1.4301 with a boron content in the range between 0.8 and 2.5% by weight. Contrary to plastic- or aluminium-bound boron carbide, which is used also as neutron absorber, boronated stainless steel never had any operational problems or needs any in-service inspection.

Nearly 50 000 absorber channels have been manufactured and have been installed in nuclear power plants in Europe and Asia by Siemens.

In **Hungary,** at the Paks NPP, the four at reactor pools were reracked. In the compact racks each fuel bundle is placed in a hexagonal stainless steel absorber tube, 3050 mm long and of 3 mm wall thickness, containing 1.05–1.25% natural boron.

In **Japan** at the BWR plants, boronated aluminium alloy racks have been used since 1978, and boronated stainless steel racks have been used since 1993. In the near future, there is a plan to install boronated stainless steel racks in further BWR and PWR plants.

3.1.3.4. Decommissioning of storage pools

In the **UK** the planned first phase of decommissioning includes:

- Removal of concrete surfaces and coating and extracting core samples;
- Cleaning, break down and packaging of the remaining fuel element skips;
- Cleaning the pool superstructure and concrete walkways.

This phase will take approximately 18 months and an assessment will then determine the course of action to be adopted.

In the **USA**, decommissioning of storage pools is becoming more significant because of the early shutdown of several reactors. Because of the high cost of maintaining the pools operational, there are major incentives to move spent fuel to AFR dry interim storage on the reactor site (to passive storage) and then shut down the pools. This strategy is complicated by the fact that, in many cases, the only way to get the fuel off-site would be to pass the fuel back through the pools to be loaded in a transport cask.

Decommissioning of a former stainless steel pool has been carried out in **France** at La Hague. The objective of the operation was to remove fixed contamination in order to reuse the pool for other purpose (conditioned waste underwater storage).

3.1.3.5. Crud[1]

In **France**, the experience gained with more than 14 000 Mg U of LWR fuels handled in the reception and storage installations have highlighted two key parameters responsible for crud being released from the fuel: the thermal shock on the fuel cladding and fuel handling operations increasing the volume of crud deposited on the bottom of the storage structure (basket at La Hague). Regular washing of the storage structures where crud is deposited limits the residual contamination of the material. At La Hague, the pool bottom is cleaned with an underwater, remotely operated vacuum cleaner where the coarse particles are trapped in a cyclone and the fines in a filter. Both cyclone and filter is remotely changeable.

[1] In the context of the BEFAST reports "crud" means a deposit on fuel surfaces.

In **Japan**, the reception and storage pools of the Tokai Reprocessing Plant have received about 950 Mg U of LWR and ATR fuel by the end of 1995 Removal of crud deposited on the bottom of these pools has been performed experimentally using a special cleaner

In Hungary, following the evaluation of the results of physical samples of corrosion isotopes taken from various points around the primary cooling circuit of Paks NPP, Unit 1 (reactor vessel cover, steam generator hot and cold leg collectors, main circulating pumps), the characteristic isotopes and composition of the crud were as follows [20, 21]

Isotope	Proportion (%)
^{51}Cr, ^{95}Nb, ^{124}Sb	< 1
^{54}Mn	10–15
^{59}Fe	1–5
^{58}Co	30–50
^{60}Co	20–30
^{110m}Ag	10–25

In the safety assessment process of the Paks Modular Vault Dry Storage Facility, to estimate the quantity and composition of fuel crud on fuel assemblies transferred from the at-reactor spent fuel storage pool to the MVDS, it was assumed that there would be a similar composition of these corrosion isotopes on the external surface of discharged fuel assemblies Due to the low proportion and short half-lives for some of the above-mentioned isotopes and assuming a 3-year minimum assembly decay period before the transfer to the MVDS, in the analyses the dominant isotopes are ^{54}Mn, ^{60}Co and ^{110m}Ag for the fuel assemblies Characterisation of fuel crud as a source of radioactivity had been based on information from USA PWR and Hungarian WWER experience with fuel crud This information establishes ^{54}Mn and ^{60}Co as the only significant isotopes within the crud 3 years after fuel discharge Accordingly it was conservatively assumed based on proportions presented above, that for freshly discharged fuel the specific activity of ^{110m}Ag was the same as the specific activity of ^{60}Co From a consideration of the relative half-lives of these two isotopes and the 3-years decay period, it was concluded that the activity of ^{110m}Ag would be an order of magnitude lower than that of ^{60}Co

Russian measurements have shown that the dominating portion of the crud inventory is due to the principal corrosion products of the primary circuit, among which iron oxides prevail

In the **USA**, crud has proven to be a significant problem in BWR plants, affecting both pool cleanliness and transportation Evaluation and development of various type of fuel bundle cleaning techniques have led to the conclusion that these cleaning methods are only partially successful and that when all factors are considered they are not cost-effective

3.1.4. Off-normal events

3 1 4 1 Spent fuel corrosion

LWR fuel structural material consists mainly of Zircaloy and stainless steel Zircaloy has proven to be insensitive to any kind of corrosion phenomena (uniform corrosion, stress corrosion cracking (SCC), electro corrosion) in the temperature range $\leq 60°C$

In such cases where Zircaloy and SS are in direct contact, in-service passivation prevents any electro-corrosive phenomena SS is also resistant to any kind of corrosion However, some attention has to be given to SCC in the neighbourhood of welds Water chemistry quality control has proven to be a reliable remedy against SCC of SS components

In the **Slovak Republic** the rest of the HWGCR fuel (A-1 Unit), which was not transported to Russia, is stored in canisters At a certain number of these fuel assemblies severe corrosion of the Be-Mg alloy cladding and fuel assembly structure has occurred during long-term storage in potassium

chromate solution. This phenomenon resulted in difficulties with fuel handling. Various alternatives to solve this problem are now under consideration.

In the **UK** some AGR fuels stored at Sellafield have experienced corrosion. As the pool was unroofed, the chloride content in the water slowly increased reaching a maximum concentration of 6 ppm. This caused corrosion of a portion of the AGR fuels stored in the pool. These elements were manually dismantled. The adoption of corrosion inhibitors has prevented any further occurrences of fuel element corrosion.

3.1.4.2. Spent fuel structure corrosion

During underwater storage of LWR fuel, sleeve corrosion has been reported for some LWR fuel with Zircaloy guide tubes and stainless steel upper sleeves. The corrosion phenomena, probably induced at the reactor, have occurred for high carbon content in the upper sleeve material and can jeopardize future spent fuel handling operations. Such incidents have been reported in France (for some French, German and Japanese fuel stored at La Hague), the Republic of Korea, Spain and the USA.

In order to minimize corrosion of the spent fuel structure in the AFR installations, technical specifications for storage water have been defined.

In **France**, a conservative approach was followed; and spent fuel assemblies identified as doubtful regarding sleeve and upper/end fitting material have been reinforced. Chloride, fluoride and sulphate concentrations are limited to 0.1 ppm, sodium and calcium contents have to be below 0.5 ppm, while the minimum value for the pH is 4.5. Keeping these water chemistry specifications will reasonably ensure that sleeve-corrosion induced at the reactor will not progress during AFR storage.

In the **USA**, one significant event that occurred during handling operations at a spent fuel storage pool involved separation of the top end fitting (nozzle) from the remainder of a PWR fuel assembly as the assembly was being lifted out of a storage rack in the pool at the Prairie Island NPP. There were no radioactive releases from this Westinghouse assembly, and no fuel rod damage occurred. The fuel assembly was subsequently lifted and inserted into a storage position. Intergranular stress corrosion cracking of the Type 304 SS sleeves, which were welded to the top nozzle and mechanically joined to the Zircaloy control rod guide thimbles, was identified by Westinghouse Electric Corporation as the cause of the failure. The Zircaloy guide thimbles appear to have remained intact. The unanticipated SCC is believed to have occurred while the assembly was in the spent fuel pool. Based on available records, the pool chemistry met specifications reasonably well during the storage period. Westinghouse has substantial experience with this type of SS-to-Zircaloy joint, and conducted examinations of fuel assemblies at other plants too. Except for Prairie Island, the results revealed no evidence of stress corrosion cracking of the SS sleeves. Westinghouse considers this type of failure to be an isolated occurrence. A total of 13 (including the one where the nozzle broke off) out of 27 assemblies examined showed evidence of corrosion, but there were no additional nozzle failures when the assemblies were removed. An evaluation conducted by the USNRC concluded that the event is an isolated incident and does not have generic impact [6].

3.2. DRY STORAGE

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-1000, RBMK-1000, AGR, MAGNOX and the HTGR). Since its conception, dry storage of spent fuel has evolved into a wide variety of dry-storage systems. Examples of this are concrete canisters, steel lined concrete dry storage containers and concrete CANSTOR Modules (modular vault-like storage system) in Canada, concrete canisters in the Republic of Korea; vaults in France, UK and the USA; and casks in Germany, Japan, Russia and the USA (see Ref. [22]). At the present time, eight countries participating in BEFAST-III (Canada, France, Germany, Japan, Republic of Korea, Russia, United Kingdom and the USA) are engaged in the dry storage of spent fuel. All of the above countries are 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

Dry storage has become a mature technology and the quantities being placed into dry storage are beginning to increase significantly The inventory of spent fuel in dry storage in the above countries as of 1 January 1996 was approximately 3425 Mg U, the largest quantity (about 1143 Mg U) being stored in Canada In Canada, starting in 1996, about 25 000 spent CANDU fuel bundles containing about 590 Mg U will be placed into dry storage annually Also, in the USA, as many of the spent fuel storage pools reach their capacity, dry storage is becoming a significant factor in the utilities spent fuel storage strategy In Japan, a dry cask storage facility for the Fukushima-I power plant started operation in September 1995 The facility has a capacity of 20 casks In the first phase, by October 1995, 9 casks were loaded Four casks are loaded with 37 BWR assemblies each, while 5 have a capacity of 52 BWR fuel assemblies [23] Russia is planning to construct dry storage facilities for the RBMK type nuclear power plants at Kursk and Smolensk

A helium storage environment is used in most of the systems, although air as well as helium is used in Canada and only air in The Republic of Korea Air is also being used for the experimental dry cask storage of RBMK-1000 fuel in Russia and in the vault for the dry storage of MAGNOX fuel in the UK Air/CO_2 is also used for the storage of prototype AGR fuel in the UK

Dry storage of spent LWR fuel in an inert atmosphere is licensed for temperatures up to 410°C in Germany and 380°C in the USA [24–26] Dry storage in air is licensed in Canada and the Republic of Korea for temperatures up to 160°C

In most cases, the dry storage containers are loaded with spent fuel under water After removal of the bulk water, the container requires drying to prevent aqueous corrosion and hydriding of the Zircaloy components The drying processes utilized to-date are vacuum and hot air 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 dried satisfactorily if the system pressure (e g ~10 millibar) remains constant for a specific time period (e g ~10 minutes) However, large amounts of residual water may require a fairly long drying time The evaporation rate will also depend upon the temperature of the surrounding area The procedure that is used for vacuum drying requires careful consideration 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 Typical vacuum level conditions are 5 to 10 millibars, which must remain stable without the vacuum system operating and with the specified allowable increase in pressure over a specified time period Vacuum drying is utilized in Canada, Germany and the USA

In the hot gas drying process (process temperature ~90°C), the fuel assembly is uniformly heated by the hot gas to evaporate the water 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 and The Republic of Korea In these countries, the fuel assembly containment 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 zirconium alloy (Zr-2, Zr-4, Zr-2 5%Nb and Zr-1%Nb), however, dry storage experience exists also with Mg, Al and 20/25NbSS clad fuels Average burnups of spent fuel in dry storage range from 4 5 to 33 5 GW d/t U, while the maximum burnups range from 7 5 to 50 GW·d/t U

In Japan the R&D for dry cask storage has been carried out mainly by Central Research Institute of Electric Power Industry (CRIEPI) [23, 27] R&D subjects include temperature analysis and tests of fuel cladding, creep tests of fuel cladding, containment tests of cask lid structure, cask drop tests,

building collapse and heavy objects dropping onto cask, and cask toppling by earthquake. CRIEPI has been carrying out an advanced research programme on spent fuel storage technology. Major activities of the new programme are evaluation and related tests on dry storage methods suitable for spent high burnup fuel and MOX fuel in terms of safety and economy, which include application of burnup credit and temperature analysis. The activities also include various design 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.

Additional details of the dry storage experience, burnup extension, failure mechanism assessment, issues related to extended storage, retrievability, transportation and system compatibility, monitoring and off-normal events are discussed in this section.

3.2.1. Dry storage experience

Dry storage experience in the BEFAST-III countries is described in this Section. Since the inception of BEFAST, other countries have developed dry storage systems which are not reported here.

Canada — Canada has been storing spent fuel in a dry environment in tile holes at the Atomic Energy of Canada Limited (AECL) Chalk River Laboratories (CRL) since the early 1960s and in AECL designed reinforced concrete canisters (CCs) since the mid-1970s. AECL has recently developed two modular dry storage systems, CANSTOR (for CANDU fuel) (See Ref. [28]) and MACSTOR (for LWR fuel) [29] for use in Canada and elsewhere. These are reinforced concrete modular structures that store spent fuel inside metal containers placed in concrete modules. CANSTOR has been 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 latter part of 1995. Ontario Hydro has also designed its own reinforced concrete Dry Storage Container (DSC) for spent fuel storage. Ontario Hydro loaded spent CANDU fuel from its Pickering-A NGS into its first DSC in December 1995.

The CCs are licensed for the storage of spent fuel to a maximum temperature of 160 °C in air, and 250°C in helium (CCs at AECL's Whiteshell Laboratories - WL). AECL CCs are being used to store fuel from Hydro Quebec's Gentilly-1 reactor, AECL's Whiteshell Research Reactor WR-1, AECL's Douglas Point NGS and the AECL Nuclear Power Demonstration (NPD) reactor, all of which have been taken out of service. Concrete canisters are also being used at New Brunswick Power Corporation's Point Lepreau NGS. As of 1 January 1996 spent CANDU fuel containing 864 Mg U had been stored in CCs in Canada.

Ontario Hydro's DSCs are transportable steel lined (inner and outer) concrete containers which are currently licensed only for storage of fuel at the Pickering NGS. Phase 1 of Ontario Hydro's spent fuel dry storage facility has a capacity for 700 DSCs containing about 5 300 Mg U and will provide storage of Pickering fuel for about 10 years of operation.

AECL and Ontario Hydro initiated a long-term experimental dry-storage investigation programme in concrete canisters at AECL's Whiteshell Laboratories in 1978 [30]. Representative fuel bundles from Ontario Hydro's Pickering-A and Bruce-A NGS were chosen for the studies and characterized to provide baseline data on their pre-storage condition. The oldest bundle had been discharged in 1971. Periodic interim storage examinations are conducted, and the results of the examinations are compared with the baseline data to evaluate the possible changes in fuel integrity. To date, no release of radioactive material from the experimental storage containers to the interior of the canisters has been detected by the canister air-monitoring systems. Undefected fuel elements examined during the interim storage examinations have exhibited no apparent change from their pre-storage condition. Although a significant amount of fuel in the intentionally defected elements have experienced UO_2 oxidation in dry air at temperatures up to 150°C, no apparent increases in fuel element diameters have been observed, and the structural integrity of the fuel cladding has been maintained. Fuel element end-cap fractures extending into the cladding were observed in a number of intentionally defected (3-mm hole in the cladding) spent CANDU fuel elements that were stored for up to almost 8 years in moisture-saturated limited air at 150°C plus almost 2 years of storage in moisture-saturated unlimited air (i.e. replenished)

at 150°C The fractures were attributed to delayed hydride cracking as a result of severe hydriding of the end caps and positive cladding strain adjacent the end caps, which occurred during in-reactor operation, in combination with damage experienced by the fuel bundle prior to dry storage

Because oxidation of UO_2 to U_3O_8 in defective fuel elements could result in splitting of the fuel cladding and the release of fuel particles to the interior of a dry storage container, complementary studies on the oxidation of UO_2 (unirradiated and irradiated) in both dry and moist air are also being conducted [31] These experiments are designed to characterize the effect of moisture on UO_2 oxidation To achieve accelerated oxidation processes, experiments have included unirradiated fuel at 200–225°C Long-term oxidation experiments were initiated recently at lower temperatures (125–175°C) on spent CANDU fuel as well as on spent LWR fuel Oxidation of UO_2 to U_3O_7 in defective fuel elements has been modelled for limited and unlimited air conditions and has been correlated with the dry-air experiments at 150°C (see Refs [32, 33])

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 1996, approximately 100 Mg U 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 180 Mg U 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

Forged metal casks have been developed by Transnucleaire for spent fuel storage (TN24, TN32 and TN40) These casks are currently used for spent fuel storage in several countries

Germany — The analysis of dry storage performance of Zircaloy-clad LWR fuel indicates the need to limit only the dry-storage insertion temperature and not storage time [34, 35] Thus, the spent fuel storage performance does not preclude any decisions for practical storage options All of the dry storage results are based on practical experience with spent fuel storage, research and development, and demonstration programmes In Germany, only casks are licensed for dry storage [36, 37] Germany has licensed two storage sites, one at Ahaus and another at Gorleben The storage capacity at Gorleben has been recently increased from 1500 t HM to 3800 t HM using the new CASTOR V/19 cask After completing the licensing of Ahaus for the CASTOR V/19 cask, the total storage capacity in Germany will increase to 6300 t HM

Republic of Korea — Concrete canisters (AECL design) are being used for the dry storage of the CANDU spent fuel bundles in Wolsung Storage started in 1992 and by the end of 1995, 21 600 CANDU spent fuel bundles (about 34% of the total CANDU spent fuel discharged) have been stored dry in concrete canisters at the Wolsung site

The CANDU spent fuel bundles are stored in a dry air environment after at least 6 years of pool storage 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

United Kingdom — The general experience of Nuclear Electric with dry storage of MAGNOX fuel at Wylfa Power Station, both in the short term CO_2 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, the largest of the MAGNOX stations, less vulnerable to transport or reprocessing difficulties Accordingly, first one (DSC4) and then a second dry store (DSC5) were built The overall design of the cells was dictated by the need to access the existing fuel discharge route, 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 lesser.

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.

United States of America — With pool storage rapidly reaching the capacity limit in many USA plants, dry storage has begun to become a significant factor in the utilities spent fuel storage strategy. On 1 January 1996, nine sites had dry storage operational and another five or six plants had made dry storage commitments. Some 1000 Mg U was in dry storage. All dry storage in the USA is in an inert helium environment. To date, no information on fuel degradation in commercial dry storage is available. An R&D programme has been started at the Idaho National Engineering Laboratory to monitor and verify dry storage performance. Periodic analysis of cask cover gas samples has verified the continued integrity of spent fuel that has been in dry storage for more than 10 years [38].

3.2.2. Burnup extension

The behaviour of high burnup bundles in storage, although a very small fraction of the total inventory of stored spent fuel, requires special consideration because of high fission gas and fission product inventories, higher cladding strains, higher hydride concentrations and more cladding oxidation. Only nondefective bundles are placed in dry storage. Special considerations such as canning of the bundles prior to dry storage would be a nominal requirement for defective bundles.

Canada — The burnup range of CANDU fuel is typically 3.8 to 11.7 GW·d/t U with an average value of 8 GW·d/t U. However, approximately 210 Bruce-type, 37-element bundles have achieved burnups in excess of 16.7 GW·d/t U [39]. The above number of bundles constitutes only 0.05% of the Bruce-type bundles irradiated. A 16-month fuelling restriction permitted three of the bundles to achieve bundle-average burnups of 19.9 to 21.6 GW·d/t U. An examination of these high burnup bundles revealed that they have a greater potential to suffer in-reactor damage involving pellet-clad interaction (PCI) failures. Four of the 210 bundles that were experienced high burnup suffered PCI related failures. In the bundles that experienced burnups above 20.8 GW·d/t U, numerous outer-element circumferential end-cap cracks were also observed. In addition, high burnup fuels also experience higher fission product inventories, higher fission-gas releases and higher cladding strains than normal burnup fuel.

France — In the CASCAD installation, fuels are stored with a burnup that varies between 15 and 30 GW·d/t U. Burnup extension, currently up to 60 GW·d/t U, is taken into account in the design of the TN storage casks. While gamma and neutron shielding are achieved by two independent materials (stainless steel and resin), extended burnup requires a thicker neutron protection. Moreover, spent fuel cooling time may be increased before cask storage, if necessary. Burnup extension is considered for the storage of 33 PWR assemblies with 50 GW·d/t U burnup and 8 years cooling time in the TN cask family.

Germany — The fuel cycle costs for LWRs and the increasing demand on the nuclear fuel industry suggest the need for higher discharge burnups. Utilisation of new cladding material will allow batch average burnups in the range of 50 to 60 GW·d/t U to be achieved. Dry storage, which is challenging from an economic aspect, faces a more complex situation than wet storage. Assuming that current developments will result in achieving burnups corresponding to batch averages in the range of 50 to 60 GW·d/t U, burnup limitations may be necessary in dry storage. Fuel cycle strategies contemplating direct disposal or even the "wait and see" approach are confronted with storage periods of up to 100 years. Those strategies clearly favour dry storage. Solutions for decoupling fuel assembly operation and long term storage will contribute to the safety and economy of the back end of the fuel

cycle [40] Presently, CASTOR V/19 is licensed for fuel assemblies with a batch average burnup up to 55 GW·d/t U

Japan — The maximum burnup of 40 GW·d/t U and 4 years of cooling time are the reference values for licensed dry cask storage of BWR fuel at the Fukushima NPP The future burnup targets for PWRs are about 55 GW d/t U with an initial enrichment of about 4 5%, and about 55 GW·d/t U for BWRs

Russia — Work recently performed in Russia to improve fuel and fuel cycles, promoted the reliable operation of WWER-1000 in a 3-year mode and WWER-440 in a 4-year mode at the average fuel burnup 42–45 GW·d/t U The work programme currently underway is aimed at new fuel to create WWER cores having the following basic characteristics

- Average burnup 55–60 GW·d/t U,
- 5–6 year fuel cycle,
- Use of integral burnable absorbers (UO_2 + Gd_2O_3),
- Optimized enrichment and UO_2 fuel charge in a fuel rod (through increased weight per fuel rod unit length),
- Use of high irradiation and corrosion resistant Zr-Sn-Nb-Fe alloy as cladding and guide thimble's material

United States of America — Burnup targets for LWR fuel continue to increase in the USA For PWRs, current reloads are designed for ~45 GW d/t U with an initial enrichment of about 4 0% ^{235}U BWRs are slightly lower with designs at about 40 GW·d/t U and enrichments of about 3 5% Plans for future reloads now indicate that LWR burnups may reach as high as 60 GW d/t U with initial enrichments near 5 0%

3.2.3. Failure mechanism assessment

The R&D programmes performed in the different BEFAST Member States cover generic investigations, defect mechanism assessments, defective fuel rod (FR) behaviour investigations, safety-related investigations and verification experiments Results presented in the different BEFAST meetings can be summarized as follows

The fission gas release from spent UO_2 is low to negligible The tritium is distributed between fuel (about 60%) and cladding (about 40%) Its release can be calculated by Sievert's law Oxide and crud layers are adherent and did not remove from FR surface during dry storage tests

Creep strain of a Zircaloy cladding in dry storage caused by the FR internal overpressure is regarded as a potential failure mechanism [41] Fuel rods with a burnup of <40 GW·d/t U exhibit a moderate internal fission gas pressure in storage Also, irradiation damage to the cladding is limited Burst tests performed with such cladding resulted in minimum uniform creep strain of 1% or higher before defection Post-pile creep rates investigated in Germany (Zry), Japan (Zry) and Russia (Zr-Nb) resulted that post-pile creep that can be described conservatively by applying creep equations from unirradiated material Calculating the end of storage cladding creep strain based upon EOL FR conditions and typical FR temperature histories from dry storage facilities results in a total strain $<1\%$ Based on the above, maximum dry storage temperatures around 400°C were licensed in different countries, e g in Germany, Russia and the USA

However, the approach described above would result in a major decrease of maximum dry storage temperature with increasing burnup since the typical uniform burst stain measured is around 0 5% and the FR internal fission gas pressures rise with increasing burnup Also, the cladding wall thickness decreases as a result of corrosion due to longer service times in higher burnup fuel Therefore the former burst test used to determine the allowable end of storage strain was replaced by creep rupture testing of irradiated cladding in Germany The main reason for the change is that creep rupture tests result in more representative strain data than burst tests Creep rupture tests are underway in Germany

with cladding from FR having a burnup up to 70 GW·d/t U. Until those results are available, the German licensing authorities assume a total allowable strain of 2% at the end of storage. However, those total strains must also include the in-service strain and account for the in-service positive EOL strain. When the results of the creep rupture tests become available, less conservative total strain restrictions are expected. The creep deformation in licensing is limited to below 1% in Japan; whereas in Russia 2% total hoop strain is allowed for Zr-1%Nb-alloys. The related maximum storage temperature is limited to less than 400°C–450°C [42–44].

To avoid any stress corrosion cracking (SCC), the hoop stress in spent LWR fuel rod cladding due to rod internal pressure should be limited to less than or equal to 100 N/mm^2 at the end of life. The above is supported by investigations performed by a number of BEFAST Member States.

Operational tests have shown that a CASTOR cask filled with FRs having operational defects could be dried by standard cask drying procedures. Water logging was not observed in highly instrumented single rod tests conducted in Germany. Fission product releases from fuel rods occur only in the heat-up phase, and only in small amounts and the release rates approached zero after a few days of storage. Demonstration tests with a CASTOR cask containing a variety of operationally defected FR supported the single rod tests.

Extended dry storage test programmes revealed that FR defects originating from in-service operation do not increase in size during dry storage. Research was carried out in Canada in air, while inert gas storage was investigated in Germany, the Republic of Korea and the USA. However, in air or oxidizing cover gases, UO_2 exposed at the location of the defect oxidizes to UO_{2+x} or U_3O_8. Only the U_3O_8 formation at elevated temperatures causes a volume increase which could stress the cladding causing defect propagation. Recent Canadian research has shown that the necessary temperature must exceed 200°C to produce recordable effects.

Verification tests in different BEFAST member states (e.g. Canada, Germany, Russia, USA) show that the integral dry storage performance of Zry or Zr-alloy clad spent LWR or HWR fuel assemblies is excellent [44].

The work on the generic phenomena of defective spent fuel oxidation is still ongoing in **Canada**.

In the **Republic of Korea**, CANDU fuel that is known to be defective is not stored in canisters. To avoid significant oxidation of spent fuel in air environment storage conditions during ~50 years of storage, the operating temperature is limited to 160°C.

The **USA** has 2100 stainless steel clad LWR assemblies in storage. EPRI has initiated an evaluation of the expected performance of this fuel and of the question: should this fuel be placed in long term dry storage or packaged for disposal? It is now not clear that stainless steel cladding will behave the same way as Zry cladding during dry storage. The most probable potential failure mechanism is stress corrosion cracking of the sensitized steel. It is anticipated that this is the limiting factor in storage performance of stainless steel clad LWR fuel [41].

3.2.4. Issues related to extended storage (>50 years)

The current licensing practice for most of the BEFAST members who have implemented dry storage covers licensing periods up to 50 years.

Three modes of storage can be used to describe extended storage of spent LWR fuel, based on available experience:

- Mode I: fast decrease of cladding temperature above 350°C;
- Mode II: medium decrease of cladding temperature in the interval between 150 and 350°C;
- Mode III: slow decrease of cladding temperature below 150°C.

Adequate performance data are available for all storage modes Mode I is typical for early interim storage with short pre-storage times in pools Mode III covers extremely long storage periods, whereas Mode II is assumed to be characteristic for extended storage if spent fuel is packed densely in large canisters or in geologic structures Canisterized fuel with direct cooling of the individual canisters, as discussed in USA, France and Germany may provide storage in Mode III Generally low FA temperatures in storage are favourable for extended storage

Today's knowledge does not suggest any severe problems that endanger spent FA performance in the extended storage periods in modes II and III According to current knowledge, some work may be required to ensure the extrapolation of the data generated in investigation periods up to 100 years Current fuel rod failure mechanisms that influence dry storage performance can be described by Arrhenius extrapolations Consequently data generated in mode I can easily be extrapolated to extended storage in modes II and III Storage performance is currently provided by as-licensed storage up to 50 years [13]

In addition to the questions related to spent fuel performance, activities may be required for long term, quality assured, records and availability of the FA data to characterize the spent fuel in storage This data is required for safety and safeguards reasons Hard copies of the documents promote the record but the readability may suffer changes and the accessibility is poor Storage of the data on electronic media allows fast access and easy data handling However, in the anticipated time period, those media will change in hard- and software areas Procedures to convert the data in a quality assured manner have to be developed

In **Canadian** studies, the investigators believe that intact spent CANDU fuel can be stored under dry conditions maintaining their integrity for at least 100 years Canada has been storing spent fuel in a dry environment since the early 1960s As of 1 January 1996 water-cooled research reactor fuel containing ~96 t HM had been stored in tile holes, organic-cooled research reactor fuel containing ~24 t HM had been stored in concrete canisters, spent CANDU fuel containing 904 Mg U had been stored dry in concrete canisters, ~7 Mg U in a DSC, and ~112 Mg U in a CANSTOR module Concrete canisters were selected to store spent fuel from the 680 MWe Point Lepreau NGS About 30 years of spent fuel production at Point Lepreau represents 147 000 CANDU fuel bundles or about 2 790 Mg U in 273 concrete canisters Thirteen CANSTOR modules will be used to store spent fuel containing 2 790 Mg U from Gentilly-2 for the production life of the station, while 1 500 DSCs will be used to store spent fuel containing ~12 000 Mg U for the production life of Pickering-A and Pickering-B

3.2.5. Retrievability, transportation and system compatibility

Following long-term storage, spent fuel may have to be retrieved and transported either for reprocessing or to a disposal facility It is therefore important that the fuel storage systems being developed are compatible with downstream requirements The experience gained so far in these areas is discussed below

Canada — CANDU fuel may be stored under dry conditions after at least 6 years of pool storage During pool storage, the fuel is stored in storage containers of several designs (called baskets, trays and modules) and, thereafter, the fuel is transferred to dry storage In the case of CC storage, the fuel bundles are removed from the pool-storage containers and transferred under water to specially designed cylindrical CC storage containers, called baskets The loaded baskets are transferred to a shielded work station where the spent fuel is dried with forced hot air and then the baskets are seal-welded The loaded baskets are transported to the CC for loading in a shielded flask As of 1 January 1996, spent CANDU fuel containing ~1 040 Mg U had been loaded successfully into 1 200 fuel storage baskets, transported individually to the CCs and CANSTOR in a shielded flask and unloaded from the flask into the CCs

At Ontario Hydro, the fuel modules are loaded directly under water into the DSCs The DSCs are removed from the water, drained, seal-welded, vacuum dried, backfilled with helium, leak tested and are then transported by tractor trailer to the storage site Two Ontario Hydro demonstration

cylindrical Concrete Integrated Containers (CICs — an earlier version of the DSC) have been successfully loaded under water, each with spent CANDU fuel containing ~8 MTU. Both containers were transported to an outside storage area. After 4 years of outside storage, the CIC containing fuel that had cooled for 10 years under water was retrieved, transported to the spent fuel storage pool and successfully unloaded. The second CIC containing 6-year water-cooled-fuel was unloaded successfully after 5 years of outside storage.

In the spent fuel dry storage experiments (Controlled Environment Experiments - Phases 1 and 2), fuel bundles are retrieved routinely from the CCs for interim storage examinations, loaded into a shielded flask and unloaded in the hot cells for the examination. The procedure is reversed for returning the fuel to the canisters. Although no large scale spent fuel shipments are required at this time in Canada because of the on-site storage practice, dry shipments are carried out frequently between the commercial reactors and AECL's Chalk River and Whiteshell Laboratories for fuel examinations. To date all loading, retrieval and transportation methods used in Canada's dry storage programme have performed without incident. Based on this record, post-storage retrievability and transportation of spent fuel are expected to be problem-free.

Germany — When choosing the long term storage and final disposal option, the following steps are required to accommodate the back end of the fuel cycle in Germany:

- Load the spent fuel into the storage and transportation casks;
- Transport to Gorleben or Ahaus sites;
- Store the fuel for an appropriate time;
- Move the spent fuel at the end of intermediate storage to the fuel conditioning site for direct disposal;
- Unload the transport and storage casks;
- Perform the spent fuel conditioning process to prepare the fuel for final disposal;
- Load the processed spent fuel into the final storage casks;
- Dispose of the final storage casks into the final repository.

The spent fuel will have to be retrieved for the back end of the fuel cycle in Germany as described above. According to German Safety Regulations, the size and weight of the CASTOR-type cask exceed the mine lifting capacities. Consequently, the Pollux containers have been developed. The conditioning process for the Pollux containers involves the following steps:

- Unloading the Castor casks;
- Consolidating the fuel into the Pollux containers;
- Consolidating the residual structural FA components for loading into the Pollux containers;
- Closing the fully loaded Pollux container.

The FA conditioning process is developed and the Pilot Konditions Anlage (PKA) is under construction at the Gorleben site. Civil engineering work is currently in progress.

Russia — Russia has constructed a dual purpose (storage/transportation) dry cask (TK13). The cask is 6035 mm long and 2430 mm in diameter; its wall is 340 mm thick and contains 12 WWER-1000 type spent fuel assemblies. Spent fuel is shipped from the nuclear power stations by railway in a horizontal position. The cask is designed primarily for transportation (> 100 shipments) and is now licensed for 1 year storage of fuel in an inert gas atmosphere (He, Ar) at a fuel temperature of 350°C.

United States of America — Increasing demand for dry storage and competition among cask vendors has resulted in the development of large capacity, dual purpose systems. These transport-storage systems include multi-element sealed canisters that could be permanently disposed of with an appropriately designed overpack.

3.2.6. Monitoring

Monitoring of fuel storage systems is performed in support of licensing to verify the proper function of the system Although the fuel itself is not examined during storage (except in some experimental programmes), the containment barrier is monitored to verify container integrity Examples of the type of monitoring that are conducted is provided below

Canada — In addition to the interim spent fuel storage examinations in support of the experimental dry storage program, the CCs have been monitored since 1975 for potential radiological releases Monitoring is a regular feature at the dry storage facilities and is carried out as a precautionary measure for public and worker protection

The CC monitoring consists of visual inspections, canister interior air monitoring to detect the potential release of radioactive material from the fuel storage containers, radiation field measurements (on contact and at 2 m), radiation field measurements at the canister site perimeter fence, and analysis of the surface and sub-surface ground water for radioisotopes Because of the accumulated monitoring experience, it has been demonstrated that the frequency of monitoring initiated at Whiteshell Laboratories was not necessary at future dry storage sites Consequently, the frequency with which these measurements are performed varies between the different dry storage facilities in Canada For example, at New Brunswick Power's Point Lepreau's dry storage site, a sliding scale depending upon the age of the fuel is used to determine the frequency of CC air monitoring Each site is also responsible for the quality assurance and quality control pertaining to storage monitoring

Germany has also monitored fuel behaviour and radiation levels while demonstrating the feasibility of dry storage [45]

Republic of Korea — The dry storage CCs have been monitored to verify proper functioning of the system These have included a physical inspection of the canister area, a leak test of the canisters, radiation field measurements on the canister surface and the fence boundary, and analyses of surface and underground water for radioisotopes The frequency of these measurements is as follows

- Physical inspection - yearly,
- Leak tests - quarterly,
- Radiation field measurements - continuously,
- Surface water analysis - weekly,
- Underground water analysis - quarterly

United States of America — A long term storage and monitoring programme has been conducted at the Department of Energy Idaho National Engineering Laboratory since 1986 Four dry-storage casks with spent nuclear fuel are used for the programme (GNS V/21, TN-24P, Westinghouse MC-10 and Sierra Nuclear VSC-17) Cask pressure, local radiation and other items are monitored, cover-gas is sampled and analysed The monitoring programme results are published annually

3.2.7. Off-normal events

United Kingdom — The general experience of dry storage of MAGNOX fuel elements at Wylfa is good However, in the summer of 1990 a small number of elements showing severe corrosion were discovered in one of the air-filled dry stores The incident is summarised below and some conclusions drawn which bear on design of long term storage facilities

During routine discharge of a fuel element from a skip in DSC4 in July 1990, it was discovered by TV inspection that only the top lifting button was removed by the grab Closer inspection of the skip tube showed a very severely corroded element with exposed uranium (The procedure is to transfer elements from a DSC4 skip to a transfer machine and then via the CO_2 filled store (DSC1) discharge route to a transport flask) The remaining elements in the skip were in excellent condition apart from one which showed signs of surface corrosion Detailed inspection of elements already placed in the transfer machine from this skip showed another element with some severe corrosion

Introduction of a mobile purpose-built camera into DSC4 allowed inspection of all elements in all skips (~21 000 elements). The number of elements affected by varying degrees of corrosion was around 45, distributed among some 11 skips. In a few instances water could be seen in the element storage tube. It was discovered that affected fuel had all resided at some time in the same small area of the store (remember there is skip movement as part of loading/unloading procedures). The only source of water was from outside the store. It was eventually shown that a water ingress had occurred through a combination of circumstances which, despite careful design precautions, had allowed water to penetrate below the steel roof and collect on the inner concrete flat roof of DSC4 and then penetrate the inner roof/wall expansion joints at the end of the store which abutted the reactor building. Once the route for water ingress was ascertained, further entry was prevented and it was confirmed that no such problem existed with the DSC5 roof.

The dry storage water leakage at Wylfa Power Station was reported by Bindon in 1992 [46] and updated at the Toronto BEFAST-III meeting [47]. In addition to the repairs of DSC4, inspection holes have been introduced on top of the dry skin, drains have been installed and additional activity monitoring and moisture monitoring have been introduced.

Following the water ingress, all 47 355 fuel elements in DSCs 4 and 5 were subjected to full remote TV inspection. This confirmed that water ingress/corrosion had been limited to DSC4 where some 45 elements had some degree of corrosion, although only 11 were judged to be of a serious nature. All skips containing damaged fuel have now been segregated and each of the damaged elements and their immediate neighbours have been examined in detail using an endoscope. A number of damaged elements have been lifted into a purpose-built, controlled atmosphere chute for loading into a special canister and transferred to DSC 1, 2 or 3 prior to despatch to Sellafield in dry casks.

There is confidence that the condition of the damaged fuel is now well understood and that it is very stable. Notwithstanding this, every precaution was taken during recovery operation to ensure that the potential for hydride blistering was minimized.

Germany performed investigations on the mechanical impacts on spent fuel. The primary objective of these investigations was associated with aerosol formation [48]. The investigations indicated that aerosol formation from mechanically compacted irradiated fuel is rather small because particle formation is mainly influenced by the smallest grain size of the fuel. Transcrystalline fracture is very unlikely.

3.3. LICENSING AND SYSTEM STRATEGY ISSUES

The technologies applied to spent fuel storage must always be considered in the context of licensing requirements and the overall spent fuel disposition strategy of each specific nation. While all countries impose regulations to ensure public health and safety, the detailed requirements often have significant country-to-country differences. Additionally, the different types of fuel (CANDU, PWR, BWR, AGR, MAGNOX, etc.) require quite different storage designs to accommodate the differences in cladding materials, heat generation rates and shielding requirements.

In the area of system strategies, the choice of storage technologies is heavily influenced by the chosen fuel cycle strategy of each specific country. For example, the storage requirements for a country involved in a closed fuel cycle using reprocessing is vastly different than in an open, once-through fuel cycle where storage may be planned for 50 years prior to geologic disposal [13]. Also, different storage strategies may be chosen where there is a single nationally-dictated storage strategy as opposed to a situation where each utility is making its own plan for how to deal with its spent fuel prior to the time of disposal or reprocessing. Decisions that may clearly be best for an overall national strategy may often not be viewed by an individual reactor plant or utility as being in their best interest.

Following is a list of factors that impact system strategies.

3.3.1. Transportation considerations

At some time, fuels that are in storage will have to be transported, either to a final repository, a reprocessing plant or to another storage location It is therefore important to consider the storage/transportation interface requirements when designing a storage and fuel cycle strategy

Factors that impact on the transportation interface include

- The size of the storage module (single or multi-element),
- Whether the fuel is bare or encapsulated,
- The condition and integrity of the fuel,
- The amount and nature of external surface crud on the fuel,
- The time the fuel is cooled prior to transportation (determines heat and shielding requirements),
- Burnup,
- Type of fuel

3.3.2. Fuel disposition options

The choice of a storage strategy is highly dependent on the country's choice of fuel disposition strategy In France, Japan and the UK, reprocessing is the chosen path and, therefore, at-reactor fuel storage requirements are reduced and storage needs can be efficiently met by a centralized pool storage facility at the reprocessing plants Other countries which have chosen reprocessing via contracts with reprocessing nations, such as Japan, Germany and some eastern European countries, expect to ship most of their fuel off-site, but must also deal with the uncertainties of contract limits that often do not cover all the country's spent fuel While most of the fuel from Japan until now has been reprocessed in the UK and France, small scale reprocessing has been carried out at PNC's Tokai Reprocessing Plant The Rokkasho Reprocessing Plant, the first commercial plant of Japan, is scheduled to go into operation after 2003 Its centralized storage facility with a capacity of 3000 Mg U is under construction and goes into operation in 1997

Several countries, including Canada, Finland, Germany, Spain, Sweden and the USA have announced a plan for permanent geological disposal of spent fuel However, the timetable for achieving this goal is uncertain Therefore, utilities or national agencies must plan on indefinite on-site or off-site storage Other countries, such as the Republic of Korea, have not committed to a fuel disposition strategy and therefore must plan for storage in a way that supports either a reprocessing or a disposal strategy

3.3.3. Storage strategies

The choice of an overall fuel storage strategy involves decisions in choosing the fundamental type of preferred storage (wet or dry) along with deciding if the storage should be dispersed (at the various reactors sites) or centralized (at only one or a few locations) These choices are normally heavily influenced by the countries' chosen fuel disposition option Economics also is a major consideration in determining each country s fuel storage strategy Transportation aspects and the geography of the country must also be factored into the decision

3.3.4. Other considerations

There are other specific factors that can impact the choice of a fuel storage strategy As an example, a country's requirement for Safeguards measures may well influence the storage technology selected Inspection and verification requirements may eliminate certain storage designs or impose technology requirements that cannot be met In any case, a country should clearly define its storage Safeguards requirements in order to permit proper integration into the overall storage system design

Another consideration in choosing a storage strategy is to assess the match between the storage mode chosen and the planned operating life of the reactor For instance, continued in-reactor pool storage may be very expensive after the time the plant is shut down and decommissioning activities have

begun. Additionally, on-site dry storage may be faced with a problem if the only way to achieve eventual off-site transport is for the fuel to be moved back through the reactor pool.

Another consideration is the conflict that often arises between choosing the least cost option to meet immediate needs compared to a more expensive option that may offer greater future benefit or be more optimum for operation of the overall fuel disposition system. The cheapest storage option often does not provide the best flexibility or interface conditions for future fuel disposition options. The classic question of how much should be spent today to provide future benefits can be very difficult. The question is further complicated by the problem that the timing of events in the future is highly variable and evolving regulations and changing political climates makes the future very uncertain.

4. BEFAST-III DATA ON STORAGE EXPERIENCE

4.1. WET (POOL) STORAGE

Data on wet storage experience of the BEFAST-III member states are presented in Table III. The following parameters were used in reporting the data:

- Reactor type;
- Kind of fuel cycle preferred in the country;
- Fuel rod cladding type;
- Burnup of the spent fuel;
- Quantity of spent fuel. (The difference between the cumulative arisings and the present inventories indicates that fuel has been transferred from wet to dry storage or from storage to reprocessing.);
- Information on going R&D activities;
- Specific remarks.

Most of the data are as of 1 January 1996. Some of the contributions are based on earlier results (e.g. January 1995). These data were corrected using general fuel cycle information to achieve a common baseline.

4.2. DRY STORAGE

Data on dry storage experience reported by the BEFAST-III member states are presented in Table IV. Additional information on spent fuel storage type and the storage environment of the spent fuel assemblies is also included.

Practically no spent fuel has been removed from dry storage so far. In Germany and in the USA, some spent fuel stored for demonstration purposes was returned to the spent fuel pool, while unloading the casks. During these operations, the rewetting behaviour of spent fuel formerly stored in a dry mode was documented in Germany. The removal of spent fuel from dry storage is normally associated with the termination of experimental programmes. So far, no fuel has been transferred from dry storage to final disposal.

5. SUMMARY AND CONCLUSIONS

The contributions of the participants to the first two phases of BEFAST indicated that all known failure mechanisms, such as oxidation, stress corrosion cracking, pitting, crevice corrosion, hydriding, galvanic effects, fission product attack, water logging, etc. could be excluded during the wet storage of undamaged zirconium alloy clad fuel under proper storage conditions.

To maintain MAGNOX and AGR spent fuel cladding integrity, the chemistry of the pool water has to be closely controlled. Under such storage conditions no problems have been experienced.

TABLE III WET STORAGE EXPERIENCE IN THE BEFAST-III COUNTRIES

Country	Reactor type	Cladding type	Burnup GW·d/t U			Longest residence time (yr)	Inventory (Mg U)		Difference (Cumulative - present)	Active R&D	Comments
			Average	Typical	Maximum		Cumulative	Current 1/1/96			
Canada	CANDU	Zircaloy	7 4-8.5		25 2[a]	32 6	22 330	21200	1130	yes[b]	Difference in dry storage
Finland	PWR	Zircaloy	-	36	42	14	635	635	0	no	
	WWER	Zr-1%Nb	34	35	43 7	8	410	177	233	no	[c]
France	LWR	Zircaloy	43	44	52	15	24500	16700	8400	no	[d]
	GCR	MgZn0 5%				10	11300	500	10800	no	[d]
	FBR	S S				10	70	50	20	no	[d]
Germany	PWR	Zry-4	35-40	40	45	5	3600	1200	2400	yes	[d]
	WWER	Zr-1%Nb	abt 30	abt 30	-	abt 10	925	925	0	no	
	BWR	Zry-2/4	35-40	40	45	5	2600	700	1900	yes	[d]
Hungary	WWER	Zr-1%Nb	33	35	36	5	560	384	176	no	[c]

[a] Maximum burnup achieved in Bruce reactors, not typical

[b] Fuel under surveillance since 1978

[c] Difference went to reprocessing in Russia

[d] Difference went to reprocessing

TABLE III. (cont.)

Country	Reactor type	Cladding type	Burnup GW·d/t U			Longest residence time (yr)	Inventory (Mg U)		Difference (Cumulative - present)	Active R&D	Comments
			Average	Typical	Maximum		Cumula-tive	Current 1/1/96			
Japan	BWR	Zry-2	11-37	20-35	39	22	6422	2756	3666	no	[d]
	PWR	Zry-4	15-32	15-46	47	17	4560	1861	2699	no	[d]
	GCR	MagnoxAL80	4	4	5.5	1	1251	25	1226	no	[d]
	ATR	Zry-2	16	17	33	16	144	96	48	no	Reactor FUGEN
Korea	PWR	Zry-4	30-35	33	45	16	1436	1436	0	yes	
	CANDU	Zry-4	6.8	7	7.5	12	1191	783	408	no	
Russia	WWER440	Zr-1%Nb	35	-	44	17	2730	230	2500	no	[e]
	WWER1000	Zr-1%Nb	35-42	-	50	10	1505	150	1355	yes	[e]
	RBMK1000	Zr-1%Nb	25	-	24.4	20	6540	6540	0	yes	
	BN-600	SS	60	-	-	7	-	42.5	-	yes	[e]
Slovakia	HWGCR	BeMg	4.8	4.8	5	22	80	20	60	no	[f]
	WWER440	Zr-1%Nb	25	30	38	12	785	700	85	no	[g]

[e] Spent fuel reprocessed after >3 years storage.

[f] Under decommissioning.

[g] 3 years storage in AR pool.

TABLE III. (cont.)

Country	Reactor type	Cladding type	Burnup GW·d/t U			Longest residence time (yr)	Inventory (Mg U)		Difference (Cumulative - present)	Active R&D	Comments
			Average	Typical	Maximum		Cumula-tive	Current 1/1/96			
Spain	PWR	Zry-4	33	35	47.9	10	1125	1026	99	no	
	BWR	Zry-2	25.1	27	35.6	13.5	81	433	352	no	
Sweden	LWR/PHWR	Zry-2	29	35	48	32	2214	2214	0	no	[h]
UK	MAGNOX	MagnoxAl80	5	abt. 6	abt. 8	5	40794	794	40000	no	[i,j]
	AGR	20/25/Nb/SS	18	22	27	15	2420	2272	144	yes	[k,l]
	PWR	Zircaloy	abt. 30	33	46	23	1343	1323	20	no	Various reactor sources
	BWR	Zircaloy	25	27	44	23	2211	2124	87	no	

[h] Fuel in Clab.

[i] AR fuel storage performance satisfactory.

[j] AFR fuel storage performance satisfactory, given adequate pond chemistry.

[k] AR, evidence of slight degradation due to sensitised cladding, dry storage is planned for 2 reactors.

[l] AFR, excellent experience with correct water chemistry control, fuel stored as complete FA.

TABLE III. (cont.)

Country	Reactor type	Cladding type	Burnup GW·d/t U			Longest residence time (yr)	Inventory (Mg U)		Difference (Cumulati-ve-present)	Active R&D	Comments
			Average	Typical	Maximum		Cumula-tive	Current 1/1/96			
USA	PWR	Zircaloy	29	38	55	27	19950	18850	1100	no	[m]
	PWR	SS	23	27	29	32	700	650	50	no	50 t reprocessed
	BWR	Zircaloy	26	34	48	30	11750	11660	90	no	90 t reprocessed
	BWR	SS	21	25	27	32	50	38	12	no	12 t reprocessed
	HTGR	n/a					24	0	24	no	In dry storage

[m] 100 t reprocessed, 1000 t stored.

TABLE IV. DRY STORAGE EXPERIENCE IN THE BEFAST-III COUNTRIES

Country	Reactor type	Storage type	Storage environment	Cladding type	Burnup GW d/t U			Longest residence time (yr)	Inventory 1/1/96 (Mg U)	Active R&D	Comments
					Average	Typical	Maximum				
Canada	CANDU	concr canister	Air	Zry-4	7 5	7	10 8	19	904	yes	
	Org cooled	concr canister	He	Zry-4 / Zr-2 5%Nb	8 6	5 5	11 5	19	24	yes	
	CANDU	Vault CANSTOR	Air	Zry-4	7 5	7	10	< 1 yr	113	no	
	CANDU	concr cont DSC	He	Zry-4				< 1 yr	7	no	
	Research	tile holes	Air	Al/Zr				> 30yr	96	no	
France	HWR	Vault	He	Zirconium		20	30	5	100	no	
Germany	PWR	metal casks	He	Zry-4	-	< 40	-	2	9	yes	[a]
	BWR	metal casks	He	Zry-2	-	< 40	-	2	2	yes	
	HTR	metal casks	He	Graphit	-	100 000	-	4	10	no	
Japan	BWR	metal casks	He	Zry-2	24	32	33	11	75	yes	4 months experience
Korea	CANDU	concr canister	Air	Zry-2	6 8	7	7 5	3	408	no	

[a] 2 Mg U in permanent storage

TABLE IV (cont)

Country	Reactor type	Storage type	Storage environment	Cladding type	Burnup GW d/t U			Longest residence time (yr)	Inventory 1/1/96 (Mg U)	Active R&D	Comments
					Average	Typical	Maximum				
Russia	V-1000	metal casks	He	Zr-1%Nb	27	-	-	1	-[b]	yes	[c]
	RBMK-1000[d]	-	Air	Zr-1%Nb	20	-	-	3	-	yes	
UK	MAGNOX	Vault	Air	Magnox AL80	4 5	5 6	8	about 8	643	no	[e]
	AGR SN	Vault	CO_2/Air	20/25/NbSS	-	-	-	17	-	yes	[f]
USA	PWR	NUHOMS	He	Zirc	30	35	35	7	490	yes	
	PWR	VSC	He	Zirc	27	29	30	5	150	yes	
	PWR	metal casks	He	Zirc	30	33	35	10	370	no	
	HTGR	Vault	He	n/a	-	-	-	2	24	no	

[b] Storage experiment only

[c] No changes during experimental storage

[d] Experiment on one fuel rod in hot cell

[e] Performance under normal conditions, also includes short-term storage under CO_2

[f] Prototype (WAGR) R&D programme to support commercial AGR dry storage installation

Both fuel and the environment was monitored in different programmes by using conventional and special methods In addition, development work on topics like corrosion monitoring was provided by the participants

Monitoring and surveillance confirmed that adherence to the specified pool water chemistry was essential to prevent fuel degradation during pool storage for all types of spent fuel Whereas near neutral pH conditions with low ion content are satisfactory for zirconium alloy clad fuels, elevated pH is beneficial for MAGNOX and AGR fuels For Al-clad fuels, a pH of 5 5 is in use

Defective fuel was in some cases stored in canisters, but no cladding deterioration, significant fuel losses or changes in defects were detected In pool components, liner and rack materials indicated no deterioration under proper water chemistry conditions

The following summaries of wet and dry storage performance of spent LWR fuel reflect the experience reported by the programme participants

5 1 SPENT FUEL WET STORAGE PERFORMANCE

Zirconium alloy clad spent fuel storage is reliable It is licensed in all BEFAST member countries Positive experience is available for up to 30 years storage — almost since the beginning of the peaceful use of nuclear power It can be expected that fuel integrity can be maintained for at least 50 years of storage However, positive spent fuel storage performance is closely related to maintaining specified pool water chemistry Even if the fuel assemblies contain operational defects, they can be stored in pools for extended time periods Recognizing this fact the licensed interim storage of spent fuel has no time limit in Sweden

For extended wet storage, the behaviour of different fuel materials (e g non-zirconium alloy cladding) and pool construction materials (e g epoxy lining) may require further investigations Capacity enhancement may require some additional work also (rod consolidation) Open questions may still exist in relation to extremely long water pool storage In this case, pool component performance could be more limiting than the FA storage performance Concerted efforts must be made in the field of corrosion monitoring and extremely long term prediction of fuel rod cladding and pool components behaviour in order to maintain good wet storage experience

5 2 SPENT FUEL DRY STORAGE PERFORMANCE

Zirconium alloy clad spent LWR fuel is stored in an inert gas atmosphere — preferably He — and is a proven technology for FA-hot spot temperatures up to about 450°C Dry storage of that fuel is licensed in Germany for temperatures 410°C and in the USA for T 380°C Experimental data is available up to more than 500°C To optimize the compliance with these temperature limits, the irradiation history and cooling time of the fuel must be considered, and well verified heat transfer codes for exact cladding temperature predictions are beneficial Positive experience with dry storage is available for storage periods of more than two decades Licensed storage periods of up to 40 years and for burnups up to 55 GW d/t U are available The storage of operationally defective zirconium alloy-clad spent fuel in an inert atmosphere is trouble free, providing that an air ingress can be prevented Degradation of FA due to the storage is unlikely In all operational applications, no significant fuel degradation in storage has been observed A few indications of krypton release have been observed indicating possible individual rod failures This suspect fuel, however, has not been examined to confirm the failures

In recent years, especially when considering rising spent fuel assembly burnup, the questions of allowable temperature was based on the allowable strain rates under storage conditions Having fixed the allowable strain (e g in a licensing process), the maximum allowable storage temperature was derived from spent fuel assembly EOL-conditions and the heat release characteristics of the spent fuel confinement

Storage of zirconium alloy clad spent LWR fuel in an oxidizing atmosphere is also being considered. Defect-free fuel should have adequate storage performance. However, the possibility of propagating of defects in operationally defective fuel limits the maximum storage temperature. Definition of the maximum storage temperature must consider the irradiation history, cooling time and proposed storage period.

Dry storage of spent MAGNOX fuel has been licensed in the UK (CO_2 or air, depending on fuel temperature) and in Italy (nitrogen). Difficulties are envisaged with dry storage after wet storage for irradiated MAGNOX fuel.

Dry storage will also continue to be used for Al-clad and MAGNOX-type fuels in the UK. Limited commercial nuclear power generation experience exists of inert gas storage for Al-clad fuel at ambient temperature.

5.3. CONCLUSIONS

The conclusions of BEFAST-III are:

The batch average burnup of PWR and BWR reloads is steadily increasing in the last decade from 40 GW·d/t U towards 50 GW·d/t U in most of the countries participating in the BEFAST.

Burnup increase results in:

- An increase in fuel rod internal pressure at the end of the service life;
- More zirconium alloy cladding corrosion due to longer in-service residence time;
- An increase in the cladding hydrogen concentration resulting from the increased Zircaloy corrosion.

Increased EOL pressure together with more corrosion results in increased stresses in the spent fuel rod cladding wall. It was concluded that this has no major consequences during wet storage. It has also been shown that under dry storage conditions for LWR fuel in an inert atmosphere the increased burnup has had no major consequences on spent fuel storage performance so far.

After 14 years co-operation under the IAEA umbrella, the fundamental R&D questions have been largely answered to the degree that licensing of both wet and dry storage is presently possible in most countries requiring additional spent fuel storage facilities.

The key objectives of spent fuel storage related R&D have changed over the years. Less effort is required to solve the basic material science questions. Work is now more focused on technological improvements as well as on economical improvements in storage. These changes indicate that spent fuel storage is becoming a mature technology in the back end of fuel cycle.

The effects of extended burnup on the behaviour of the fuel in storage conditions need to be assessed since there is a tendency in the Member States to achieve increasingly higher burnups.

Although there is positive storage experience so far, the extrapolation of current results for very long storage times (>50 years, exceeding the periods covered by the CRP) has yet to be confirmed.

Annex I

REPORTS ON RESEARCH PROJECTS WITHIN THE BEFAST-III PROGRAMME

TITLE **The integrity of spent CANDU fuel in the long term experimental AECL/Ontario Hydro Fuel Storage Program**

COUNTRY **Canada** CHIEF SCIENTIFIC INVESTIGATOR **K.M. Wasywich**
CONTRACT NUMBER **7020** COMPANY **AECL**

BACKGROUND In 1974, a concrete canister (CC) design and development program was initiated at AECL's Whiteshell Laboratories Since then, spent fuel dry storage in CCs has been licensed and the technology is now being used to store fuel at five sites in Canada To investigate the long term dry storage behaviour of spent fuel, AECL and Ontario Hydro initiated a long-term experimental program in CCs in 1978

OBJECTIVE The objectives of the Research Agreement are as follows
1 To continue monitoring the fuel in the dry-storage experiments,
2 To complete an examination of fuel stored for 7 8 years in moisture-saturated air at 150°C,
3 To continue to support research on UO_2 oxidation,
4 To initiate a program to model UO_2 oxidation of defective fuel stored in CCs

RESEARCH APPROACH Experiments are being conducted in CCs in dry air at seasonally varying temperatures and in dry and moisture-saturated air at 150°C Representative spent fuel bundles chosen for the experiments were characterized prior to storage and periodic examinations are performed to evaluate their storage integrity Because of the differences observed between the oxidation behaviour of spent CANDU fuel in dry air versus moisture-saturated air at 150°C in the canister experiments, laboratory experiments were initiated on unirradiated and spent CANDU fuel to improve our understanding of these differences A program was also initiated to model the oxidation behaviour of UO_2 in defected spent CANDU fuel based on the results obtained following 8 3 years of spent CANDU fuel storage in a limited-air environment at 150°C

RESEARCH RESULTS The canister experiments are continuing to be monitored To date, no release of radioactive material from the experimental storage containers to the interior of the canisters has been detected by the canister air monitoring system

Undefected fuel elements examined during the interim storage examinations have exhibited no apparent change form their pre-storage condition The amount of fuel experiencing UO_2 oxidation in the intentionally defected elements stored in moisture-saturated air at 150°C has been increasing progressively with cumulative exposure, however, the oxidation occurs only within a thin film on the surface of the UO_2 grains Although a significant amount of fuel in the intentionally defected elements has experienced UO_2 oxidation in dry air at temperatures up to 150°C, no apparent increases in fuel element diameters have been observed, and the structural integrity of the fuel cladding has been maintained

The influence of moisture on the air oxidation of unirradiated CANDU fuel at 200 to 225°C identified the following processes (a) oxidation dissolution of U(VI) and precipitation of hydrated UO_3, (b) back-reduction of dissolved U(VI) and precipitation of U_3O_8 on the UO_2/U_3O_7 surface, (c) solid-state surface and grain boundary oxidation of UO_2 to β- U_3O_7 , and (d) in wet oxidation conditions only, preferential dissolution of UO_2 grain boundaries Since the formation of U_3O_8 from aqueous solutions is rarely reported, this observation was documented in detail

The UO_2 oxidation model uses an equivalent porous medium representation of the fuel and describes the oxygen concentration in the fuel element using a reaction-diffusion equation The model assumes that the principal oxidation product at temperatures below 170°C is U_3O_7, this is consistent with the results of the ceramography and X-ray diffraction on the dry-air oxidation experiments Since the densities of UO_2 and U_3O_7 are nearly identical, oxidation of the fuel to these phases will not affect the geometry of the fuel nor increase the size of the defect in the Zircaloy cladding This work is ongoing

TITLE: **Behaviour of spent fuel and storage facility components during long term storage in Finland**

COUNTRY: **Finland**
CONTRACT NUMBER: **7021**

CHIEF SCIENTIFIC INVESTIGATOR: **E. Vitikainen**
COMPANY: **VTT Manufacturing Technology**

BACKGROUND:

In Finland, there are two 445 MWe PWR (WWER) units of Russian-Finnish design at the Loviisa power station of Imatran Voima Oy (IVO) and two 710 MWe BWR units of Swedish design at the Olkiluoto power station of Teollisuuden Voima Oy (TVO). Spent fuel of both power stations is stored in water pools many decades until it is disposed in the bedrock. Up to now the WWER spent fuel has been returned to Russia, but after 1996 also this fuel will also remain stored in Finland. More experience of wet storage is needed to ascertain the long term good performance of the spent fuel in the storage pools and the possibility of its handling after storage. Surveillance programmes are, therefore, going on in Finland to collect data on spent fuel and pool components' behaviour during storage.

OBJECTIVE:

Collection of spent fuel storage experience from the power companies in Finland and reporting of the experience to the BEFAST-III.

RESEARCH APPROACH:

Spent fuel surveillance consists of pool water surveillance and periodical spent fuel inspections. In addition, the condition of fuel is observed by the operators during handling, transfer and storage.

RESEARCH RESULTS:

Experience of 14 years storage of spent BWR-fuel and 7 years storage of spent WWER-fuel confirms the good performance of the fuel designs used. The experience is presented in the wet storage experience table (Table III.). No off-normal events have been reported.

TITLE: **Spent fuel structure corrosion during underwater storage**

COUNTRY: **France**
CONTRACT NUMBER: **7134**

CHIEF SCIENTIFIC INVESTIGATOR: **L. Baillif**
COMPANY: **NUSYS**

BACKGROUND:

In the framework of reprocessing contracts signed between COGEMA and French and foreign utilities (German, Japanese, Belgian, Switzerland, The Netherlands) spent fuel is stored in storage pools at La Hague waiting for reprocessing. Depending on water chemistry and quality, corrosion phenomena may have been induced at the reactor on same fuel structure components (namely sleeve, upper- and end-fittings of the fuel). At the La Hague AFR storage pools, fuel structure component corrosion has to be minimized in order to allow spent fuel handling and reshuffling prior to reprocessing.

RESEARCH APPROACH:

- Identify the fuel potentially affected by corrosion of structure components that may induce spent fuel embrittlement. A systematic analysis of the various materials used for fuel manufacturing was undertaken for each type of fuel unloaded and stored at the La Hague reprocessing plant. Special attention was given to the materials used for sleeve and upper- and end-fitting manufacturing.
- Define and propose countermeasures to limit corrosion propagation of the spent fuel structure components.
  Chemical and mechanical countermeasures were investigated. Chemical measures rely on a water quality control (chloride and sulfur contents have to be maintained below specific values). This will avoid any corrosion propagation. Mechanical measures are mainly based on reinforcement of spent fuel structure.

RESEARCH RESULTS:

- Upper- and end-fitting material specifications have been defined. Zircaloy or stainless steel with low carbon content (less than 0.005%) is required. In view of extended wet storage, more than 566 spent fuel assemblies received from French, Japanese and Belgian reactors were identified as doubtful regarding material specification. A conservative approach was adopted: for spent fuel that does not meet the material specification, fuel structure reinforcement was undertaken to guarantee spent fuel retrievability after a long-term storage period.
- Water chemistry was specified (with maximum chloride and sulfur content).

Parameter	Dimension	Acceptable limit
$[Cl^-]$ $[F^-]$ $[SO_4^{2-}]$	ppm	≤ 0.1
$[Na^+]$ $[Ca^{2+}]$	ppm	< 0.5
Conductivity	$\mu S/m$	3.5
pH		4.5 minimum

Keeping this water chemistry specification guarantees that sleeve corrosion that may have been induced at the reactor does not progress during the away-from-reactor storage period.

TITLE **Assessment of extended storage of spent LWR fuel**

COUNTRY **Germany** CHIEF SCIENTIFIC INVESTIGATOR **Dr. M. Peehs**
CONTRACT NUMBER **7022** COMPANY **SIEMENS AG**

BACKGROUND

Storage of spent fuel is a viable technology both in wet and dry modes Worthwhile storage experience is available in licensed wet and dry storage facilities Storage periods are licensed up to 40 years in several countries and in Sweden wet storage is licensed for an unlimited time period A review of different national approaches for extended fuel storage opens interesting possibilities for improving the nuclear fuel cycle The forecast of safe and reliable storage beyond the already licensed periods and burnups seems to be justified

OBJECTIVE

- Review FA degradation mechanisms,
- Assess the materials database,
- Review database of long term storage,
- Identify problem areas, if any

RESEARCH APPROACH

- Evaluation of the 18 years old wet storage surveillance programme,
- Assessment of identified potential FR failure mechanisms in case of increasing burnup

RESEARCH RESULTS

Wet storage performance of higher burn up spent fuel will scarcely be affected as already known since the storage temperature is low thus minimizing hoop stresses in the cladding In addition the stress dependant thermally activated cladding degradation mechanisms are practically frozen at typical pond water temperatures of around 40°C An 18 years surveillance program underlined wet storage performance Dry storage in comparison to wet storage faces a much more complex situation in case of high burn up spent fuel by 3 reasons

- Higher dry storage temperature,
- Higher fuel rod internal gas pressure,
- Reduced residual metallic wall thickness of the cladding EOL

Typically higher dry storage temperatures and higher fission gas pressure originates from higher EOL burnup and from higher average gas temperature in the fuel rod at the beginning of storage when using the same confinements as for lower burnup FA This will generate higher stresses and strain in the cladding

Since longer residence in the reactor tends to decrease the residual metallic wall thickness by corrosion of the cladding material, stress and strain in the cladding is amplified Higher storage temperatures in dry storage due to the higher decay heat which depends from the storage technology applied promote all thermally activated degradation processes Currently performed R&D work especially on post-pile creep strain provided the basis to license the CASTOR V/19 cask for transportation and storage of spent PWR fuel with a burnup less or equal to 55 GWd/tHM batch average However, limitations in the dry storage of spent fuel with even higher EOL burnup cannot be absolutely excluded with the present status of knowledge It should be mentioned that 55 GWd/tHM burnup is the highest value ever licensed so far for dry storage of spent LWR fuel

TITLE **Behaviour of spent fuel and storage facility components during storage**

COUNTRY **Hungary** CHIEF SCIENTIFIC INVESTIGATOR **G. Ferenczi**
CONTRACT NUMBER **7023** COMPANY **ETV-ERÔTERV Rt**

BACKGROUND

Four units of the Paks Nuclear Power Plant are now in operation, each with a WWER-440 type pressurized water reactor The fresh fuel is imported from Russia (previously from the Soviet Union) and, according to the agreement between the Operator of the plant and the Supplier, the spent fuel can be shipped back to Russia after 5 years of decay storage in the spent fuel pools of the plant

OBJECTIVE

The objective of the research is to study the behaviour of spent fuel and storage facility components during the storage in at-reactor pools, including

- Collection, systematisation and evaluation of data concerning,
  - thermal behaviour of spent fuel storage pools
  - chemical composition and pollutants in the pool water
  - isotope content and specific activity
- Visual control of storage structures and the fuel,
- Correlation of primary coolant and pool water activity, taking into account the results of fuel failure detection results

RESEARCH APPROACH

Aiming to evaluate physical and chemical conditions in the storage pools the following parameters are monitored temperature (°C), pH, optical transparency (%), chemical composition (Cl, H_3BO_3), and isotope-specific activity (Bq/kg)

The overall limitations ($t < 70$ °C in case of refuelling operations, < 60 °C in case of storage, $H_3BO_3 > 12$ g/kg) are continuously controlled. Chemical conditions are measured and corrected if necessary at least daily during refuelling operations, and weekly during storage, the boron concentration is monitored continuously

As routine measurements, the following activity data are measured

- corrosion products (^{51}Cr, ^{54}Mn, ^{59}Fe, ^{58}Co, ^{60}Co, ^{110m}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 measurement of activity is carried out -as a rule- once in month, during refuelling measurements are taken more frequently

RESEARCH RESULTS

When summarising the experiences it must be taken into account that in the timeframe of the research there was no detectable fuel failure, that means there was no spent fuel bundle stored in hermetic casing in the storage pool

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

- Differences in pool water temperature cannot be evaluated because of the good thermal regulations of the pools Temperatures are stable around 28-30 °C mean value in the case of both the original storage concept and compact storage
- 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 iron content in every case was below the detection limit (0 05 mg/kg)
- ^{51}Cr, ^{91}Sr, ^{92}Sr, ^{59}Fe, ^{124}Sb, ^{140}La and ^{138}Cs isotopes are missing or are of minor importance The dominating activity component is ^{110m}Ag The activity levels can be managed safely using pool water purification
- The main sources of activity in the storage pools are surface and water contaminants Activity in the pool water probably transferred there during refuelling periods Corrosion processes during storage periods are less important in the activity build-up processes

TITLE:
Development of Under Water Remote Welding System for Pool-Lining

COUNTRY: Japan
Contract number: No.7024

Chief Scientific
Investigator/ Toru Onishi
Company: PNC

BACKGROUND:
If leakage of water would take place at the stainless steel pool-lining of fuel storage pool because of the unforeseen accident, under water remote welding system to repair the pool-lining is required. However, such a system is not available now. PNC planned to develop the under water remote welding system for repairing pool-lining of fuel storage pool of Tokai Reprocessing Plant.

OBJECTIVE:
-To develop the under water remote welding system for repairing pool-lining of fuel storage pool of Tokai Reprocessing Plant.

RESEARCH APPROACH:
-To design and manufacture the under water remote welding system for repairing pool-lining of fuel storage pool of Tokai Reprocessing Plant.
-To confirm the performance of the under water remote welding system by mock-up test at the depth of 10m under water.

RESEARCH RESULTS:
Under water remote welding system, which consists of the welder, the inspection device and the manipulator, was designed and manufactured. This system is designed to weld around the patch for repairing the leaked pool lining under the water depth of 15m, and to inspect on the weld zone by CCD camera and ultrasonic flaw detector after the welding .
The mock-up test showed that this system could weld and inspect the pool floor and wall lining under the water depth of 10m The visual test, radiographic testing, penetrant testing, ultrasonic testing and microscopic test on the weld zone of test pieces showed that the quality of the weld under water was equivalent to that under air

TITLE

Behaviors of Intentionally Defected PWR Spent Fuel Rods in Pool Storage

COUNTRY	KOREA	Chief Scientific	Dr. S.G. Ro
Contract Number	7025	Investigator/ Company	KAERI-NEMAC

BACKGROUND

Research program on the long-term storage behaviors of the defective fuels in pool water was initiated in 1988. Since then, 10 fuel rods among which 7 were intentionally made thru-hole defective have been examined non-destructively and/or destructively on the periodic basis The first and second examinations were performed in 1989 and 1990, respectively

The third examination on the spent fuel rods stored in pool water for 5 years, was carried out in 1993 Summary of the third examinations is given here

OBJECTIVE

Understanding of the storage behaviors of the defective spent fuel rods

RESEARCH APPROACH

- Preparation of fuel rod for examination
  seven PWR fuel rods were intentionally made defective After making defects on the fuel rods, all the fuel rods were examined by NDT. and then have been stored in the PIEF pool of KAERI since February 1988
- Hot cell examination
  . non-destructive examination
  dimensional changes around defects
  . destructive examination
  micro-structural changes of fuel cladding and pellet around defects
- 1st examination in 1989 (1-year storage after making defects)
  2nd examination in 1990 (2 5-year ")
  3rd examination in 1993 (5-year ")

RESEARCH RESULTS

- No change in defected geometry was observed in the fuel rod stored in pool for 5-year after making defects
- Unidentified yellow color material was observed to be stuck around the defect area

TITLE

Zircaloy-4 Cladding and UO_2 Oxidation Behaviors in Air

COUNTRY	KOREA	Chief Scientific	Dr. S.G. Ro
Contract Number	7025	Investigator/ Company	KAERI-NEMAC

BACKGROUND

The oxidation behaviors of Zircaloy-4 cladding and UO_2 are very important for the evaluation of the storage temperature limit of the spent fuel in air environment. This work is to study the oxidation behaviors of Zircaloy-4 cladding with the different preoxide-film and the UO_2 oxidation to U_3O_8, which could split the cladding.

OBJECTIVES

- Evaluation of the storage temperature limit in air environment
- Evaluation of the time and temperature for U_3O_8 formation of UO_2

RESEARCH APPROACH

- Un-irradiated samples were taken from PWR type fresh fuel
- Irradiated samples were taken from PWR spent fuel rods
  . UO_2 fragment samples with different burnup
  Zircaloy-4 specimen with different oxide layer thickness
- Un-irradiated and irradiated samples were oxidized in a furnace, and then the sample weight was measured by a continuous and an interrupt weighing method
  test temperature . 150-350℃ for UO_2 and 450℃ for Zircaloy-4 clad
- Samples are examined, photographed, and analyzed by XRD to observe the U_3O_8 formation.
- Burnup and pre-oxidation effects on oxidation are evaluated.

RESEARCH RESULTS

- Un-irradiated UO_2
  there are no significant differences in oxidation of UO_2 samples with different aging histories.
  the rate equation for U_3O_8 formation derived is as
  Log t = *-12 89 + 7650/T(K)* for 150-350 ℃, and the activation energy for the reaction is obtained to be 145 kJ/mole.
- Irradiated UO_2 and Zircaloy cladding
  there is a significant difference in oxidation of gadolinia and UO_2 fuels
  oxidation of Zircaloy cladding with white-oxide(post-transition) is faster than that of the cladding with black-oxide(pre-transition) or transition oxide

TITLE

Adsorption and Corrosion Behavior on Stainless Steel during Storage in Pool

COUNTRY	KOREA	Chief Scientific	Dr S G Ro
Contract Number	7025	Investigator/ Company	KAERI-NEMAC

BACKGROUND

Several types of stainless steels have been the predominant materials for the pool liner, storage racks, pipings and fuel handling equipment They have functioned well, but their adsorption and corrosion behaviors should be investigated to predict the long-term integrity of the storage facility

OBJECTIVES

- Investigation of the adsorption and corrosion behaviors of 304 304L, and 316 stainless steels in the storage pool

RESEARCH APPROACH

- A periodical analysis of pool water of post-irradiation examination facility (PIEF) of KAERI
  corrosion products(Cr, Ni, Fe) and Halogen (F, Cl)
  gamma radioactivity (Co-60, Cs-134, Cs-137)
  pH and temperature
- A number of stainless steel samples with different materials are stored in the PIEF pool
  materials 304, 304L, 316 stainless steel
  pre-treatment non-trearment, polishing, pre-oxidation, heat-treatment (sensitization)
- Periodical measurements of the radioactivity adsorbed in sample by well type HPGe gamma detector
- Periodical measurements of corrosion rate by electrochemical method

RESEARCH RESULTS

- The Cs-137 adsorption rate of the stainless steels stored for 18-month in pool appears to be about 78 pCi/cm^2-month for 144 pCi/ml of Cs-137 concentration in pool water, while that of Co-60 seems to be about 7 pCi/cm^2-month for 27 pCi/ml of Co-60 concentration in pool water
- The preliminary results indicate that gamma irradiation enhances the corrosion of stainless steel
- The corrosion rates of stainless steels stored for 18-month in pool are very small (10^{-5} - 10^{-4} mm/year) regardless of stainless steel types and pre-treatment histories

TITLE **Investigation of the LWR Spent Fuel Behaviour During Long-Term Storage in Russia**

COUNTRY	Russia	**Chief Scientific Investigator**	Dr I Kadarmetov
Contract number	7027	**Company**	All-Russia Scientific Research Institute of Inorganic Materials

BACKGROUND

The nuclear power in Russia is based on three types of water cooled reactors VVER-440,-1000, and RBMK-1000 The fuel strategy of nuclear power plants that assumed the closed fuel cycle for all types of reactors has recently undergone significant changes As before, only VVER-440 spent fuel are subject to reprocessing Based on economic efficiency the solution was taken not to reprocess of RBMK spent fuel Interim and long-term dry storage of VVER-1000 fuel is under consideration

OBJECTIVES

- Review the potential fuel degradation mechanisms of Zr-1%Nb cladding and their relevance for fine tuning of allowable temperatures in dry storage
- Review the safe storage criteria of long-term storage in an inert gas

RESEARCH APPROACH

- Experiments to determine the strength of Zr-1%Nb cladding with purposely made defects at the inner surface in presence of aggressive fission products
- Examination of post-irradiation cladding thermal creep
- Verification of the calculated codes for fuel cladding failure resistance under dry storage conditions

RESEARCH RESULTS

Post-irradiation thermal creep and iodine SCC of Zr-1%Nb cladding were studied experimentally The finite elements program was developed to assess the fuel rod temperature conditions in gas filled casks

A method is suggested to determine the time of safe long-term storage of spent VVER-1000 fuel takes into account the pre-history of fuel assembly irradiation and the preliminary cooling in water pools The safe storage criteria are assumed to be

- hoop creep strains are not to exceed 2% limit,
- elimination of a through penetration of a corrosion defect from inner to outer fuel cladding surface

The preliminary assessment of the maximum allowable temperature of the VVER-1000 spent fuel dry storage in inert gases shows that the initial temperature of fuel must be within 375-400° C

TITLE: **Assessment of Extended Storage of Spent LWR Fuel**

COUNTRY: **Russia**

CHIEF SCIENTIFIC INVESTIGATOR: **V.G. Kritskij**

CONTRACT NUMBER: **7026**

COMPANY: **VNIPIET**

BACKGROUND:

In Russia wet storage of spent fuel is a prevailing technology. Experience with RBMK spent fuel storage up to 20 years is available and justification of reliable and safe storage for up to 50 years is needed. In addition, the need for longterm storage of WWER-1000 spent fuel has arisen. Spent fuel shows residual energy release that produces changes in the storage medium and in this way intensifies the corrosion of fuel assemblies and water pool components. The result is accumulation of radioactive corrosion products, which means potential threat to the hydrosphere.

OBJECTIVE:

- Review of fuel assembly degradation mechanisms;
- Review of the database of long term storage;
- Identification of the problems:

    - Monitoring of fuel integrity while storing large fuel quantities;
    - Increasing spent fuel storage density;
    - Storing defective fuel.

RESEARCH APPROACH:

Spent fuel surveillance consists of pool water surveillance and periodical spent fuel inspections. In addition, the condition of fuel is observed by the operators during handling, transfer and storage.

RESEARCH RESULTS:

Experience of 14 years storage of spent BWR fuel and 7 years storage of spent WWER fuel confirms the good performance of the fuel designs used. The experience is presented in the wet storage experience table (Table III.). No off-normal events have been reported.

Maximum corrosion occurs at the initial storage stage in small water volumes (cans) at the points of Zr/SS contact (under spacers). Corrosion of SS pool components amounts to 1 μ/year.

A technique has been developed for assessing the fuel element integrity by calculating the ratio of dissolved ^{137}Cs and ^{134}Cs in water. Gaining the licence for increasing the storage density from 3 to 6 t/m^2 for the 40-year storage period is the main result of investigation.

TITLE **Assessment of long-term storage of spent fuel from water cooled and moderated reactors (WWER - 440)**

COUNTRY **Slovak Republic**
CONTRACT NUMBER **7069**

CHIEF SCIENTIFIC INVESTIGATOR **J. Kmošena**
COMPANY **SE - NPP Jaslovské Bohunice**

BACKGROUND

Wet storage has been proved as useful safe technology for storage of spent fuel from nuclear power reactors This option of intermediate storage provides sufficient time for thorough future considerations in the back-end of fuel cycle strategy, keeping all alternatives open and available

OBJECTIVE

- Spent fuel and storage system behaviour investigation at WWER-440 spent fuel AFR wet storage facility
- Hydrogen generation study in wet loaded C-30 casks

RESEARCH APPROACH

Evaluate AFR wet storage facility operation and surveillance data Perform laboratory experiments and special measurements Visually inspect stored spent fuel and storage system components Perform experiments with transport cask C-30

RESEARCH RESULTS

There were no off-normal events or incidents during AFR spent fuel storage facility operation since 1987 90% of storage capacity (600 Mg U) is already used Water chemistry in AFR storage pools is maintained within specified range without any difficulties

Water activity shows a trend of slight increase, mainly due to Caesium isotopes entering pool water through fuel cladding leaks at a few leaking spent fuel assemblies Periodically started, ion-exchange resin based cleanup system effectively reduces water activity

Biological species growth has occurred because of favourable conditions (temperature, low flow) in storage pool water Micromycetes, psychrophylic and mesophylic bacteria were found as major part of the biomass, no coliform and only few corrosive bacteria were found Biomass was mostly localized in empty pools filled with water but without fuel (inspection pool, emergency pool) where is no water circulation most of time Another source -mixed-bed ion-exchange filter- was identified, where because of long standby periods, biomass growth has favourable conditions Hydrogen peroxide treatment laboratory tests have been performed, but have not been considered as optimal solution of this problem Mechanical methods (skimming, filtration) have been effectively used for biomass removal

Visual observation of the oldest stored spent fuel assemblies using an underwater camera did not reveal any signs of corrosion neither on the hexagonal zirconium shroud tube nor at stainless steel top and bottom end fittings Visual inspection of pool stainless steel liner weld areas (as the most sensitive region) after recognizing biological growth did not discover any corrosion attack

In wet loaded C-30 transport casks, concentration of radiolytically generated hydrogen in the gaseous phase reached, after 30 days, 0 8 vol % without and 0 12 vol % with catalytic recombiners installed at the cask lid

TITLE **Fuel performance behaviour in wet and dry storage conditions**

COUNTRY **Spain**
CONTACT NUMBER **7356**

CHIEF SCIENTIFIC INVESTIGATOR **J.M. Gravalos**
COMPANY **ENRESA**

BACKGROUND

Most of the R&D work on spent fuel storage dates back to the 1970s, when investigations on cladding and overall fuel performance were carried out with the main target of achieving high burnups Since then, fuel elements stored at the pools are monitored regularly both in a direct way (sipping, eddy current) and indirectly through periodic checking of the pool chemistry All failed fuel stored at the pools are fuel elements damaged during their irradiation, except for a few PWR fuel elements who suffered damage in the connection of their upper end fittings to the guide tubes BWR failures are due to PCI, CILC, debris, etc , whilst PWR ones are mostly due to debris, grid fretting, baffle jetting, clad corrosion, etc

Dry storage is expected to take place around 1999 in dual purpose metal casks Six complete fuel rods with burnups around 55 GWd/tU have been stored in air since 1972 The rods were completely characterized at that time and show no visible defects There are presently plans to perform a limited research programme on these fuel rods to establish their performance behaviour after more than 20 years of dry storage in an oxidizing environment (air)

OBJECTIVES

To determine the integrity of the fuel after extended dry and wet storage

RESEARCH APPROACH

Mostly by visual methods, all fuel elements suspected to have failed during their irradiation are inspected when they are discharged from the reactor core One plant performs routine determination of the cladding corrosion on all their fuel elements if they are to be reloaded in the core

RESEARCH RESULTS

No cladding damage has been observed in any fuel element during storage in the pool The upper-end fitting problem was due to a combination of the material and chemistry of the pool and has not been reported again The overall cladding behaviour has proven to be excellent in wet conditions

The investigations concerning dry storage have not been finalized yet Results are expected from a non-destructive (visual inspection, cladding oxide thickness measurement and eddy current testing) and destructive (cladding metallography, hydrogen content axial tensile testing, etc) examination programs

TITLE: Work in support of the storage of AGR fuels

COUNTRY: UK **COMPANY: BNFL**

BACKGROUND:

The early storage regime for AGR fuel was found to be detrimental to the integrity of fuel pins and large numbers of pin failures were reported due to the high levels of chloride ions resulting from site atmospheric conditions. To combat this problem investigations into corrosion inhibitors were initiated and sodium hydroxide identified as a suitable material Since the introduction of caustic dosing no further pin failures have been reported A watching brief is kept on this situation and an activity release model has been developed as an effective means of supporting this work

As dosing in the main THORP storage pond is incompatable with the storage of LWR fuel, methods have been developed to store AGR fuel in sealed containers to isolate any activity arisings due to pin failures As the fuel is stored in deionised water the effect of this deviation from its normal pond water chemistry needs to be evaluated

OBJECTIVE:

- develop a model to predict activity releases to AGR storage ponds
- determine if any further degradation of AGR fuel is occurring
- determine the effect of deviations from optimum pond water chemistry

RESEARCH APPROACH:

The activity release calculated by the model is compared with the calculated cumulative ^{137}Cs arisings derived from measured pond water activity levels and the weekly pond purge volume The model has been tested on the last ten years of plant records

Predosed AGR fuel sealed in a deionised water environment is subjected to repeated chloride ion dosing The effects are measured by liquor sampling

RESEARCH RESULTS:

The predicted and actual activity arisings in the AGR storage pond are in very good agreement suggesting that further AGR fuel pin perforations are not occurring

There have been no indications of fuel failures even after prolonged storage periods in chloride ion concentrations up to 30ppm

TITLE: Work in support of the storage of LWR fuels

COUNTRY: UK **COMPANY: BNFL**

BACKGROUND:

Storage of LWR fuels at Sellafield dates from the early 1970s when fuel was received in basketed wet transport flasks and then transferred into open topped pond storage skips. Some of these early receipts still reside in these older forms of pond storage container and have experienced the flutuations of pond water chemistry over the years; including the effects of biocide dosing. In order to prepare these fuels for reprocessing in THORP rebottling into MEBs is necessary to comply with the storage regime and handling/transport arrangements.

The integrity of long stored fuel both in terms of fuel in open topped skips for rebottling and long stored fuel in MEBs for reprocessing has safety implications in respect of fuel drops; especially considering the reported instances of PWR top nozzle separation in some Westinghouse designed/manufactured fuels.

OBJECTIVE:

- determine the integrity of long stored fuels
- establish if there are any corrosion problems with respect to PWR top nozzles

RESEARCH APPROACH:

Visual inspections using CCTV on PWR, BWR and SGHWR assemblies having cooling times up to 23 years. Both containerised and fuel open to the bulk pond water were included. Each assembly was assessed for structual integrity, component oxidation, crud levels and for general deterioration due to long term storage.

RESEARCH RESULTS:

Visual inspections have reported large numbers of failed pins in some long cooled fuels. Shipping records for these assemblies have shown these failures to be in reactor and not as the result of long term storage. It is inconclusive whether prolonged storage in bulk pond water conditions is propogating these failures, but none were considered to have structual defects which would impair fuel handling. In the case of one assembly (stored in an MEB) known to have areas of corrosion these were assessed to not have deteriorated since the last inspection nine years previous.

TITLE: **Spent fuel storage experience of Magnox and AGR fuel by Nuclear Electric plc and supporting technical studies**

COUNTRY: **UK** CHIEF SCIENTIFIC INVESTIGATOR: **K.A. Simpson/G. Knowles**
CONTRACT NUMBER: **7028** COMPANY: **Nuclear Electric**

BACKGROUND:

Nuclear Electric plc monitors the behaviour of fuel from its MAGNOX and AGR power stations during storage on reactor sites. This covers wet storage for AGR and both wet and dry storage for MAGNOX. Monitoring is supported by work concerned with the susceptibility of fuel and cladding to degrade during storage under normal and fault conditions, this is relevant to the specification of acceptable conditions for very long term storage.

OBJECTIVE:

MAGNOX fuel:
- Continued monitoring of fuel behaviour in wet storage to include water chemistry and activity release;
- Continuation of the current database on dry storage experience.

AGR fuel:
- Continued monitoring of fuel behaviour in wet storage to include water chemistry, activity release and fuel examination;
- The development of irradiation induced microstructural changes in AGR cladding and the corrosion/cracking susceptibility.

RESEARCH APPROACH:

MAGNOX and AGR power stations have on-site spent fuel storage facilities which are regularly monitored for compliance with operational specifications. Endoscope facilities are available for detailed fuel inspection in AGR on site fuel ponds.
Comprehensive shielded facilities are used for examining irradiated fuel and for experiments on the corrosion/oxidation behaviour of irradiated fuel and cladding samples. A range of electron-optical techniques is available for the microstructural characterisation of irradiated materials.

RESEARCH RESULTS:

Wet storage of AGR and MAGNOX fuel during short term storage station ponds is good provided that water chemistry is controlled. The dry buffer storage of MAGNOX fuel in air has been excellent under controlled atmosphere conditions.
Analytical electron microscopy has been used to determine compositional profiles at grain boundaries in irradiated AGR fuel cladding. There is evidence for profile development during the early stage if irradiation in the temperature range 350 - 520 °C. The most severe effects are found at 400 - 420 °C but these can be alleviated by annealing at 600 °C.
Experiments using peak sensitised AGR cladding have quantified the corrosion cracking of this material in moist air environments. This work has established the influence of tensile stress and air moisture level on corrosion cracking rates during transients from nominal dry storage.

TITLE **Behaviour of spent fuel and storage facility components during storage**

COUNTRY **USA** CHIEF SCIENTIFIC INVESTIGATORS **Jeffrey Williams/ Leroy Stewart-USDOE**

CONTRACT NUMBER **7029** **Ray Lambert - Electric Power Research Institute**

BACKGROUND

The USA has broad experience in both wet and dry storage of LWR spent fuels Storage technologies have now been developed, tested and put into routine plant applications The current focus of spent fuel R&D has shifted away from system development towards documentation of long term system performance Additionally, the one area of technology development that is being pursued in the USA is the design and implementation of an MPC (Multi-Purpose Canister) system

OBJECTIVE

- Report on storage experience in the USA, including testing of new metal cask systems and concrete cask systems,
- Develop a practical design to effect dry transfer of spent fuel from cask to cask,
- Validate shielding codes by comparison of calculated data with actual test measurements,
- Develop the technical basis for burnup credit,
- Evaluate welding techniques for joining borated stainless steel

RESEARCH APPROACH

Research reported under this contract has, in general, been undertaken by DOE and EPRI on jointly sponsored projects Results are often documented in EPRI reports The actual research work is contracted out to a variety of organizations, including utilities, national laboratories, and commercial vendors

RESEARCH RESULTS

- Storage Experience Experience in the USA continues to be excellent LWR fuel in dry storage was about 1034 MTU as of 1 January 1996, and is expected to grow by about 200 MTU/yr for the next few years Pool inventory stood at slightly above 32 000 MTU Test experience with dry storage in a new metal cask (NAC-128) was documented in EPRI report TR-101091 and results of the concrete VSC demonstration in TR-100305 Long term dry storage behaviour was reported by Leroy Stewart in the paper "Spent Nuclear Fuel Behavior in Long-Term Dry Storage"at the International HLW Conference in Las Vegas, Nevada in May 1995
- Dry Transfer of Spent Fuel Transnuclear, Inc has completed a detail design of a transfer system as part of a cooperative DOE/EPRI program Results are documented in EPRI Report TR-105570 dated December 1995 Shielding Code Validation Oak Ridge National Laboratory has completed the project of benchmarking calculated gamma and neutron doses from a filled metal spent fuel storage cask with actual measured values Results are in EPRI Report TR-104329
- Burnup Credit Sandia National Laboratories has prepared a Topical Report that has been submitted to the USNRC establishing the licensing basis for awarding burnup credit This report is designated "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages," DOE/RW-0472 Rev 0 Additionally, the demonstration of a forked detector to validate the burnup of spent fuel has been documented in EPRI Report TR-103591
- Welding of Borated Stainless Steel Various joining techniques were evaluated by Sandia National Laboratories and the parameters of welding necessary to ensure high strength welds investigated Although the project was terminated prior to completion, the results are contained in EPRI Report TR-104627