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Long term storage of spent nuclear fuel — Survey and recommendations

Final report of a co-ordinated research project 1994–1997

INTERNATIONAL ATOMIC ENERGY AGENCY

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FOREWORD

The management of spent nuclear fuel is a major consideration in the nuclear power industry and an integral part of the nuclear fuel cycle. Spent fuel is a residual product of nuclear electric power generation. The length and method of storage is a topic of ongoing discussions within the utility industry because of economic, licensing and societal issues associated with the development or re-licensing of storage facilities. Also, an increasing number of Member States view spent fuel as a valuable energy resource that should be maintained for future use to sustain the development of nuclear energy. In several Member States, storage facilities have been initially licensed to operate for periods up to 50 years. In many cases periods up to 100 years or longer are now anticipated. Finally, delays in decisions on final disposal options have already increased the demand for spent fuel storage capacity and this trend is expected to continue.

Under a new work programme to address the trends in extended duration of spent fuel storage and the related potential technological and regulatory impacts, two Technical Committee meetings (TCM) were held. The TCM on Good Practices on Long Term Storage of Spent Fuel Including Advanced, High Burnup and MOX Fuel was held in November 1999, and the TCM on Requirements for Extremely Long Term Storage Facilities was held in October 2000. The results of both TCMs are included in this TECDOC.

This work complements the IAEAs programme on safety of spent fuel interim storage. Three safety publications were prepared on the subject: Design of Spent Fuel Storage Facilities: A Safety Standard, Safety Series No. 116, Operation of Spent Fuel Storage Facilities: A Safety Standard, Safety Series No. 117, and the Safety Practice on Safety Assessment for Spent Fuel Storage Facilities: A Safety Practice, Safety Series No. 118. These three publications contain detailed guidance and are applicable to the facilities covered by this publication.

The preparation of this TECDOC was recommended by the members of the IAEA Regular Advisory Group on Spent Fuel Management at its meeting in 1997.

The IAEA wishes to express appreciation to all the experts for their contributions in the meetings and in particular V. Fajman (Czech Republic), W. Goll (Germany), A. Marvy (France), P. Standring (United Kingdom) and L. Stewart (United States of America) for their assistance in the preparation of this TECDOC.

The IAEA officer responsible for this publication was P. Dyck of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

1. RESULTS OF GROUP DISCUSSIONS

1.1. INTRODUCTION

Two Technical Committee meetings were held at the IAEA in Vienna to share the experiences of Member States with regard to extended or long term storage of spent nuclear fuel. Potential technological, regulatory and societal concerns and impacts associated with storage were discussed. Participants presented papers during these two meetings covering all the relevant issues. Smaller discussion groups were formed which enabled the participants to provide detailed information within the context of four general subjects:

1. Long term behaviour of spent fuel.
2. Long term behaviour of dry storage systems.
3. Wet spent fuel storage facilities.
4. Regulatory concerns related to long term spent fuel storage.

The nuclear industry worldwide has accumulated significant fuel storage operating experience over the past decades. This experience, however, is largely based on safe and effective wet storage and the effect of time on structures and materials during this limited period of time. The new challenges are to extend the life of existing and new wet and dry storage facilities and guarantee their safe performance for much longer periods of time.

Participants in each group have therefore reflected on past experience in order to define the issues and questions, which should to be addressed through future research and development (R&D). The following pages summarize their contributions.

1.2. LONG TERM BEHAVIOUR OF SPENT FUEL

The prediction of the integrity and retrievability of spent fuel constitute the main discussion topics for spent fuel behaviour regardless of the storage system and time period envisaged.

Retrievability is strongly dependent on the conditioning route for the fuel after storage, individual licensing situation, and licensing practices in Member States, and characteristics of the fuel (e.g. type of defects).

Therefore, requirements may depend on the ultimate back end solution for the fuel. Nevertheless, an aspect of retrievability is the integrity of the spent fuel including its structural components.

The mechanisms that might have the potential to degrade the fuel and fuel structure were reviewed to identify possible gaps in knowledge, especially with respect to the long term behaviour of the materials during storage. Cask storage, in comparison to all other storage techniques, presents the greatest challenge (stress/strain) to long term fuel cladding performance, as a result of the high initial operating temperatures during the early years of storage. Figure 1. shows the interaction of the material properties, in-pile and post-pile behaviour, and cask temperatures to calculate strain and stress of the cladding. Stress and strain and the approach to the stress limit are the most important criteria in assessing cladding integrity. For calculating the stress/creep behaviour, a creep law for the cladding material, temperatures in the cask, and in-pile data such as fuel rod design data and gas contents in the rod free volume are needed. Furthermore, assumptions on the post-pile behaviour of the fuel have to be made with regard to long term gas retention at low temperatures.

The available data with respect to materials performance in extended storage periods was discussed in relation to creep tests and surveillance programmes. The findings are valid for intact fuel under normal conditions. Fuel with small defects can be stored under normal conditions without encapsulation, however an appropriate drying procedure is needed to ensure that moisture content is minimised in the failed fuel rod. Fuel with large defects in general should be encapsulated to provide containment.

The following conclusions and recommendations were derived:

1.2.1. Conclusions

(1) The creep strain capability has to be known for all materials used. Creep tests should be done under realistic conditions (irradiation, oxidation, hydrogen, local effects etc.) to assess the straining capability,

(2) A representative creep law is needed. Dependent on the margins, irradiated or non-irradiated material can be used.

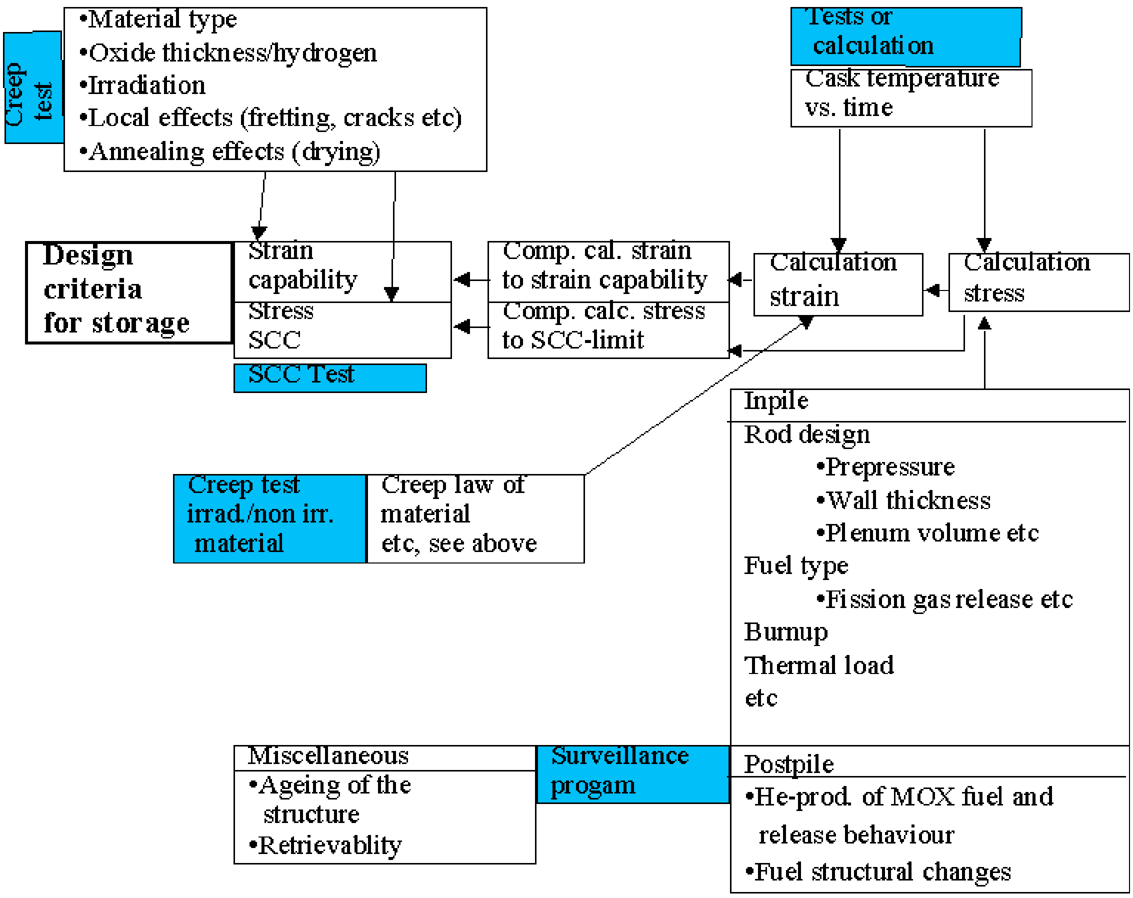


FIG. 1. Interaction of materials' properties, inpile and postpile behaviour to assess fuel integrity under normal conditions.

1.2.2. Recommendation

To assess the long term effects, for example the interaction of different materials and low temperature gas release, an international surveillance programme on the spent fuel behaviour under dry storage conditions is proposed. This programme should monitor long term behaviour to assure that no unexpected phenomena occur and to verify the predicted behaviour.

1.3. LONG TERM BEHAVIOUR OF DRY STORAGE SYSTEMS

For implementation of the of long term storage for periods up to 100 years and beyond the scientific, technical and regulatory communities have to extrapolate existing views and practices to safely operate and control the nuclear energy cycle. It also creates new challenges for Society and policy makers.

The key conceptual aspect of the long term storage is that it must not be regarded as a final disposal option or solution. This entails the capability to safely re-handle the spent fuel at any point in time after initial storage.

Through discussions it quickly appeared that key assumptions had first to be made on the following points in order to set the appropriate context for the assessment of the long term behaviour of a dry storage system:

(1) A certain amount of relevant technical knowledge pertaining to the nuclear business must be secured throughout the periods of time envisaged for the operation,
(2) Expertise as well will be needed, which raises the question on how to maintain it,
(3) Funding will also have to be secured,
(4) Sufficient general industrial infrastructure must be kept operational:
(5) Relevant data and information pertaining to the storage system, including spent fuel packages stored in it, must be kept in a secured way, to be readable and usable much later.

Key design principles have been then identified and are as follows:

(1) The destination of spent fuel at the end of the storage period (options should be unknowingly foreclosed: geological disposal, reprocessing, others) should be known, if possible, and considered in the storage system design,
(2) Retrievability is a key function for long term storage and a safety goal,
(3) The storage design must incorporate a certain degree of flexibility to adapt to future needs and conditions,
(4) Safeguarding the storage facility for long periods of time is a prerequisite condition,
(5) Total cost to construct and operate the facility must be affordable now and in the future,
(6) The operating system must make use of as many passive components as practicable during the storage phase,
(7) Maintenance needs have to be evaluated and considered at the design stage,
(8) The overall environmental assessment must lead to a prediction of cumulative impacts over time, which meet or exceed current standards and complies with the ALARA principle,
(9) The long term storage facility should be publicly acceptable.

The following Tables I to VIII summarize the consensus views of the working group with respect to the importance of specific technical issues in relation to the long term storage concept, and whether they are currently understood in a satisfactory manner or would require additional R&D work. An R&D programme to support extended storage periods can be developed from the areas of lack of knowledge given in the tables.

Table I. Storage conditions

	Item	Known	Partly known	Unknown
1.	Safety case	X		
2.	Duration		X	
3.	Heat Removal		X	
4.	Temperatures		X	
5.	Monitoring		X	
6.	Radiation protection	X		
7.	Horizontal or vertical storage	X		
8.	System type (Cask, Silo, Vault)		X	
9.	Indoors or Outdoors	X		
10.	Handling systems		X	
11.	Modelling measurement		X	
12.	Contingency accident / incident	X		
13.	Export /import facility	X		
14.	Inspection / repack?		X	
15.	Size configuration		X	
16.	Contingency measures		X	
17.	Decommissioning + Activation of mat.		X	
18.	Waste arising + costs		X	
19.	Program–endpoints–review–Decision Point		X	

Table II. Long term behaviour of materials (for safety function)

	Item	Known	Partly known	Unknown
1.	Neutron absorber			X
2.	Gasket		X	
3.	Basket		X	
4.	Welds		X	

Table III. Fuel/Cladding defects

Item	Known	Partly known	Unknown
1. Failure in storage		X	
2. Failure extension		X	
3. Tolerance to failure –criticality –retrievability		X	
4. Response times		X	

Table IV. New technology impact + Incorporation of lessons learned

Item	Known	Partly known	Unknown
1. Regular review process	X		
2. Technology watch	X		

Table V. R&D + Operational needs

What it is	Additional statement in TCM, 2000. Oct.	Last statement in TCM, 1999. Nov.
1. End point (Spent Fuel)	* Final Disposal * Reprocessing * Continued storage Note: Conditioning might be required	
2. System type (Cask, Silo, Vault)		
3. Environmental Impact Assessment (EIA)		
4. Criticality		
5. Fuel characteristics (Start, Post irradiation, Post wet storage)		
6. Can we mix store contents		
7. Retrievability	"Object to be retrieved must be defined at the design stage"	"Ability to remove the whole spent fuel assembly (as a single item) from storage for further processing or disposal after extended storage periods"
8. Handling system and transport		
9. Decommissioning issues / Activation		
10. Cost + continued funding		
11. Sabotage + Malicious acts		
12. Safeguards +Intrusion detection		

	What is happening	Additional statement in TCM, 2000.Oct.	Last statement in TCM, 1999. Nov.
13.	Monitoring requirements	* Items a. Temperatures b. Pressures c. Radiation d. Safeguards seals (Cameras) e. Full scale or features exams. f. Environmental monitoring	a. Visual inspection for degradation b. Radiation monitoring c. Neutron monitoring d. Neutron absorber e. Sealing f. Temperature
14.	Maintenance + Inspection needs program	*Inspection (Non intrusive in normal conditions) a. Housekeeping b. Record checks c. Operating procedures + compliance	

Table VI. Regulatory

	Item	Known	Partly known	Unknown
1.	Safeguards	X		
2.	Q.A.		X	
3.	Operation security	X		
4.	Documentation		X	
5.	Continuing cover + sustainability	X	X	
6.	Licence duration	X		
7.	Licence renewal		X	

Table VII. Container / Containment

	Item	Known	Partly known	Unknown
1.	Barriers + Confinement		X	
2.	Shielding	X		
3.	Atmosphere		X	
4.	Monitoring		X	
5.	Criticality		X	
6.	Retrievability		X	
7.	Transportable		X	

Table VIII. Contents

	Item	Known	Partly known	Unknown
1.	Basket type		X	
2.	Post irradiation condition	X		
3.	Fuel characteristics		X	
4.	Enrichment	X		
5.	Intact or failed	X		
6.	UO_2, Th, MOX, Metal, Other	X		
7.	Irradiation history		X	
8.	Post-wet storage condition		X	

1.4. WET FUEL STORAGE FACILITIES

1.4.1. Storage conditions

One consensus opinion from the TCM is, that material characteristics and properties will vary with time. This may be a result of general corrosion. A second point is, that the demands required for ten-year storage periods are different from those required for 100 years. For example the following parameters will change with time:

(1) Fuel decay heat,
(2) Total activity,
(3) Degradation of materials,
(4) The conditions that these materials experience over time (e.g. thermal, environmental, radiation).

Several questions were raised with regard to wet storage of spent fuel. First, considering the role played by the pool water, as coolant and biological shield, does there come a point in time when it is no longer required as the required duty, as thermal conditions and dose have changed dramatically from the starting position? The answer, in principle yes, if doing so does not impact on fuel assembly integrity, the residual radiation and airborne activities etc. can be managed.

Second, can the new duty be performed by another material, for example, a glass or a polymeric cover? Possibly. Although the above questions appear to be radical, there are examples, where time is used to downgrade the storage conditions/requirements. For example the management of Magnox fuel at Wylfa NPP has changed over the years where initial fuel cooling is undertaken in carbon dioxide cooled stores, and subsequent storage is in air.

Storage conditions are defined to ensure that safety margins are not compromised in time. Secondly, the conditions should also be defined to ensure best practice.

1.4.1.1. Water chemistry

Water chemistry requirements go hand in hand with those required to ensure that fuel assembly integrity is not compromised over the storage period. The need to ensure that aggressive ions, such as chloride, are removed or kept to an absolute minimum is paramount, in order to maximize material safety margins and to ensure premature material failure is avoided. The only potential problem will be, if the requirements to maintain fuel assembly integrity have a detrimental effect on the fuel storage components.

With a move to much longer storage duration, the impact on fuel storage facility materials also needs to be taken into consideration. This may lead to the introduction of corrosion inhibitors, to minimise the rate of general corrosion.

Concerns must be re-examined. For example, are welds a weak point, where pools are lined with stainless steel? Is water ingress between liner and primary structure a problem? It depends upon original weld standard and techniques. There is experience of such failures and repair techniques have been developed. The main issue is the need for spare storage capacity, for clearing some space, to enable the repair to be undertaken. It is believed that water ingress should be avoided, as this is a potential concrete degradation mechanism.

The particular fuel being stored defines the water chemistry required. For example the typical water chemistry for LWR at AFR pools is:

pH	4–6
Conductivity	100–350 µS/m
Chloride	<0.1 ppm
Sulphate	<0.1 ppm
Fluoride	<0.1 ppm
Na^{+} and Ca^{++}	<0.5 ppm

1.4.1.2. Temperature

Environmental factors have to be considered. A question to countries that experience extremes in temperature is the impact of thermal cycling (repeated freeze-thaw cycles) on the concrete structure (especially the external surfaces). Does this define a requirement to protect against this function? The consensus opinion from global experience is, that it is an issue that needs to be closely monitored, and measures taken to minimise the effect.

Decay heat natural cooling requirements become less onerous with time. Depending upon the type of fuel and storage duration cooling may no longer be required. In fact the reverse is true, the need to heat the water may become applicable.

One of the prime functions of minimising pool temperature is the impact on activity leach rate. Minimising the activity release during storage will impact on the lifetime cost of plant cleanup operations (volume of ion exchange resin used), and the cost of final pool decommissioning.

1.4.1.3. Activity levels

See also comments given above.

The long term management is the issue. It can be concluded that in order to avoid problems associated with activity buildup the fuel should be either canned or contained (to introduce a secondary barrier between pool water and fuel) or have a cleaning system.
The choice would be determined through a lifetime cost/benefit analysis.

1.4.1.4. Corrosion products

Controlling corrosion products can be important to avoid secondary electrochemical reactions between the fuel and the storage rack or container. This relies on good housekeeping policy. Uncontrolled release of corrosion products will impact on final pool decommissioning cost, and potentially lead to early failure of the fuel, or affect its retrievability.

The general conclusion in moving to longer storage periods invokes an extension to current best practices, but there needs to be some consideration to the effects of fuel decay heat, total activity, material degradation and the conditions that these materials experience over time (e.g. thermal, environmental, radiation). Effective management of water chemistry, temperature, activity levels and corrosion products are required, to ensure that safety is not compromised, the fuel can be retrieved, to minimise lifetime operating costs, and final decommissioning cost/requirements.

1.4.2. Long term behaviour of materials with safety functions

1.4.2.1. Loss of pool water

Although covered here, the loss of pool water does not affect the safety of the fuel within the pool, in reality the criticality safety case is improved. It does, however, present other problems, e.g. loss of containment, radiological, etc.

1.4.2.2. Storage racks, etc.

The robustness of most neutron absorbing materials has been questioned at some point in time. Examples include swelling of Boraflex and Boral and the durability of borated stainless steel under impact conditions.

The life limiting factors include general corrosion, changes in environmental conditions, and radiation damage. These issues can be addressed by research and monitoring programmes.

1.4.2.3. Fuel cladding defects

It is important to minimise the effects of cladding defects:

- For dose reasons, impact on operators integrated over time,
- Impact on maintenance activities,
- Activity buildup in cooling systems,
- Cost of remediation,
- Impact on final decommissioning as activity will leach into cracks, pool coatings etc.,
- Ion exchange usage.

Resolution is affected by restoring the primary containment barrier. Identification that a defect has occurred during prolonged pool storage can be determined through activity release monitoring in conjunction with activity release models.

1.4.3. Research & development and operational needs

R & D activities should be geared to being able to demonstrate safe storage for the projected storage duration and at any point in time.

Investigations to support projected new life may be required.

1.4.3.1. Development of detection systems

Analysis of material samples should be developed.

1.4.3.2. Monitoring techniques

Multiple monitoring techniques involve a mixture of traditional techniques and on-line state of art monitors. NDT monitors linked to the facility control room. Ideally they will alarm, when there is a change in conditions, not when a failure is underway.

1.4.3.3. Operational needs

Some consideration should be given to the issue to plan up front the lifetime cost and to make financial provision (managed fund) as per decommissioning activities. The way the facility is operated will impact on lifetime cost. Examples of factors affected include:

- Minimising activity levels within the storage pool,
- Minimising degradation of materials,
- The operation of auxiliary plant,
- Management of water leakage,
- Temperature, dose levels, physical protection.

1.4.3.4. People

- Continuity of expertise,
- How they respond to an event,
- Operator behaviour as a function of time (behavioural safety),
- Quality assurance programme also implies more reliance on automatic alarms. Regular inspections by regulators unannounced to ensure standards are being maintained.

1.4.3.5. Documentation

- Information for decommissioning,
- Records (how it was build, activity levels, the fuel being stored, etc.),
- The systems used to store the information,
- Plant items, continuity of plant spares.

1.4.3.6. Non-proliferation aspects

This may be easier for wet storage as it is feasible to inspect the fuel in situ at any point in time by examination or on-line monitoring (cameras).

1.4.3.7. Licence extension

The requirements for licence extension (or periodic safety review) are different from the previous, because there is always change, for example building codes. The impact of these changes needs to be evaluated and assessed against the ability to safely store the fuel. If there are no safety implications then modifications to meet the last standard and should only be carried out if cost effective to do so. Where safety is compromised modifications will be necessary, however, this may lead to facility closure on the grounds that the modifications can not be carried out, or the costs are too restrictive.

1.4.4. Conclusions

There are no foreseen issues presented by increasing the operating life of wet storage facilities, provided that safe operation can be demonstrated through continued monitoring and maintenance of quality assurance systems.

1.4.5. Recommendations

- There is need for spare capacity or contingencies for unforeseen events,
- Need to have a monitoring programme for key plant areas to demonstrate safety,
- Provision for funding to cover plant life needs to be considered,
- Further assessment of the technical issues outlined above should be considered.

1.5. REGULATORY CONCERNS RELATED TO THE LONG TERM STORAGE OF SPENT FUEL

Member States have similar regulatory objectives regarding the management of spent nuclear fuel. Those objectives are to protect public health and safety, by implementing regulations to:

- Maintain subcriticality of spent fuel,
- Prevent the release of radioactive material,
- Ensure that radiation rates and doses do not exceed acceptable limits, and
- Maintain retrievability of the spent fuel throughout the lifetime of the storage facility.

Table IX shows the licensing periods among the member states, however, 40–to–50 years appears to be a general target for the licensing of storage systems and facilities based on their design life. However, additional research on the behaviour of spent fuel and storage systems is needed to establish a technical basis for global support of licensing periods that match the design life of equipment and facilities. A number of unforeseen factors, e.g., emerging technologies and socio-environmental changes, could impact on regulatory requirements over this time period. Records management will also be a key regulatory concern.

Potential environmental changes such as, insulation, erosion, and rising ground water tables could result in unscheduled reviews of spent fuel storage facility licences. An increase in population could also result in the need to evaluate collective doses to the population from storage facilities and may potentially affect licences.

The development of new technologies will change the perception regarding the safety of storage systems. New information on shielding or gasket materials may affect periodic maintenance requirements. Results from risk studies may show which aspect of design or manufacture requires the most attention. This is likely to be a benefit to regulators, as resources are limited, and must be directed at the most significant safety related issues.

Table IX. Current/Planned AFR spent fuel management practices

Country	Type of Storage	Design Life (Years)	Licence Duration (Years)	Disposal Option	Comments
Belgium	Wet/Dry				
Bulgaria	Wet/Dry (planned)	30	10	Not decided	Temporary licence
Canada	Wet	40	2 (linked to reactor licence)	2035	
	Dry	50	Under evaluation		
Peoples Republic of China	Wet	30 (facilities)	30 (facilities)		Reprocessing
Czech Republic	Dual-Purpose Casks	40 (cask) 50 (facility)	10 20 (under evaluation)	2066	
France				Not decided	Reprocessing
Germany	Dual-Purpose Casks	40 (cask) 70 (applied for the facility)	40	2030	
India	Dry				Reprocessing
Italy	Dual-Purpose Casks	50	Not yet issued- 20 (planned)	Yes - Date not decided	
Japan	Metal Casks	40	Not Issued	No	Reprocessing
Korea	Concrete Silo	50	Linked to Reactor Licence	Not decided	
Lithuania	Dual-Purpose Casks	50	20	Not decided	
Netherlands	Dry	100	Not limited		
Romania	Wet/Dry	30	30	Not decided	
Russia	Wet	30	30		
Slovak Republic	Wet	50	General licence Not issued	Yes 2040	Reprocessing Not excluded
Switzerland	Dual-Purpose Casks	40	Not limited	Yes - Date undecided	Reprocessing Not excluded
Sweden	Wet	60	Not limited	Yes - 2010	
Ukraine	Concrete Casks	50 50-100	Not issued Not issued	Not decided	
United Kingdom	Wet Modular Vault	Varies	Not limited	No	Reprocessing
United States	Dry Wet	40 Generally linked to NPP licence	20 20	Yes - 2007	Geologic disposal

One of the more significant issues for long term storage will be the management of regulatory and storage operation records. It is vital that records be retained and retrievable. Technology for records management has changed markedly in the last 20 years and will likely change even more in the next 50–100 years. Reliable and diverse means must be used and records must be maintained at geographically separate locations. Because current electronic means of data storage may not be reliable for extended periods, it may be necessary to transfer data from one system to another as technology improves. Shortages in the availability of people trained in nuclear technology over the next 50–100 years may require maintenance of very basic information, such as definition of terms and principles of design, in addition to the details of design.

The long term behaviour of safety critical storage cask components is not well known. Examples include sealing of bolted lids, organic neutron shielding and fuel cladding. This lack of knowledge will lead to regulatory requirements for monitoring, analysis or replacement to assure continued safety of the public. As experience is gained, there may be a relaxation of requirements commensurate with safety.

As new materials are developed and used, it will be necessary to determine their behaviour under long term storage conditions. For new cladding materials, it might be appropriate to initiate a lead test assembly programme. Another major consideration for long term storage is retrievability. The definition of retrievability may well depend upon the intended end use (disposal in repository or reprocessing) and the level of residual mechanical handling properties required. A simplified approach would be to define retrievability as “the ability to remove the spent fuel assembly as a single intact item from storage for further processing or disposal after extended storage period”. However, it may be appropriate to consider retrievability in terms of integrity of storage canister or basket containing the fuel.

A special situation will exist at former nuclear plants, that will be decommissioned after the fuel has been loaded into the casks for interim dry storage at away-from-reactor sites. No pool will be available in the event of problems, e.g. leakage in a cask. A possible regulatory requirement will be provision for cask maintenance if, at the time of the cask loading, no firm date exists regarding the availability of a centralized national storage facility for radioactive waste (and temporary storage of fuel). Potential solutions include: a double lid — double gasket that increases the flexibility of cask repairs without the need of a hot cell or pool, or a dry transfer system similar to that recently developed in the United States.

1.6. SUMMARY OF WORKING GROUP DISCUSSIONS

The group discussions were held under the perspective of extensions to existing storage periods both UO_2 and MOX fuel, reaching higher burnup by means of advanced materials.

Up to now, the nuclear industry worldwide has accumulated significant fuel storage operating experience over the past 50 years. This experience is mainly based on wet storage systems, which have been found to be safe and effective. Wet fuel storage is now considered to be a mature technology. In comparison, dry storage is an evolving technology, which has been developed over the past 20 years. Under present boundary conditions, dry storage can also be regarded as an established technology. Unlike wet storage, dry storage can be more sensitive to fuel design changes and burnup increase, because of higher storage temperatures, which give rise to thermally activated processes.

The results of the group discussions can be summarised as follows:

(1) In wet storage there exist no urgent questions to be solved with regard to increasing operating life times. However, some recommendations, e.g. in the area of monitoring or technical optimisation were made,

(2) In dry storage, there also exists a certain amount of supporting technical data, covering the burnup of the fuel loaded and the performance of the systems to date. For high burnup and MOX fuel, an extension of the knowledge on the creep behaviour of future cladding materials is needed. Additionally, a surveillance programme could demonstrate the long time behaviour of cask and fuel. For the development of advanced dry storage systems further R&D activities are needed, such as system performance for the perceived duty,

(3) The regulatory objectives are very similar for all member states. Regulatory concerns were identified in a number of areas. Examples include:
    - How to handle technology changes,
    - Extrapolation of material behaviour or performance for increasing storage duration.

2. OVERVIEW OF SPENT FUEL STORAGE SITUATION

2.1. INTRODUCTION

Long term management of spent fuel has become of increasing concern to many Member States, since few decisions are yet available with regard to the implementation of final disposal. The reasons are multiple and country-dependent, but overall there has been no strong public support in favour of the final disposal of spent fuel. Additional studies are being conducted to better address the long term performance of spent fuel and materials.

While awaiting disposal decisions and to address increasing at reactor (AR) storage needs, the nuclear industry began with re-racking of spent fuel pools. However, in most cases because of economic considerations, the industry is constructing additional dry storage capacity either at or away from the reactor site. At the same time, it is looking at how a much longer operating lifetime for existing and future storage facilities can be implemented, which entails technological and R&D issues, that must be addressed co-operatively by Member States.

2.2. STATUS OF SPENT FUEL ARISING

The data presented, current and projected spent fuel inventories, is based on the Agency's own statistics and forecasts for nuclear power generation. This data will be presented in terms of economic regions and on a worldwide basis.

2.2.1. Status of nuclear power

Currently the growth of nuclear power is at a standstill in Western Europe and North America, while expanding in parts of Asia and Eastern Europe. At the end of last year, 433 nuclear reactors were operating in 31 countries. They provided about 17 per cent of the global electricity supply. The total operating nuclear capacity was 349 GW_e. There are 37 reactors under construction, which will provide an additional 31 GW_e of capacity.

2.2.2. Global spent fuel arising

The total amount of spent fuel generated by the end of 2000 is expected to be close to 230,000 t HM (worldwide). The annual discharge rate is about 10,500 t HM.

Projections (Figure 2) indicate, that the amount generated by the year 2010 may be close to 340,000 t HM.

By 2020, the date by which most of the presently operated nuclear power reactors will be close to the end of their design operation life, the total quantity of spent fuel generated is expected to be of the order of 445,000 t HM.

Based on the assumption that the global reprocessing capacity is 4,000 tonnes per year, we can state that the current spent fuel inventory is around 35 times the reprocessing capacity. Alternatively, it would take 35 years to reprocess all the spent fuel that has been stored to date.

2.2.3. Spent fuel storage in world regions

On a regional basis, the picture for current and projected spent fuel arising is given in Figure 3.

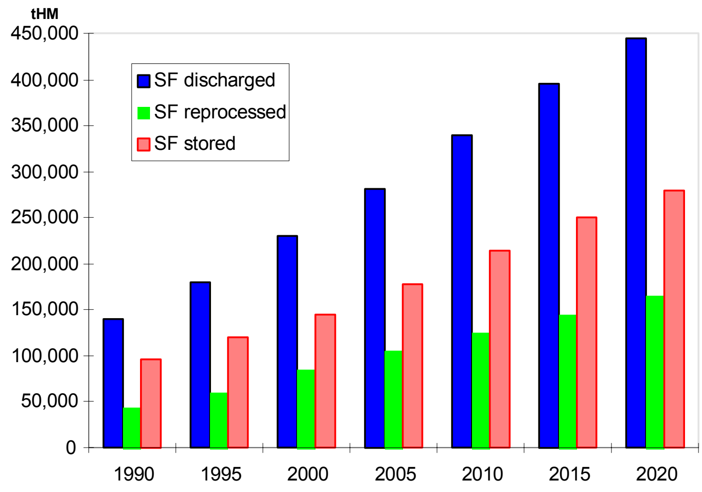


FIG. 2. Spent fuel worldwide discharged, reprocessed and stored.

By the end of 2000, the amount of the spent fuel that we expect to be in storage is:

- West Europe: about 32.5 kt HM,
- East Europe: about 22.8 kt HM,
- North and South America: about 72 kt HM,
- Asia and Africa: about 17.7 kt HM,

The trends for the coming decades are:

- Western Europe storage inventories are predicted to remain around current levels, primarily as a result of reprocessing activities,
- Asia, Africa and Eastern Europe spent fuel volumes are predicted to double by 2010, this is at a rate greater than current discharge levels,
- In America spent fuel inventories are predicted to increase in-line with current discharge rates.

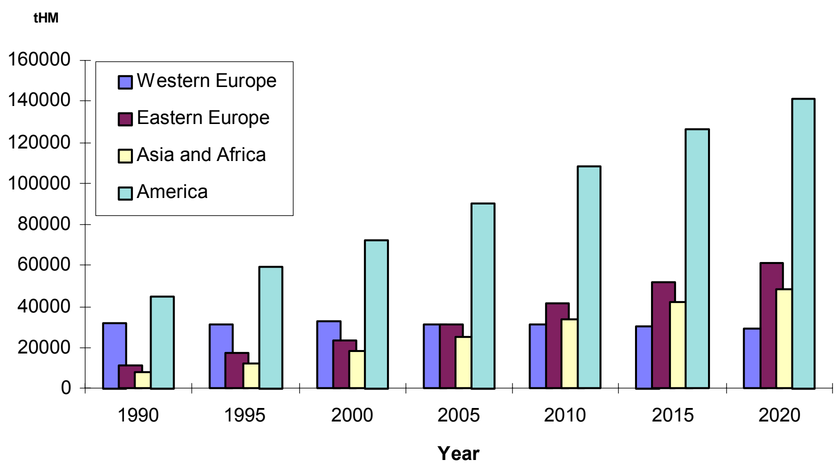


FIG. 3. Spent fuel stored by world regions.

2.2.4. Spent fuel storage according to storage type

Table X provides a summary of wet and dry storage capacity in Member States. The current global world storage capacity is slightly more than 239,000 t HM, and thus exceeds, by about 94,000 tonnes, the capacity needed by the end of 2000.

Table X. Spent fuel storage capacity by region

Region	NPP pool storage capacity, t HM	AFR wet storage capacity, t HM	AFR dry storage capacity, t HM	Total storage capacity t HM
Western Europe	28,265	32,270	10,416	70,951
Eastern Europe	11,913	20,788	1,471	34,172
America	94,662	1,712	6,342	102,716
Asia & Africa	27,924	1,725	1,737	31,386
Total	162,764	56,495	19,966	239,225

In countries where storage capacity at nuclear power plants is becoming limited, one or more of the following options are being exercised: re-racking of the storage pools with high density racks, implementation of burnup credit, commissioning of AFR storage facilities.

Globally all types of storage facilities have excess capacity available. Figure 4 compares the capacities of the various storage types with their current inventories.

Table XI summarises the situation with respect to the construction of new storage facilities. Total capacity by region ranges from 1.5 kt HM in Asia to 6.8 kt HM in America. The growth in storage capacity, some 13.9 kt HM, is in equilibrium with projected spent fuel arising. The Table also indicates the increasing preference for away form reactor dry storage systems.

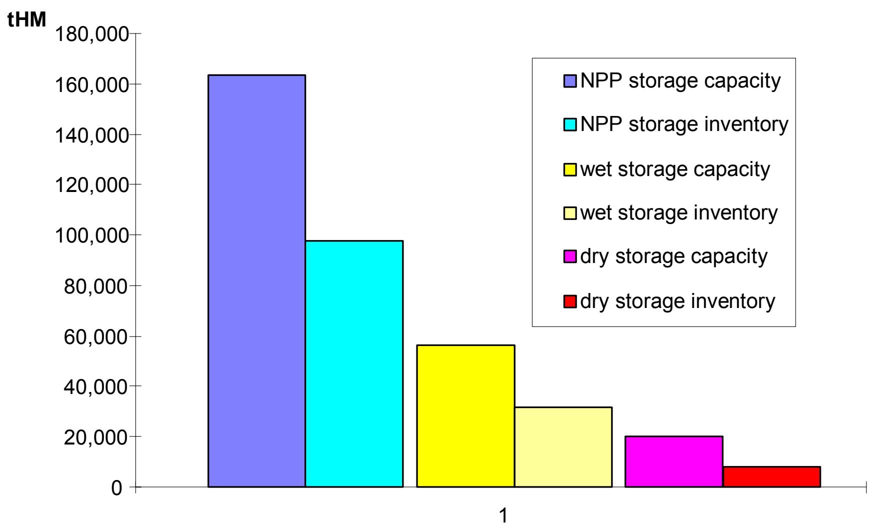


FIG. 4. Comparison of capacities and inventories of different types of spent fuel storage.

Table XI. Spent fuel storage capacity under construction

Region	Under construction AFR wet storage, t HM	AFR dry storage, t HM	Total capacity, t HM
Western Europe	3,000	1,000	3,000
Eastern Europe		1,600	1,600
North America		6,800	6,800
Asia	700	800	1,500
Total	3,700	10,200	13,900

2.2.5. Balance between spent fuel arising and storage capacity

It is estimated that on a global scale the spent fuel arising will fill the existing storage facilities and those under construction by around the 2015 if additional facilities are not provided. However, it is unrealistic that no new facilities will be provided on such a timescale. Figure 5 shows the predicted storage capacity by world regions.

Figure 5 shows the predicted storage capacities by world regions.

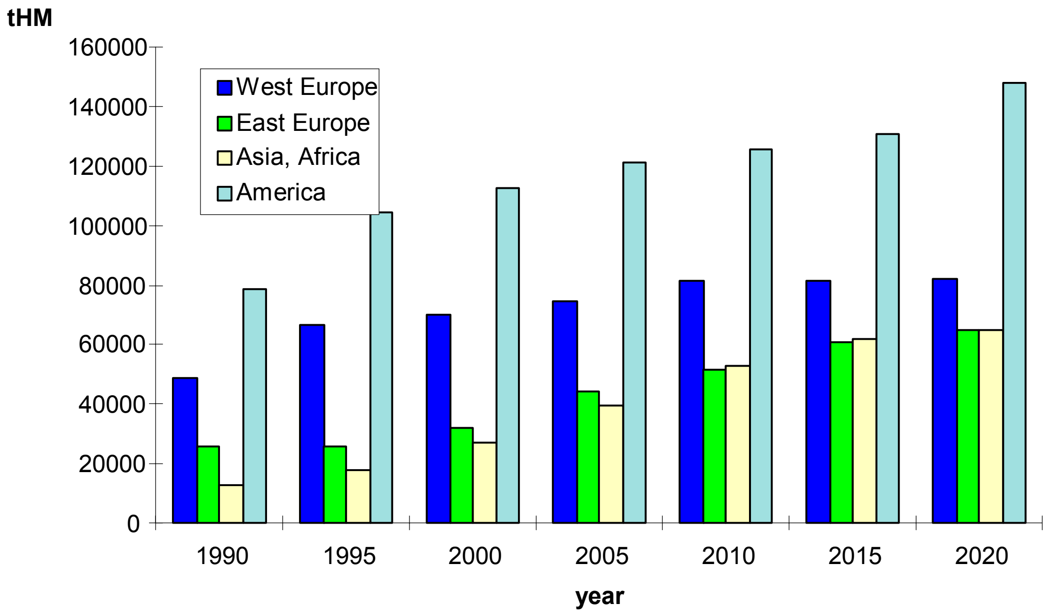


FIG. 5. Predicted storage capacities by world regions.

At a national level the situation differs from one country to the next and often from one utility to the other. In some cases, the storage pools are already full, occupied by spent fuel, allowing emergency core unloading only by special measures. Hence, additional storage capacity has to be installed in a timely fashion to restore normal operating capabilities. In other cases, additional storage capacity has to be provided to replace wet storage facilities, which cannot be refurbished, or continued safe operation is in question. In particular in some countries of Eastern Europe, plant operation might be jeopardized if additional storage capacity cannot be installed in time.

The preferred options are to extend the life of existing storage facilities (if their safety can be assured), then to adopt the dry storage concept where a new facility is required. In the later case, preference has been towards cask storage because of the shorter time to build, pay as you store, and building on previous licensing experiences.

Any further delay to decisions on the development of final repositories or new reprocessing plants' operating licences will also lead to a higher demand for spent fuel storage capacity. The storage time will be much longer than originally anticipated. In particular the "wait and see" policy leads to storage periods of undetermined duration. Proper engineering of such storage facilities, the corresponding selection of construction materials and operating processes will have to achieve the safety performance required, throughout the storage periods envisaged.

2.3. SPENT FUEL STORAGE SITUATION IN DIFFERENT MEMBER STATES

The following information is based on the papers presented during the 1999 and 2000 TCMs:

Belgium has sent more than 600 t HM of spent fuel for reprocessing. According to a recent decision, reprocessing contracts will be terminated and unreprocessed fuel will be stored pending a decision with respect to disposal in a final repository. To accommodate current and future fuel arising Belgium has designed and licenced a storage facilities equivalent to 40 years operation of all NPPs. The Tihange NPP operates an AFR wet storage facility with a pool capacity for 3700 fuel assemblies. The dry storage facility at DOEL can accommodate 60 casks, each cask can accommodate up to 28 fuel assemblies. An extension to the current facilities at DOEL has been authorized.

Up to 1988, **Bulgaria** had shipped over 3000 spent fuel assemblies back to Russia under the intergovernmental agreement. Due to an increase in storage period from 3 to 5 years prior to shipping for reprocessing, an AFR wet storage facility has been constructed on the site of the Kozloduy NPP with a capacity for 600 t HM. In 1999, Bulgaria returned 480 spent WWER-440 fuel assemblies to Russia. A new contract is to be negotiated for the return of WWER-1000 fuel. The AFR at Kozloduy was recently upgraded to meet current seismic standards. To increase storage capacity, it is planned to rerack the reactor pools and to investigate the use of similar storage baskets to those used at Bohunice AFR for Kozloduy AFR. These measures could potentially double storage capacity. However, the current seismic standards have to be considered. The decision to develop a dry storage facility was postponed but will be reconsidered.

In 1996, **Canada** announced a policy document out-lining a set of principles governing the arrangements for disposal of spent fuel and radioactive waste. In May 2000 the Atomic Energy Control Act was replaced by the new Nuclear Safety and Control Act, which provides the framework under which the Canadian Nuclear Safety Commission issues licences. When in-pool storage capacity is approached, the older fuel is transferred to dry storage facilities. There are three types of dry storage system currently in-use in Canada: Concrete canisters, Dry storage containers (DCS), Modular vault storage systems. The dry AFR storage facilities are located at the decommissioned stations of Chalk River, Gentilly 1, Whiteshell, and Douglas Point (465 t HM in storage) and operational sites of Point Lepreu (concrete silos with a capacity of 722 t HM), Gentilly 2 (CANSTOR with a 953 t HM capacity), and Pickering (Dry Storage Containers (DSC) with a 693 t HM capacity). Bruce is to adopt the DSC technology, which should be operational in 2002. Applications for new/additional facilities have also been made for at Pickering (stage II), Bruce, and Darlington sites.

Czech Republic sees underground disposal in a repository as the ultimate solution for the back end of the fuel cycle. Reprocessing in the future and use of fissile material for power generation, however, has not as yet been abandoned. Currently, the utility will exercise its option to dry store the fuel for up to 100 years pending future developments in the industry. Handing over of spent fuel to the governmental waste agency is not expected before 2065. The Czech Republic operates an AFR dry cask storage facility employing CASTOR-WWER-440 casks. There are 60 cask storage positions, which is equivalent to around 600 t HM. An extension to this facility to accommodate another 1,340 t HM is planned (2006). A new storage hall will be built parallel to the existing one. In the case of Temelin, a dry storage facility (to be operational by 2014) with a design capacity for 1,370 t HM is planned. The total amount of spent fuel generated in the Czech Republic is expected to be around 3,300 t HM.

France implements the reprocessing policy for UO_2 fuel. The law passed in 1991 sets the frame for R&D work exploring partitioning and transmutation, geological disposal, waste conditioning processes and long term interim storage. In 2006, the country will review the entire situation and new decisions might be made in light of the achieved R&D results. France has a wet AFR storage capacity (14,400 t HM) at the La Hague reprocessing plant used for spent fuel awaiting reprocessing. Cogema has applied for a licence to extend this storage capacity to 18,000 t HM. A public inquiry was held in spring 2000, but no decision has been made as yet.

A dry vault storage “Cascade” is operated at Cadarache. It is used for research reactor fuel.

In **Germany,** part of the fuel are covered by reprocessing contracts with Cogema and BNFL, and the remainder will be stored prior to final disposal in a deep geological repository. Germany operates four AFR dry storage facilities, using mainly CASTOR casks, and two AFR wet storage facilities. The Ahaus dry storage facility is licenced for 370 LWR and 305 THTR casks. 6 CASTOR V LWR-casks and 305 THTR-casks are currently stored. The Gorleben dry storage facility is licenced for 420 LWR and HLW casks. 5 LWR-casks and 3 HLW-casks are currently stored. At Greifswald the dry storage facility for WWER fuel is commissioned and licenced. Loading operations are due to start in 2001 with the WWER fuel currently stored in the Greifswald wet storage facility. The dry storage facility at Juelich, licenced for 158 casks, has 154 CASTOR-AVR-casks containing pebble bed test reactor fuel (operated at the Juelich research center) in store. Obrigheim NPP commissioned an AFR wet storage facility for 530 fuel assemblies, which can be extended to 980 fuel assemblies, if necessary.

Due to the new policy in Germany to avoid transportation of spent fuel, licence applications were filed for 13 AFR dry storage facilities at the NPP sites, using dual purpose casks. One licence application was filed at the end of 1998, 9 were filed in November and December 1999, and three in February 2000. In four cases the licence application includes temporary storage. These temporary storage facilities are foreseen for about 24 casks each, and are aimed at bridging the gap between existing NPP storage capacity and licensing and building of the new dry storage facilities. The temporary storage will use the same CASTOR casks, that will be used in the future. The difference is, the casks will be placed horizontally on a pad and covered by a concrete hood.

The NPP Emsland at Lingen, Lower Saxony, was the first to file the licence application and the first to receive a licence. Construction started in October 2000. This facility will have 100 positions for storage casks. The concept is similar to the Ahaus and Gorleben storage facilities. Neckarwestheim has, due to local conditions, a different concept with two tunnels below surface.

Hungary has not yet decided on the final policy for their spent fuel management. An AFR modular vault dry storage (MVDS) facility was built at the Paks NPP site and the first three vaults went into operation in 1997. Vaults 4 to 7 became operational in 2000, and construction on vaults 8 to 11 has also started. The facility is licenced for 50 years. An extension to this licence might become necessary. Hungary is currently considering alternative dry storage technologies for the future.

India has limited Uranium resources and therefore has adopted the closed fuel cycle approach for optimising Uranium and Thorium resources. Spent fuel arising is increasing, due to the planned growth of nuclear power generation in India. The reprocessing capacity is also being augmented, to reduce the spent fuel storage load. However, long term storage AFR facilities' storage pools and dry cask storage are being pursued for interim storage.

Existing AFR facilities for BWR fuel include a wet storage facility at Tarapur for up to 2000 fuel assemblies, and modular dry cask storage facility employing casks holding up to 37 fuel assemblies, four casks have been loaded to date. In the case of PHWR fuels, the Canadian designed DSC system is being employed for ten year cooled fuel, 28 casks have been loaded to date. The annual spent fuel arising is around 230 t HM.

Italy shut down all four nuclear power plants in 1987, following a public referendum brought on as a result of the Chernobyl incident. All the spent fuel from the gas cooled reactor and some from the LWR reactors have been sent to BNFL for reprocessing. After a strategy review, it has been decided to transfer the remaining 233 t HM of spent fuel into dry storage whilst a final solution is found. Dry cask storage technology is currently being investigated.

Lithuania operates an AFR dry storage facility, based on the CASTOR-RBMK cask, at the Ignalina site. The facility can accommodate up to 72 casks, equivalent to a total operating capacity of 410 t HM. In September 2000, 20 CASTOR casks (116 t HM) were loaded. Ignalina will now switch to the CONSTOR-RBMK cask, which is currently being licenced. The cold tests with this cask have already been performed. The hot trial tests are foreseen for spring 2001, followed by regular loading. 15 CONSTOR casks are already available at Ignalina.

Japan has an extensive nuclear programme with about 45 GW_e installed capacity and more than 13,000 t HM of spent fuel discharged as of 1998. Almost 6,000 t HM have been sent for reprocessing abroad and almost 1,000 t HM has been reprocessed at the Tokai reprocessing plant in Japan. The remainder is mostly stored in pools at the reactor sites. The Japanese policy is based on reprocessing and utilisation of the recovered plutonium and uranium. In 2000, the long term programme was revised, to include AFR storage. Japan currently operates AFR wet storage facilities at Fukushima NPP, Tokai reprocessing plant and Rokkasho (1,500 t HM). At the present time there is only the one operational dry AFR storage facility, at Fukushima NPP, but there are plans to provide additional facilities, to offer greater flexibility, by 2010.

Republic of Korea does not plan to reprocess their spent fuel for the time being. Dry storage for 100 years is regarded as an attractive option. A decision to choose either open or closed fuel cycle will be taken later. Various measures have been taken to increase the at reactor storage capacity by reracking with high-density racks. The share of nuclear energy in Korea is about 43% with 16 operating units. Four more units are under construction and another 10 are planned. Korea has two AFR Candu dry concrete silo storage facilities in operation at the Wolsong site, total capacity 3,078 t HM. The at reactor wet storage capacity at the Kori, Yongwang, and Ulcin NPPs were increased by using high-density spent fuel racks. Korea is considering a central wet storage facility to be operational in 2006.

Mexico has adopted a "wait and see" policy while awaiting guidance on future decision to be made regarding SF disposal. The spent fuel is stored in the reactor pools of the two Laguna Verde units.

Netherlands has committed all its spent power reactor fuel for reprocessing at Cogema (Borssele) and BNFL (Dodewaard). The resulting reprocessing waste, the residues from research reactor fuel and other radioactive waste will be stored for a minimum of 100 years in the HABOG facility, which is currently under construction.

Romania is considering deep geological disposal, as the final disposal option. The criteria to evaluate the suitability of four different hosting rock-types (salt, granite, volcanic tuff and green schist) has been developed. In-depth studies have been performed on salt structures, as they appear to be the best candidates. Romania is awaiting and monitoring international developments in this area. Meanwhile Romania plans to build an AFR dry spent fuel storage facility, capacity for 6000 t HM, at their Cernavoda NPP site. The tendering process was initiated in late 2000 and 5 tenders were received. The lead-time for selection and delivery of the facility is very short, as additional storage capacity is required by April 2003.

The **Russian Federation** plans to reprocess WWER spent fuel, whilst for RBMK fuel the open fuel cycle is foreseen. Russia operates AFR wet storage facilities for RBMK fuel at Kursk, Leningrad, and Smolensk NPPs (total capacity 15,972 t HM), and for WWER fuel at Novo-Voronezh NPP (400 t HM capacity), Mayak reprocessing plant (560 t HM capacity) and Krasnoyarsk (6,000 t HM capacity). It is planned to increase the capacity at Krasnoyarsk to 9,000 t HM by using new storage baskets that can accommodate up to 16 WWER-1000 fuel assemblies (currently 12). A reprocessing plant is planned for Krasnoyarsk, but has been postponed for the time being. The Mayak storage capacity will be extended, but only for submarine fuel.

Russia is considering a dry multi-vault storage for submarine and icebreaker fuel in the arctic area based on the technology used at Krasnoyarsk for RBMK fuel. Storage casks have been developed for submarine fuel. The same cask type is foreseen for RBMK fuel.

Slovakia: The original Slovak spent fuel management plan was based on the assumption that all spent fuel from Russian designed WWER reactors would be transported to the Russian Federation for reprocessing or permanent disposal. Because of political changes in the Russian Federation only, 85 t HM have been returned. Currently, spent fuel is stored in at-reactor pools and at the ISFSF (wet) located at the Jaslovské Bohunice site.

The ISFS was commissioned in 1986. After the decision to store spent fuel in Slovakia, Bohunice decided to extend the capacity and lifetime of the existing ISFSF from 600 t HM to 1693 t HM (14,112 FAs) by using new high density baskets. The ÚJD SR is going to issue permission to operate the modified ISFSF for a limited period (approximately 10 years). An extension to the operating licence will be subject to positive results from the required long term monitoring programme. Mochovce NPP has in-pool storage capacity until 2006 (unit 1) and 2008 (unit 2). Consideration is being given to sending some fuel to the Bohunice AFR storage facility, but in the long run Mochovce will require an AFR storage facility of its own.

Slovenia does not intend to send its spent fuel for reprocessing. The fuel is stored in the reactor pool with a capacity of 470 fuel assemblies, equivalent to 193 t HM at Krsko NPP. In spring 2000, the Krsko operator signed a contract for reracking the spent fuel pool that should be completed by 2003. The installed capacity (1,750 locations) will be sufficient for the whole plant life. In case of plant life extension, there will be also a possibility to extend the capacity to 2,400 locations.

Spain sent all their spent fuel from the gas cooled reactor to Cogema, and some LWR fuel to BNFL for reprocessing. The present Spanish policy for the management of spent fuel and other high level wastes considers an open cycle strategy and foresees their direct disposal into deep geological formations after an adequate period of interim storage. The spent LWR fuel arising in Spain will amount to about 7,000 t HM. At reactor site, the strategy to handle that inventory has been based on reracking all reactor pools, with high-density racks using burnup credit. An AFR storage strategy is also being considered as a future goal. It is to provide extra storage by means of dual-purpose (storage and transport) casks. The construction of a dry cask storage facility for 80 casks at the Trillo site was started in early 2000. This facility should be commissioned by end of 2001. The fabrication of 2 casks has already started. Trillo is the first NPP to run out of storage space (by 2003). The ultimate goal is to provide a central storage facility by 2010. Both wet and dry systems have been studied, and the preferred design is based on a vault-type storage system. Storage for a minimum of 50 years in envisaged.

In **Switzerland,** the amount of spent LWR fuel resulting from nuclear power generation will be around 3,100 t HM, assuming that each of the 5 reactors operates for 40 years. Within the existing contracts about 1,100 t HM of spent fuel has been sent to COGEMA and BNFL for reprocessing. The option to continue reprocessing may or may not be exercised depending on the new atomic law. Should reprocessing be discontinued the remaining 2,000 t HM of spent fuel will have to be stored. During the last years, two AFR dry storage facilities were commissioned, the central storage ZZL, operated by the ZWILAG utility, and the ZWIBEZ facility operated by the NOK utility. The ZZL has 200 cask positions for spent fuel and HLW from reprocessing. The ZWIBEZ facility, used only for the fuel and HLW from the reactors at Beznau, has 48 cask positions. ZZL will receive its first casks for loading probably during 2001.

United Kingdom uses wet storage at BNFL's Sellafield site for Magnox, AGR and foreign LWR fuel. The fuel is stored in one of four large storage pools prior to reprocessing in either the Magnox or Thorp Reprocessing Plants. The majority of the Magnox NPPs are due to be closed during the current decade, with the termination of Magnox reprocessing early next decade. Life extensions to the remaining Magnox reactors are dependent upon continued safe and economic operation. Beyond the projected life of Thorp, continued operation of AGR

NPPs will be supported by wet storing AGR fuel for up to 80 years at BNFL's Sellafield site until alternative spent fuel remediation options become available. In addition to wet storage facilities at Sellafield there is an AFR modular vault dry store (MVDS) facility at the Wylfa NPP, with a capacity of 2x350 t HM. This facility differs from other MVDS design, as the fuel storage tubes are open to the air and housed on trolleys on a circular conveyor system. A new dry export route has recently been built onto the MVDS to aid the retrieval and export of the current inventory to Sellafield for reprocessing. Wylfa NPP differs from the majority of NPPs because the two reactors have no spent fuel storage pool. The Magnox fuel is transferred, after discharge, to one of three CO_2 cooled vault stores for decay heat cooling, each vault has a capacity of 86 t HM each.

Ukraine has stored all its RBMK spent fuel whilst the WWER spent fuel was returned to Russia. According to the approved "National Energy Program", interim dry spent fuel storage facilities will be commissioned for all spent fuel types. Ukraine currently has one AFR wet storage facility for the RBMK fuel in Chernobyl. An AFR VSC dry storage facility was commissioned at Zaporozhe NPP, but a licence has not yet been granted, due to outstanding issues awaiting resolution. The initial capacity for this facility is 14 VSC-24 units (~140 t HM). Commissioning of 256 NUHOMS modules is scheduled for late 2001 at the Chernobyl site. The NUHOMS modules will be built in two parallel lines of 128 modules. This capacity is sufficient for the 25,000 RBMK fuel assemblies (2,867 t HM) currently stored on the Chernobyl site. Each module contains around 11.2 t HM. It is planned to load about 2,500 fuel assemblies per year into the storage facility. Other NPPs are to adopt similar AFR dry storage facilities.

The **United States of America** stores approximately 95% of its spent fuel at reactor pools. Storage in many pools has been increased over the years with high density racks. Some utilities are using additional dry storage systems at their reactor sites. Currently more than 2,000 t HM of spent fuel are stored under dry conditions at 17 sites in 13 states. Several different systems are in use: NUHOMS, VSC, CASTOR-casks, TN-casks, NAC-casks, MC, and TranStor. The only AFR wet storage facility is operated at Morris, Illinois. There are another 18 potential sites for AFR dry storage under investigation.

Characterization of a geologic repository is underway at Yucca Mountain, Nevada. Initial spent fuel receipts could begin in 2010. A significant change in the design will enable spent fuel to be stored for potential retrievability for a period of up to 300 years. For the near term, however, a trend toward increased use of dry storage is expected to continue. By 2003, cumulative dry storage requirements will increase to an estimated 5,000 t HM at 26 sites in 22 states. In the absence of a geologic repository or centralized interim storage facility, commercial nuclear plant requirements for dry storage rise to an estimated 10,600 t HM at 47 sites by 2010.

3. TECHNICAL ISSUES

3.1. WET STORAGE FACILITIES

Wet storage facilities have been with us for 50+ years and therefore must be considered as a mature technology. The major technical challenge facing the use of wet storage facilities is their continued operation beyond perceived design life. Secondly the continued operation of the older facilities presents us with the opportunity to learn from experience in the design and operation of newer wet storage facilities. Some of this experience is also applicable to the operation of dry storage facilities.

3.1.1. Approach taken to continue operation of existing facilities (UK)

The LWR Storage Pond (BNFL Sellafield) currently stores LWR fuel that is up to 32 years cooled. Continued operation of this plant for another 25 years is not envisaged as design standards, construction materials/practices have moved on several stages, since the facility was built. Continued operation as a fuel storage facility, however, needs to be ensured in the interim period to facilitate fuel removal activities.

The approach that has been taken by BNFL in addressing continued operation is outlined below:

(1) The facility is first divided up into discrete work packages: civil structures, mechanical engineering, electrical systems,

(2) Each Package is assessed against:
- Original design (codes/standards/materials),
- The current status of each item is established — A series of reports reflecting, condition/integrity,
- Current Standards (Seismic, Electrical etc.),

(3) From the above assessments, a work programme can be generated and nominally costed against:

– Short term	improvements to achieve current standards,
– Medium term	e.g. building fabric replacement (cladding),
– Longer term	refurbishment (replacement of plant items because design duty cycles have been completed).

An Example of how the approach works in practice is given for building seismic qualification:

Issue....Seismic Qualification.....

Design Standard.... N/A.....

Comment:.... Plant predates requirement

Assessment:

Seismic Damage Assessment undertaken on the facility. Damage assessments carried-out for 0.125 g, 0.25 g and 0.35 g accelerations.

Conclusions:

No major loss of water under 0.25 g acceleration

Potential catastrophic failure of inlet pond wall in the event of building crane becoming derailed, etc.

Ideal Solution:

Replace current building crane with seismically qualified to 0.25 g.

Review of Solution:

Replacement is not feasible, because building infrastructure restricts the installation of a new crane. Secondly the cost is prohibitive.

Most Practical Alternative Solution:

Restraints are fitted to the rails to avoid crane falling on pond wall.

The process is then repeated for all issues identified.

Examples of the types of improvements recommended for the LWR Storage Pond are given in Table XII.

Table XII. Continued use improvements

Plant Item	Affected Area	Safety Issue	Upgrade
Building Crane	Rotation of the Rams Horn is manual	Manual Handling	Motorised Rams Horn
Receipt Bay	Under water camera system manual	Operability/ visibility	Remote control closed circuit camera system
Storage Bays	Storage location system markers on pond wall side	Operability/ handling	Improved location system for fuel storage containers
Skip Handler	Control system location	Operability/ ergonomics	Improved operator control system

As can be seen from the example, there are some plant items that cannot be upgraded, another example could be biological shielding (to reduce operator dose uptake). More reliance on remote systems would minimise the effect in the later case, when applying the principles of ALARA or ALARP. In other cases the upgrades become restrictive on cost grounds, therefore, a compromise position has to be taken. The decision is then taken on the application of the "As Reasonable" philosophy.

3.1.2. Storage pool optimisation (Slovakia)

After the decision to store spent fuel on the territory of Slovakia till its final disposal, the utility decided that pool storage optimization was the most effective solution for extending the capacity and life of the existing ISFSF at Jaslovské Bohunice. To facilitate this process the operator upgraded the pool cooling system, designed a new storage basket (Figure 6) to accommodate up to 48 spent FAs and proposed a new layout for the storage baskets (Figure 7), these measures have increased the total capacity of the facility to 14,112 spent FAs.

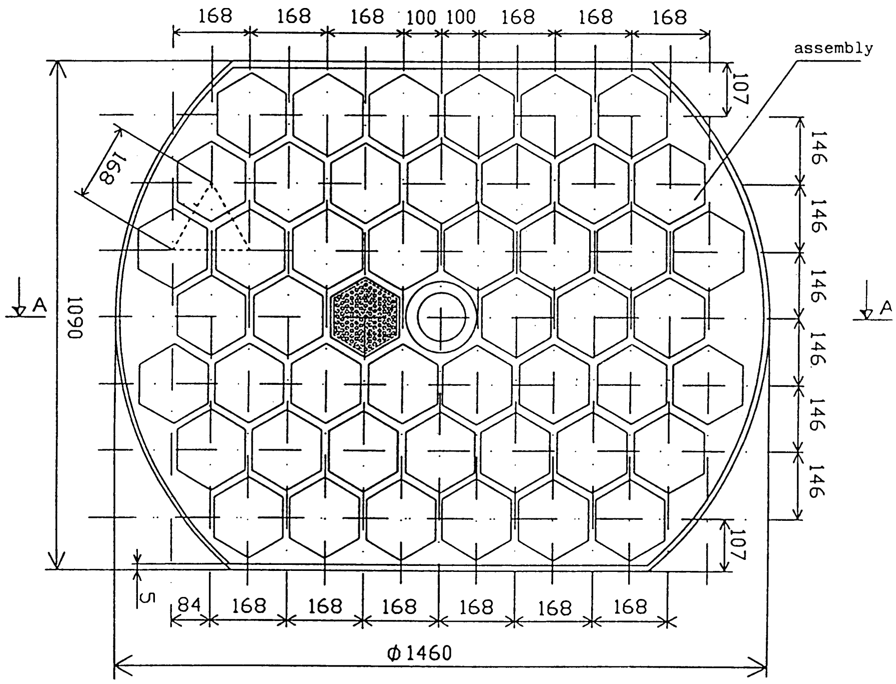


FIG. 6. New compact storage basket.

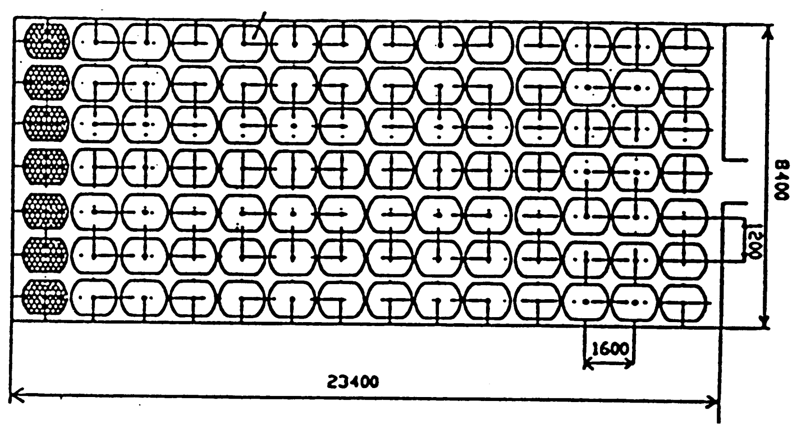


FIG. 7. Proposed new basket lay out.

Further examples of storage capacity optimisation in various Member States are summarised in Table XIII.

Table XIII. Summary of Member State storage optimisation exercises

Member State	Facility	Original Storage Capacity	Optimised Storage Capacity
Slovakia	Bohunice	600 t HM	1,693 t HM
Spain	All	6,584 t HM	14,487 t HM
Korea	Ulchin, Kori, Yonggwang	3,113 t HM	4,304 t HM

3.1.3. Learning from experience

The continued operation of existing facilities presents us with a learning curve. The main areas that have been learnt are:

(1) Ensuring good Quality Assurance Data is available covering:
- Build standards,
- Materials used,
- Retention of technical information on the material being stored, plant equipment and design drawings.

(2) Conservatively designed plant or high redundancy has proved an advantage against current standards.
(3) Doing things differently. For example:
- Roofing the storage bays,
- building with final decommissioning in mind,

(4) Condition monitoring of key plant.
(5) The ability to predict future changes in standards, etc.
(6) Provision for extension, or for post operation clean out (e.g. in terms of new export route from the facility).

3.2. DRY STORAGE CASKS

The approach being taken by the Czech Republic for addressing technical issues associated with the long term fuel storage in dry casks is typical of most countries adopting this option. The main differences between the various Member States are the perceived service duty of these casks until the availability of a final solution is derived. In the Czech Republic's case it is 70 years, Germany 40 years and so on.

In addition to this general approach some countries have specific technical requirements to meet either the regulatory requirements of the country, or the environment that the casks have to operate in.

3.2.1. Spent fuel condition during storage

It is necessary to provide sufficient evidence that the spent fuel condition will not deteriorate during storage to such an extent that any subsequent handling of the fuel would be prevented or become more difficult, or result in non-compliance with nuclear and radiation protection limits. In particular, the following criteria are assessed:

(1) The degradation mechanism of spent fuel during storage:
   - Qualitative and quantitative analyses,
   - Calculation formula.
(2) Impact of fuel cooling in the cask:
   - Cooling down profile in the cask,
   - Transient temperature profiles when cask is dried,
   - Fuel rod temperature profiles.
(3) Marginal conditions for spent fuel handling:
   - For current and future Dukovany fuel,
   - For Temelin fuel.
(4) The effect of non-compliance with marginal conditions:
   - Exceeded temperature in transients,
   - Presence of air or humidity exceeding the limit inside the cask.

In this respect, extensive work has been carried out in the **Czech Republic** focused on the development of the following models:

(1) Cooling model for spent fuel with a defined burn up, in a dry high volume storage cask,
(2) Low temperature creep model for Zr1Nb cladding, applicable up to 370°–450°C according to the stress levels,
(3) High temperature creep model for Zr1Nb cladding, applicable from 370°–450°C to 550°C and hoop strain <10%.

The fuel cladding both in normal and abnormal conditions has been systematically studied, since 1995. As part of these studies extensive experiments have been carried out which, apart from qualitative evaluation of significant factors in the fuel cladding degradation process, also allow quantitative verification of creep models. The research programmes were funded by the Nuclear Fuel Institute, Ministry of Industry and Trade, State Office for Nuclear Safety and CEZ. Apart from the Nuclear Fuel Institute, other research institutes also participated in this programme such as the Nuclear Research Institute and Skoda – Nuclear Engineering.

Figure 8 provides a comparison of the data obtained from measurement of hoop strain deformations in Zr1Nb cladding with those predicted by the creep model. As can be seen there is a very good correlation between calculated and measured values.

In the future, CEZ will extend the models to include Temelin fuel (WWER 1000) and will focus on standardisation of codes in order that they can be used for predicting the stability of spent fuel in long term storage.

General conclusions on fuel cladding behaviour during storage made in Germany:

(1) For Zircaloy-2 and 4 and their improved variants, there are no new problems recognisable with respect to increased burnup and storage times. Ongoing studies will provide improved knowledge on the details of microstructural mechanisms responsible for the creep behaviour, influences of corrosion and hydrogen uptake on the dimensional behaviour of those materials under long term storage conditions,
(2) Advanced cladding materials are of increasing commercial importance for PWR fuel. From the existing inpile experience no dramatic consequences are expected on long term storage. However, systematic studies comparable to those available or underway for Zircaloy materials are desirable and recommended,

(3) Advanced cladding materials are of increasing commercial importance for PWR fuel. From the existing inpile experience no dramatic consequences are expected on long term storage. However, systematic studies comparable to those available or underway for Zircaloy materials are desirable and recommended,
(4) Today the long term storage experience with Russian Zr1Nb material (E 110) is based on very long wet storage experience. In the future dry storage will become of increasing importance for WWER fuel and studies comparable to those available for BWR/PWR fuel will be necessary.

A more in-depth study of long term behaviour of spent fuel is part of the IAEA's Spent Fuel Performance Assessment and Research Programme (TECDOC in writing).

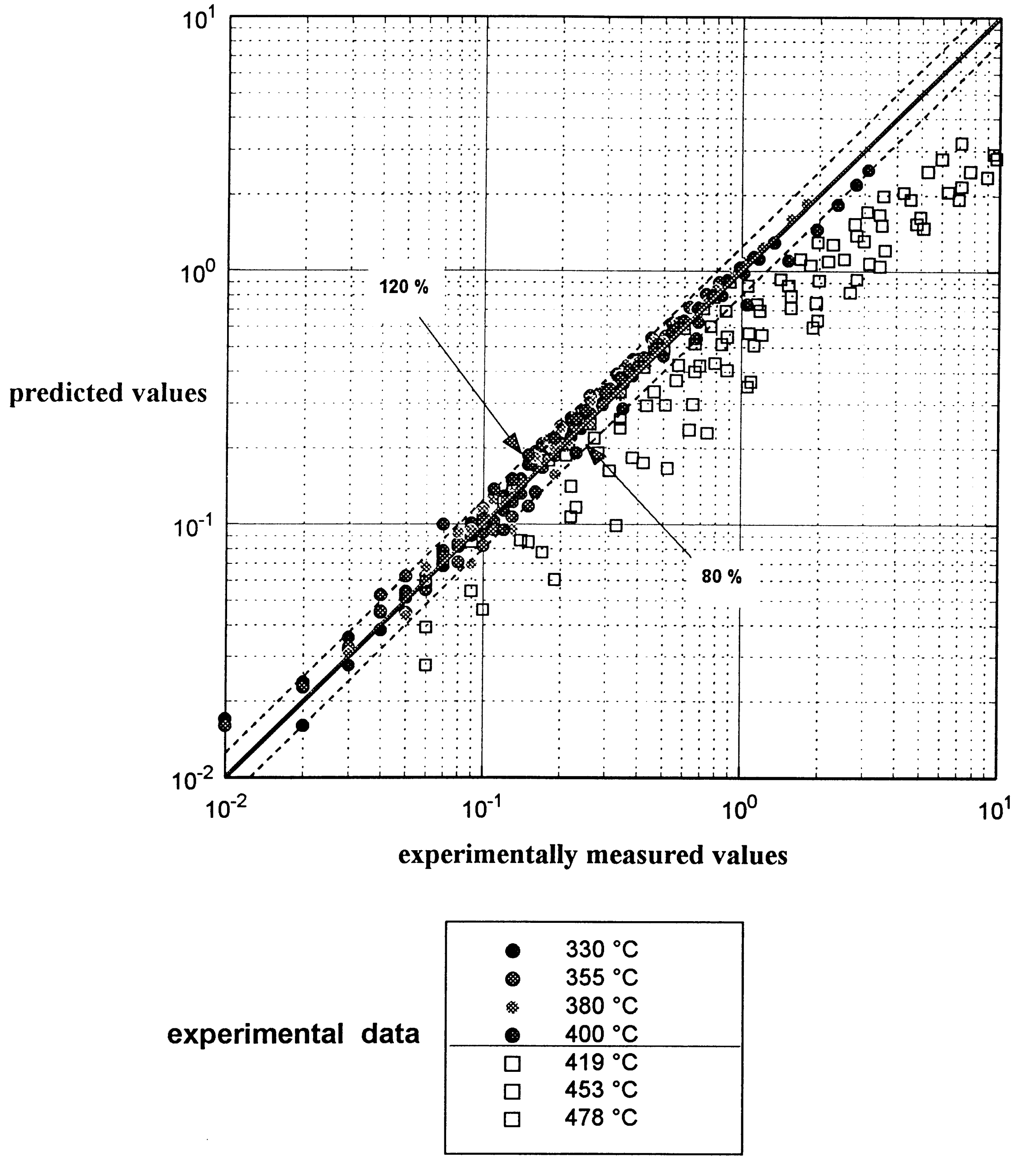


FIG. 8. Comparison of calculated and measured hoop strain in Zr1Nb cladding.

3.2.2. Condition of storage system with stored spent fuel

The ability of the storage system to meet its function during extended storage plays an important role. Unlike spent fuel, those elements of the storage systems, which would not achieve the required service life, and are not life limiting, can be replaced or substituted with improved or more advanced designed components.

The following aspects are assessed:

(1) Life time of key structural elements,
(2) Cask seals,
(3) Protective layer stability,
(4) Neutron moderator stability,
(5) Resistance of key structural materials,
(6) Life of welds, which provide leak tightness of the storage system (if applicable),
(7) Maintaining system subcriticality and shielding properties,
(8) Required sealing capacity.

The approach taken by one cask manufacturer (GNB) in addressing the issues associated with external and internal stressors was outlined.

3.2.2.1. External stressors

Mitigation against the effects of rain, snow, humidity, changes in temperature are achieved:

(1) Material choice (stainless steels) and/or applied coatings (multi-layer epoxy, Ni-plating) are sufficient for established and extended storage periods,
(2) Possible defects of coatings are detectable and repairable. Investigations have shown for non-protected DCI (ductile cast iron) in salt water that the equilibrium corrosion rate is limited to (≈ 1/10 mm / year),
(3) The high thermal capacity of the materials used in fabrication of the casks minimises the possible effects from thermal cycling,
1. Periodic inspection of casks has confirmed maintenance of as-built properties.

3.2.2.2. Internal stressors

The effects of Radiation:

(1) Stable cask material properties have been verified up to 10^{19} n/cm^2
(2) Metal seals are positioned to minimise the effects of neutrons (< 10^4 n/cm^2 · s),
(3) Suitability of elastomeric seals is limited to a maximum of 10 MRad. The seals, however, have no containment function,
(4) Polyethylene moderator is integrated in the cask structure so that possible influence on the effectivity will be compensated by decrease of the neutron source terms.

The effects of humidity and agents from potential fuel rod failures:

(1) Residual humidity is limited after loading. Maximum permissible condensate on the metal seal surface would result in ≤ 0.1 mm oxidation layer at outer liner of the metal seal,

(2) Attack of the seal surface by the most aggressive ions from defective fuel rods, Cs (one of the most reactive elements known) and Rb (from Kr-Decay) is minimised by limiting the surface area available for attack through machined tolerances and compression of the seal (about 1 MN/m),
(3) Theoretical diffusion rate reaches maximum value about 5 10^{-5} g/cm^2 after 40 years of storage and decreases afterwards.

3.2.3. Building and auxiliary systems

For both current and planned storage systems (dry casks) in the Czech Republic, the building does not serve as a nuclear or radiation protection barrier, these functions are provided by the cask. The cask integrity and proper functioning of auxiliary system is essential.

The building and auxiliary systems (including the crane for handling, monitoring system, etc.), which will require relatively low maintenance, are not viewed as the life limiting factors for the envisaged 70 years storage duration. Nevertheless, a lot of attention is paid in this area, including the stability of building and functioning of auxiliary systems, to ensure the safety of the storage system (individual casks) is not compromised.

3.3. SPECIFIC TECHNICAL ISSUES

3.3.1. The impact of temperature on cask long term integrity

The following temperature effects have been identified in Canada that may limit the life of concrete casks:

(1) Elevated temperature,
(2) Temperature gradient,
(3) Temperature cycling.

3.3.1.1. Elevated temperature

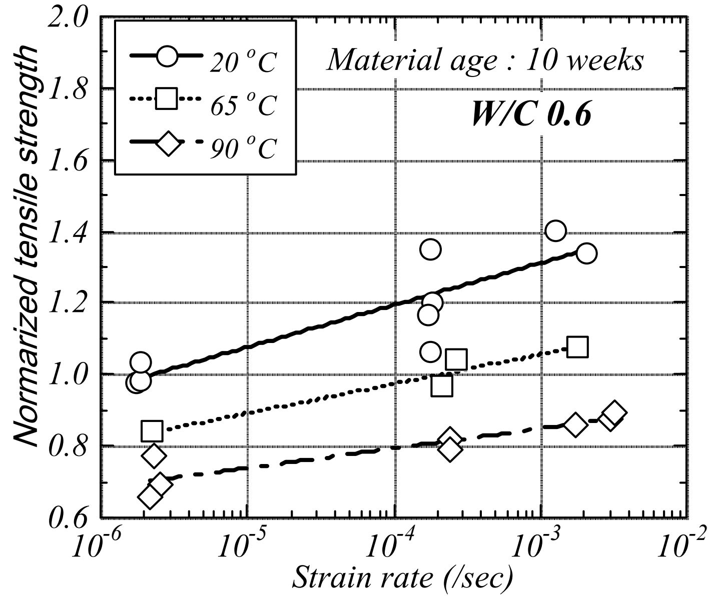
The impact of increasing operating temperature on the dynamic strength of concrete under accident conditions has been studied in Japan. The effect on concrete compressive and tensile strength for increasing temperature and strain rate is shown in Figure 9. These tests confirm that compressive and tensile strength decreases with increasing temperature for non reinforced concrete structures.

3.3.1.2. Temperature gradient

A temperature gradient across the concrete will affect free water distribution. One low cost measure of mitigating the impact of temperature gradients proposed in Canada would be to use silica fume blended concrete (CSA type 10SF), and a low water content of approximately 135 kg/m^3 to minimize free water. Adding the capability of draining excess water, via openings in the outer shell, should also be considered.

The Japanese, as part of their overall cask storage development programme, are specifically investigating the formation of cracks due to temperature gradients initiated in reinforced concrete structures. These investigations are being carried-out on full-scale concrete structures, some results of the heat transfer studies to date are shown in Figure 10.

(Tensile strength)



(Compressive strength)

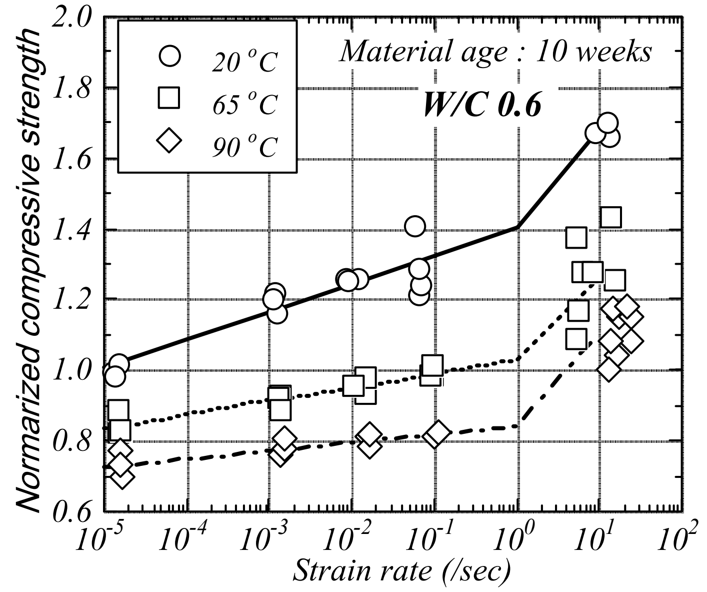


FIG. 9. The effect of temperature on concrete properties under accident conditions (Normal concrete: W/C 60%).

3.3.1.3. Temperature cycling

The effect of temperature cycling, where there are extremes in temperature, e.g. in Canada, may be a life limiting process for concrete structures. Prevention of initial cracks by limiting the temperature gradient across the concrete (e.g., loading older than 10 year cooled fuel) and by suitable choice of aggregate and cementitious materials would be a low cost solution.

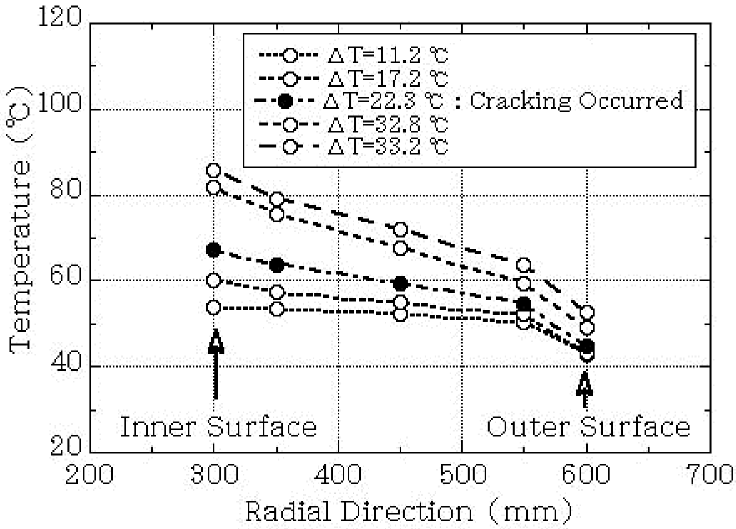


FIG. 10. Example of the heat transfer test.

Further analysis of this phenomenon, including stress relaxation using non-linear finite element methods is proposed in Canada.

Freeze-thaw/ cracking model (Canada)

The mechanisms that effect crack propagation in concrete dry storage containers (DSC), due to freeze-thaw cycling are described:

(1) Pre-existing cracks:

Cracking of the outside (cold) concrete is predicted to occur if the DSC wall is subjected to a high temperature gradient. The maximum temperature gradient would occur when a DSC is first loaded with fuel that has cooled for the minimum time. The nature of the cracking is not known but it is expected to be a number of shallow, narrow cracks. These cracks will partially close as the temperature gradient diminishes over time. In the case of micro-cracks, self-healing can be expected to occur when excess (free) moisture reacts with the non-hydrated cement in the crack region.

(2) Free-water:

Free water will initially be evenly distributed throughout the concrete. After loading the container with fuel, water migration from the warm region to the cold region will take place via capillaries in the concrete. However, even if capillaries become fully saturated, freeze-thaw damage is expected to be minimal or not to occur, if the concrete has an appropriate distribution of voids provided by the air-entrainement process.
When the DSCs are exposed to below freezing temperatures, the free water would saturate the outside (cold) layer of concrete, condense on the inside of the outer shell, and fill the gap between the steel and concrete as well as any pre-existing cracks. Water would run down the gap and saturate the lower regions of the concrete mass.

(3) Freezing:

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

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

The following study to assess the potential damage from freeze-thaw cycles via direct examination of concrete samples is proposed inCanada. It is recommended that core samples be removed from the prototype Concrete Integrated Containers (CIC) and from a currently empty Dry Storage Module (at Pickering) for examination. This assessment should correlate in situ moisture patterns with actual thermal history of the CIC. If necessary, the test containers should be subjected to thermal gradients, to monitor moisture migration.

3.3.2. Shielding studies (Japan)

The shielding studies currently being conducted in Japan for concrete casks are typical of the supporting data required to confirm that specific country/company internally set dose limits are met. Some of the studies are outlined below:

3.3.2.1. Evaluation of dose rate due to streaming through air inlet and outlet

It is very important to estimate the dose rate due to streaming through air inlet and outlet. A numerical code to evaluate shielding performance, by Monte-Carlo calculation method, is used for analysis. A full scale-model of the cooling ducts is also planned, to investigate the accuracy of the Monte-Carlo calculations and the model used.

3.3.2.2. Effects of cracks in the concrete body on shielding performance

As a result of thermal stress, cracking of the concrete body occurs. The impact on shielding performance therefore will depend on the width of the resultant cracks. Since there are no design shielding criteria in relation to permitted crack size, Monte-Carlo based analyses, using a slit model to simulate concrete cracking, has been carried-out. Figure 11 shows some results from slit modelling. The modelling shows that shielding is unaffected for cracks up to 0.4 mm in width.

3.3.2.3. Evaluation of dose rate at site boundary (Japan)

The conventional design approach to evaluating the resultant dose rate at the site boundary of a spent fuel storage facility uses either point or uniform volume source models, without taking any account of "mutual shielding effect" between casks. If the real geometry of the cask storage area is considered, then mitigation of the dose rate at site boundary can be expected because of this effect. A rational design approach has been proposed in Japan, which will consider the real geometry of the cask storage area.

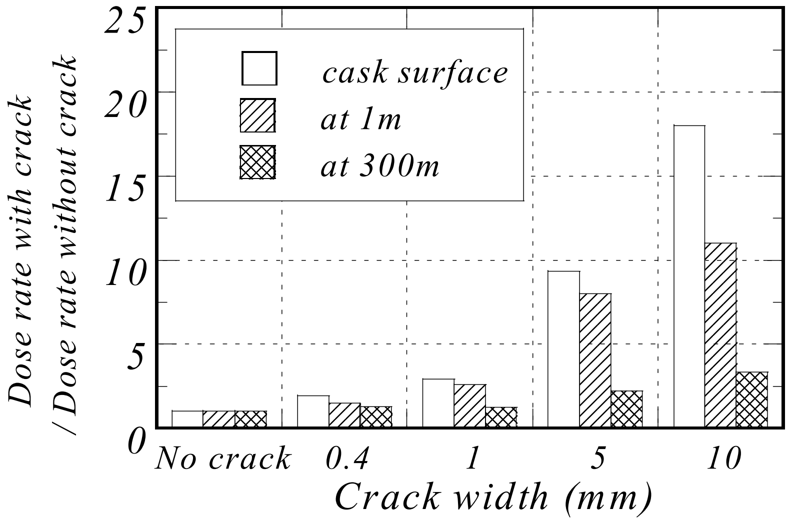


FIG. 11. Influence of crack width on shielding performance.

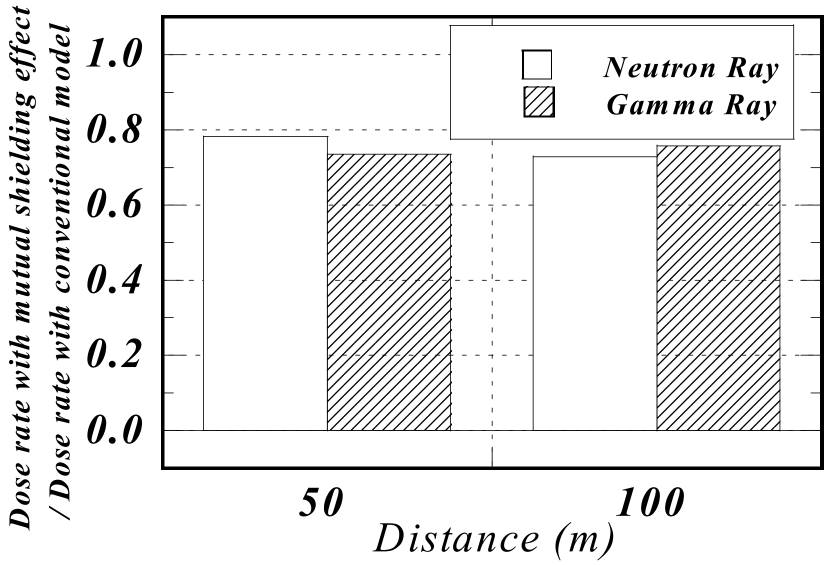


FIG. 12. Mutual shielding effect.

Application of Monte-Carlo modelling is being used to develop the rational approach. Figure 12 shows some initial results. It seems that an additional shielding effect due to the "mutual shielding effect" can be expected.

3.3.3. Long term integrity of seals

3.3.3.1. Current studies

Seal material performance

A series of containment tests with several types of gaskets are being performed in Japan. The following data is being collated in the course of these studies:

– Stress relaxation features of metallic gaskets,
– Relationship between the temperature/time dependence of the plastic deformation and the containment features of the metallic gaskets,
– Long term evaluation.

The performance of two full-scale metallic gasketed storage cask lid models has been monitored for around eight and half years and is continuing. To date very low permeability has been observed as is shown in Figure 13. The performance tests are carried out under accelerated condition, therefore the storage duration is equivalent to around 30 years.

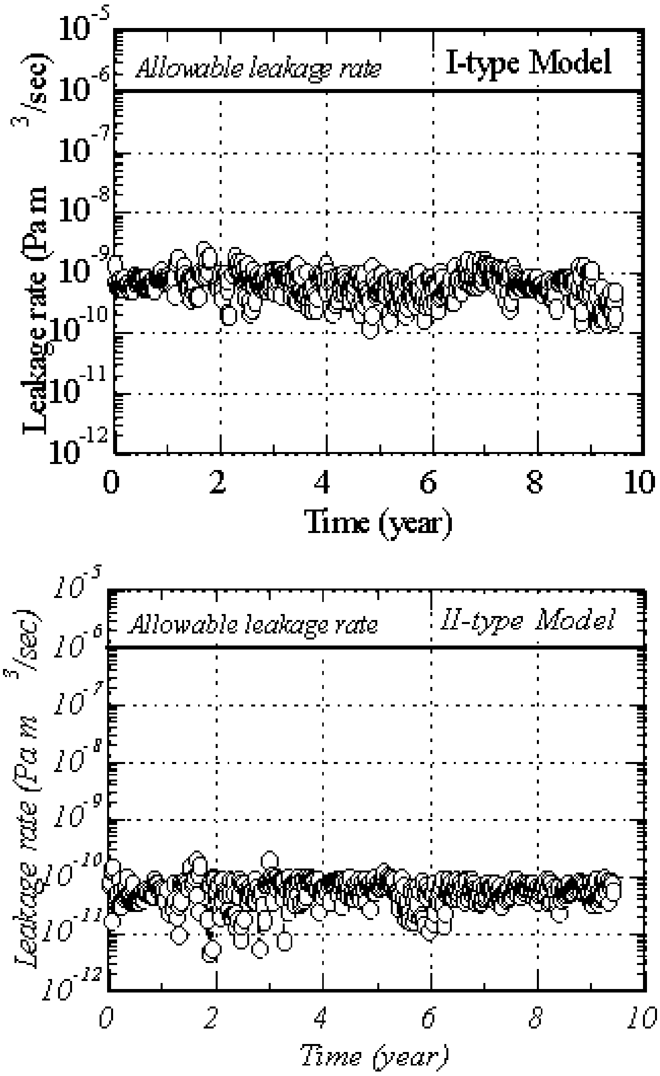


FIG. 13. Sealing tests of the metallic gaskets with two full-scale lid models of the storage cask.

Impact of corrosion on cask sealability

A series of corrosion tests, evaluating the impact on cask sealing, from aggressive chloride attack, which included natural exposure and accelerated tests, have been conducted recently in Japan. The following tests were carried-out to evaluate the long term behaviour:

– Loading effect to lid bolts, through corrosion of surface contact area between cask body and lid surface,

- Tensile load to lid bolts was progressively increased, with increasing corrosion of the surface area between cask body and lid surface. The actual effect to the cask was found to be very small, even under severe corrosive conditions. It has been concluded from this study that the integrity of cask sealing can be maintained for at least 40 years,
- Effect of corrosion at gasket interface on sealability.

Through small-scale storage cask lid modelling and by natural exposure tests, the effect on sealability as a result of corrosion at the gasket interface has been evaluated. It has been concluded that cask sealing can be maintained for a minimum of 40 years.

3.3.3.2. In-Service Experiences

Germany (GNB):

- 25 years of CASTOR® seal type in use, without systematic decrease of properties,
- 10 years under cyclic loading 20°C–130°C, without systematic decrease of properties,
- 300,000 seals exposed to UF_6-atmosphere for > 10 years, without damage,
- Fatigue fracture of internal gasket spring is only possible at elevated temperatures (> 380 °C), c.f. max. temperature on loading is ≤ 100 °C.

USA (Surry NPP)

Dominion Power has different types of storage casks in operation, among others, TN-32s and Castor X/33s. To date there has been two reported occurrences of secondary lid failure. In neither case was primary containment affected. In both cases the failure of the metallic lid gasket was as a result of the right corrosive conditions being set-up. The events highlight the need for careful cask preparation and the importance of ensuring there is no water ingress.

The failure mechanisms are summarized below:

TN-32

The initiating event leading to lid gasket failure was water ingress through a leaking Conax connector. The accumulated water (which would have had a high chloride content given the NPP location) when combined with residual boric acid from loading operations lead to accelerated pitting corrosion of the high purity aluminium seal. Galvanic attack through metal coupling is then believed to have been initiated, which lead to the final failure of the O-ring aluminium jacket. Alternative seals for such duties are currently being evaluated.

Notably, given that the TN-32 is a one-lid two-gasket system, remediation involved returning the cask to the reactor pool.

Castor X/33

In this particular case the cause of failure of the secondary metallic O-ring seal was determined to be corrosion at the cask side interface. The cause of the corrosion is unknown, although it is believed to be the result of residues from lid cleaning.

3.3.3.3. Country specific requirements for seals (Italy)

In Italy due to the peculiar situation (no facility for possible cask repair until the final repository), the confinement system must offer the highest degree of reliability and flexibility (e.g. detection of leaking gasket and changing one gasket without the need for a hot cell).

Although a 2-lid/2 gaskets solution appears to fulfil the above requirements, a 1-lid (2 gaskets) solution could be accepted, if it can be demonstrated that there is a high degree of independence of the gaskets, and the required flexibility.

3.3.3.4. Casks loading operational experience –sealing aspects (Lithuania)

The need for pool water cleanliness has been emphasised from Lithuanian Castor cask loading experiences. In the case of their first cask loading, difficulties were experienced in obtaining the required leak-tightness due to the migration of crud, entrained on the fuel assemblies/baskets, into the cask primary seal region. Difficulties were also experienced with loading the second cask, in this case the root cause was concluded to be prolonged in-pool residence time. Seal inspection showed that extensive oxidation had occurred, which resulted in crack formation on the seal, when the cask lid was tightened.

Appropriate measures have now been taken to eliminate reoccurrence of these problems. All subsequent cask loadings have been carried out successfully.

3.3.4. Corrosion of outer carbon-steel wall liner in dry storage containers (Canada)

The potential for corrosion of the DSC bottom in areas where the external protective coat might develop defects, exists. The possibility of making the whole DSC in stainless steel has been considered.

Corrosion prevention

The DSCs are protected by a dual layer of epoxy paint. The application of an additional layer of coal tar epoxy coating (which provides sealing effects) to the bottom plate is recommended. A thicker bottom plate, up to 19 mm (0.75 inch) should be considered to provide added protection.

It is recognised that a low risk of galvanic corrosion exists due to the accumulation of water, in the bottom part of the DSC, particularly in the weld areas. The risk of galvanic corrosion is minimised by the proper weldment design and welding procedure. This is expected to provide adequate protection against corrosion, resulting from the micro structural gradient across the heat-affected zone. It is recommended, however, that the possibility of galvanic corrosion at the joints, where stainless steel (e.g. vent and drain ducts) is joined to C-steel plate be investigated in more detail.

The Bruce DSCs (BDSC) will be placed on concrete platforms. The bottom of these containers may come into contact with trapped water or moisture, and corrosion will occur only if the coating is damaged. Coating damage will most likely result from rubbing the container bottom against the concrete platform, during the initial placement. Raising the containers by using rubber mats or other means was considered. This idea was rejected, however, because of container stability considerations and the absence of information on the long term performance of an appropriate material.

In-service inspection

This assessment has concluded that corrosion of the BDSC leading to any serious structural damage over the service life will not happen. However, there is a chance for minor corrosion to occur, if there is a defect in the paint/coating. This defect may be induced during the handling of the DSC, or due to the natural deterioration of the coating system. This minor corrosion may occur at a slightly higher rate around the lid closure weldment region because of the residual stresses. Similarly, the bottom plate region may have a slightly higher corrosion rate, due to potential presence of stagnant water and the potential for crevice corrosion. Therefore, in-service inspection should focus attention on these regions. In general, an overall inspection for signs of external rusting or paint damage/cracking should be adequate. Periodically, a few randomly chosen DSCs should be raised for visual inspection of the bottom plate for damage to the paint layer. The paint layers of suspicious regions should be removed for inspection of the underlying metal surface.

The DSC coating system is expected to have a limited useful life, typically about 25 years. It is recommended that at the end of their useful life, the paints/coating be re-applied to the BDSCs.

3.3.5. Welding of multi-place leak-tight basket (MLB) — otherwise known as canister for VSC casks (Ukraine)

Canister for VSC casks

Analysis of steel SA 516-70, used in the fabrication of storage baskets shows the carbon content to vary between manufacturers, 0.17 and 0.23%. These compositional changes affect weldability. When the carbon content is 0.23% satisfactory welds can be achieved, however, this is not the case when the carbon content drops to 0.17%. To overcome these problems, the Institute of Electric Welding (named after E. Paton) of the National Academy of Sciences of Ukraine have developed new welding procedures for MLB sealing.

The procedures were first developed in the laboratory and then tested at Zaporizhya NPP on MLB mock-ups. This has resulted in a “Technological instruction (procedure) for the welding of lids to Multi-place Leaktight Basket at dry SNFSF” TIC 218-0261-98. It should be noted that Sierra Nuclear Corporation have also performed studies, identified the root causes and developed a procedure for taking corrective action (see Table XIV).

3.4. DESIGN LIFE/OPERATING LICENCE EXTENSIONS (CASK STORAGE)

3.4.1. 20 year extension to operating licences (USA)

The NRC granted 20-year operating licences in July and August 1986 to Virginia Power Company and Carolina Power and Light, respectively, to operate ISFSIs at the Surry and H.B. Robinson plants. Those facilities will require renewal of licences by 2006. In this regard, the NRC has established a Task Force to address “Site Specific Independent Spent Fuel Storage Installation (ISFSI) Licence Renewal”. The licence renewal task force will prepare regulatory guidelines, to enable reviewers to make a finding that the effects of ageing on dry cask systems, structures, and components (SSCs) and support activities are understood and will be managed such, that the ISFSI can be operated during the renewal period (the next 20 years) without undue risk to the health and safety of the public.

Table XIV. List of correcting measures related to manufacturing and sealing of Multi-place Leaktight Basket

Root cause	Correcting actions
Defect of casing material: welds without proper documentation	1. Etching of MLB inner surface at points of lid welding. 2. Ultrasonic inspection of welds at lids. 3. Certification of MLB. 4. Use of steel with low content of sulphur to manufacture structural elements of MLB.
Error during assembling of lid and under slating ring	1. Assembling with strict observation of technological scheme. 2. Improvement of weld gaps which value exceeds that established by the technical documentation.
Moisture ingress into weld	1. Provision for water level inside MLB below protecting lid. 2. Ventilation of below-lid space during process of welding. 3. Preheating up to 100 ^{0} C.
Hydrogen cracking	1. Preheating up to 100 ^{0} C. 2. Subsequent heating at 100 ^{0} C during 1 hr. 3. Use of welding materials with low hydrogen content. 4. Provision for uniform distribution of compressing forces under cooling of weld. 5. Provision for interruption during 2 hrs between completion of welding and start of weld joints quality inspection. 6. Use of materials with low carbon coefficient for manufacturing of MLB.

According to Task Force information, the operation of an ISFSI during a renewal period must maintain safety criteria to minimize dose to the public and to avoid the release of radioactive material. Those criteria include subcriticality, shielding, confinement, heat transfer and structural integrity.

The following issues will need to be addressed as part of the renewal process:

(1) The intended functions of the pertinent SSCs of the ISFSI,
(2) Degradation mechanisms and effects (ageing) for specific SSCs,
(3) Existing maintenance and monitoring programs and proposed modifications thereto,
(4) Changes in environmental factors and conditions for the ISFSI,
(5) Realised and predicted changes in site-specific characteristics,
(6) Operational experience at this ISFSI and at others with similar or identical cask designs,
(7) Applicable reported or non-reported ISFSI or cask events, and
(8) Realised and projected decreases in thermal loading and source term for loaded casks.

3.4.2. Design life extension beyond 50 years casks (Lithuania)

Although the Castor and Constor systems being utilised at Ignalina NPP are licensed for up to 50 years, an extension to the operating life of these systems is likely based on:

(1) Cask materials are stable, verified for neutron loads up to $10^{19}/cm^2$
(2) Inner loads are not limiting, as:
  - Metal seals (only CASTOR) are only loaded with a neutron fluence < 10^4 n/$cm^2 \cdot s$,
  - Elastomeric seals (only CONSTOR) have no containment function during storage. This function is guaranteed by full metal containment,
  - For CASTOR, hypothetical oxidation of the metal seals from residual humidity after vacuum drying is < 0.1 mm.
(3) Outer loads are insignificant as well, because:
  - Verified mechanical/physical properties of all containment parts:
    - Cask Body
    - Lids
    - Bolts
    - Sealings (metal seals: CASTOR and weldings: CONSTOR),are guaranteed for unlimited fulfilment of tightness and shielding criteria,
  - Manufacturing is performed according to EN ISO 9001,
  - Corrosion-resistant state can be verified during all handling and storage stages.

4. RESEARCH AND DEVELOPMENT

This chapter summarises the R&D-activities, by country, in support of new storage technologies, extensions to storage periods and the impact of MOX & high burnup UO_2 fuels.

4.1. BELGIUM

An international programme named STONE (STOred UO_2 and MOX Nuclear Elements) has been proposed by BELGONUCLEAIRE. The aim of the programme is to answer to questions such as:

(1) Does MOX fuel show the same performance as UO_2 fuel? (Taking into account the significantly high energy deposition and decay heat, as well as its helium generation during storage, Figure 14 and 15).
(2) Can the conclusions obtained for low burnup fuel be extrapolated for high burnup fuel?
(3) Have all mechanisms affecting the fuel rod performance during storage been identified?
(4) Is there any possible lower temperature (= no restoration) low kinetics dependent damaging mechanism (resulting for example from energy deposition and helium production) that could lead to fuel swelling and embrittlement?

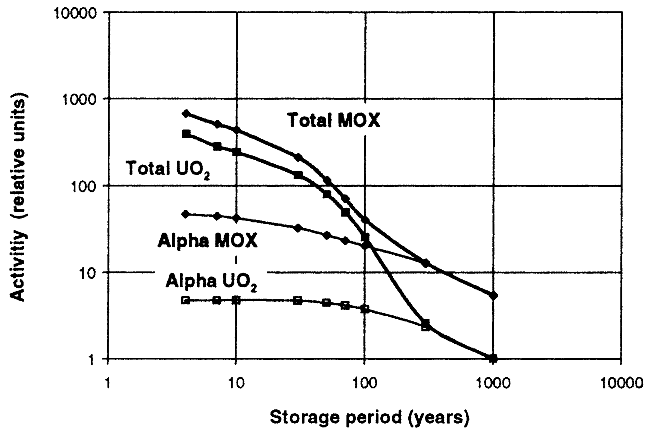


FIG. 14. Total and alpha activities.

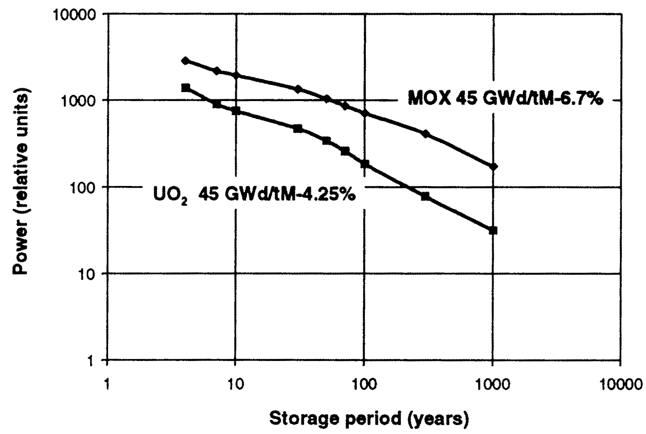


FIG. 15. Residual thermal power during storage of UO_2 and MOX fuel after irradiation to 45 GW·d/t HM.

The plan is to characterize UO_2 and MOX spent fuel rods with burnup up to ~ 50 GW·d/t HM and stored for more than 20–25 years at SCK-CEN in water pool or in dry conditions. Since the fuel was extensively characterized prior to storage, the effect of storage conditions on fuel and cladding performance can be characterized accurately.

The fuel rods selected for post-storage examination cover a broad range of fabrication, irradiation and storage conditions. Examples include:

(1) Rod internal pressure (1 or 20 bars at fabrication, up to 60 bars (cold condition) at end of irradiation),
(2) Range of burnup (up to 50 GW·d/t HM rod average),
(3) UO_2 and MOX (up to 8.3 % Pu/U + Pu),
(4) PWR and BWR rod designs and operating conditions,
(5) Wet conditions of storage for most rods with some fuel pieces stored in dry conditions.

The programme is intended to examine the cladding with regard to outer corrosion, mechanical properties and deformation under inner pressure, and the fuel with regard to integrity (microstructure, swelling) and chemical stability. Rod examinations will provide the inner gas pressure and composition.

Special attention will be given to MOX fuel by performing additional helium release tests. Additional studies of fresh MOX fuel doped with an α-emitter have also been proposed. These studies hopefully will lead to rapid conclusions on the possible influence of helium generation on long term fuel storage performance.

The current selection of rods proposed for examination is based on the availability of rods stored at SCK-CEN. The examination of other rods (e.g. analysis of not tight rod stored in wet conditions) or from alternative storage sites may also be possible.

4.2. FRANCE

Part of the R&D programme launched by the Commissariat à l’Energie Atomique (CEA) looks at the long term behaviour of nuclear waste packages (CLTC programme) including spent fuel (SF). The objective of the study is to assess the ability of various waste packages (SF included) to maintain their functions during interim storage and geological disposal. The emphasis is on containment and recovery for durations of 100 years and beyond, see Figures 13–20. It is also seen as important to obtain reliable data, to demonstrate safety performance and to support design work for long term storage.

A particular aspect of the CLTC programme explores how SF will undergo intrinsic evolution regarding its physical state (porosity, fracture network, surface area, rod internal pressure, etc.) and its chemical state (radionuclide inventories, radionuclide location between the rim, the grain boundaries and the UO_2 matrix, mechanisms and kinetics for radionuclide release, etc.) see Figures 21 and 22. This work is conducted in collaboration with the French utility company EDF.

During interim storage, the intrinsic evolution of SF (matrix and cladding) is expected to take place within a closed system with no exchange of matter with the external medium. Helium will be produced by Pu daughters and for MOX fuel significant amounts can be expected, see

Figure 23. The data available with respect to helium diffusion and solubility properties is limited. Helium generation raises the following questions:

(1) Will it result in over-pressure within the cladding or/and in the swelling of the UO_2 lattice?
(2) Will the cladding remain an effective barrier over time and for how long?

These questions must be answered to help design a safe storage facility along with its operating procedures.

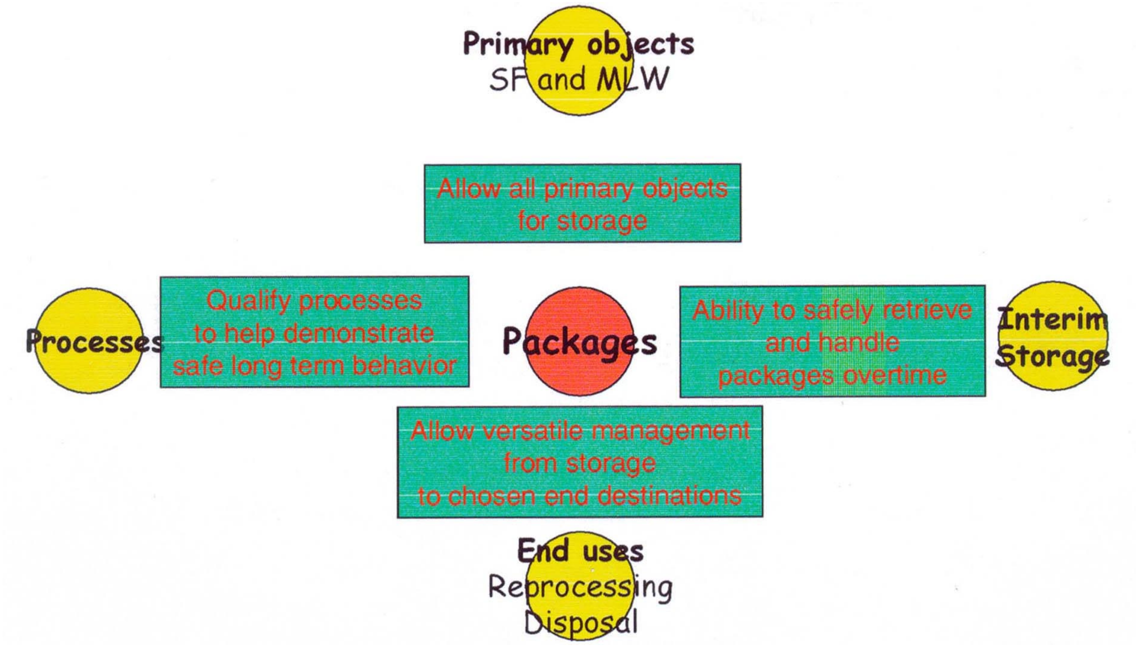


FIG. 16. Design goals and requirements for spent fuel and MLW packages.

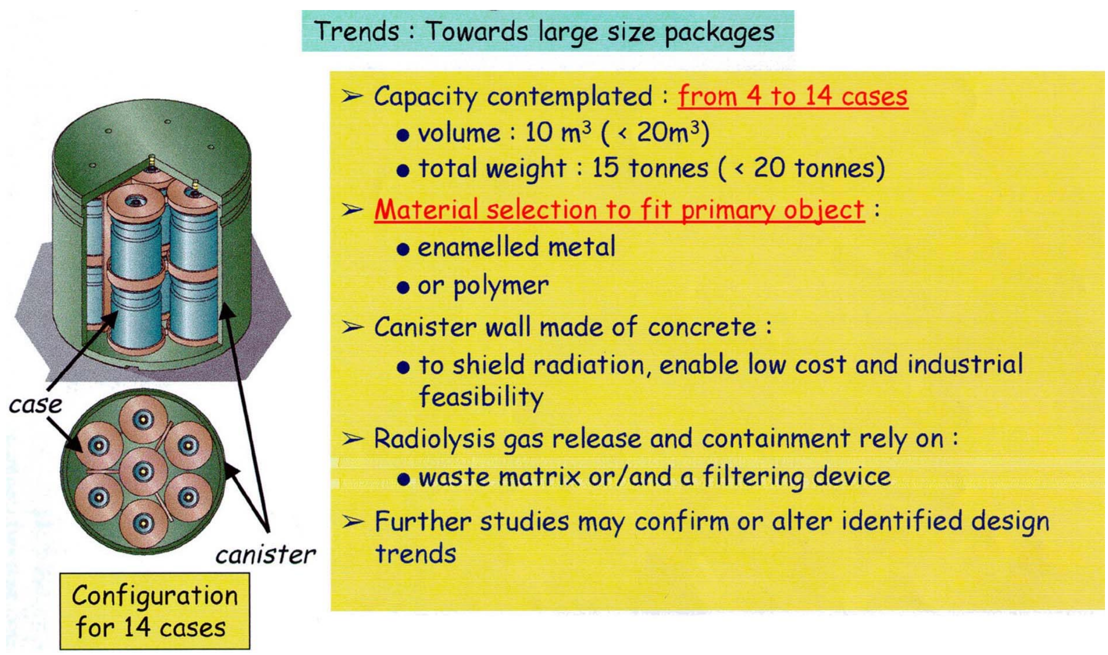
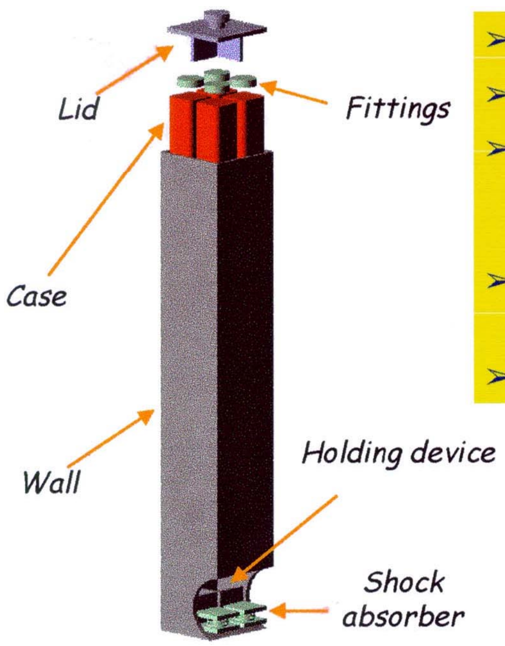


FIG. 17. MLW multipurpose package design trends.

case : 1st confinement barrier for the time scale envisaged
An add-on to the cladding confinement role
canister : second confinement barrier
barrier integrity monitoring device : check each barrier performance

- No corrosion allowance in sizing wall thickness
- Conditioning process must :
  - restrict water presence within the package below a set target
  - guarantee surface contamination below acceptable limit for case and canister outer walls
- Additional requirements needed to allow transportation and radiation shielding

FIG. 18. SF multipurpose package — storage design trends.



- Thickness ~ 50 mm
- Materials : carbon steel or cast steel
- Capacity of 4 cases : fit disposal requirements
  - for UOX SF
  - not for MOX SF
- Closing by welding including the provision of an optional bolted sealed lid
- Square cross section

Length	: ~ 5500 mm
Cross section	: ~ 750x750 mm
Total weight (loaded)	: ~ 13 tonnes

FIG. 19. SF multipurpose package for storage canister — design trends.

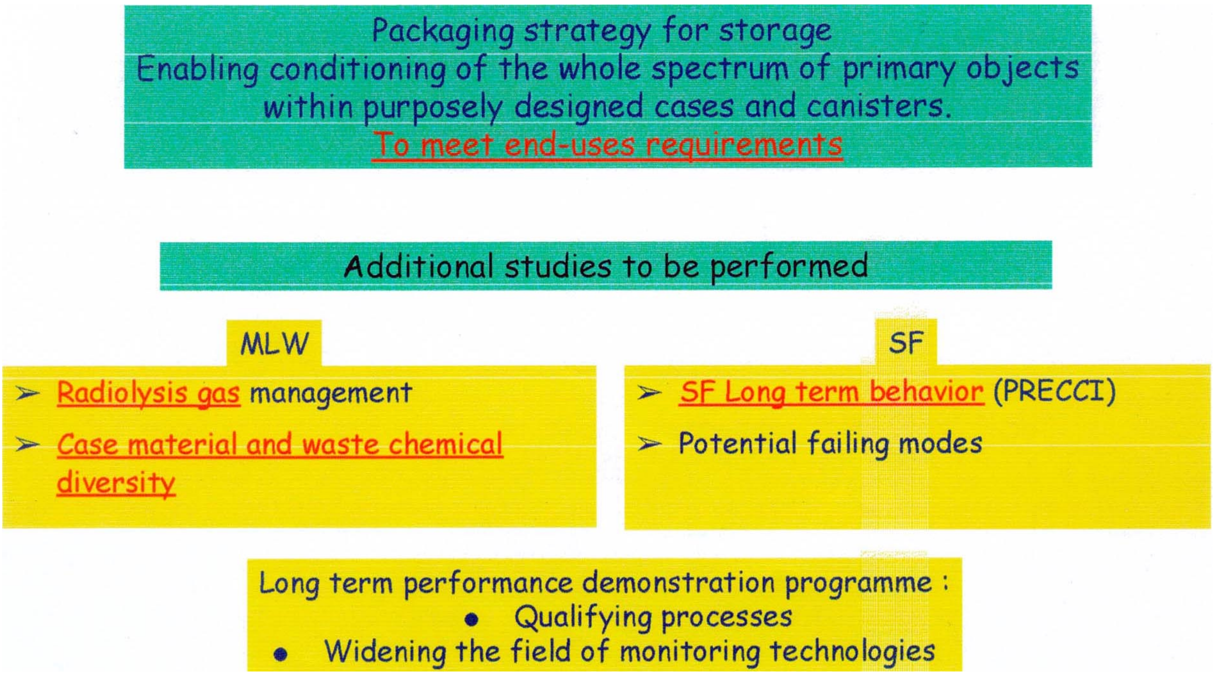


FIG. 20. Conclusions.

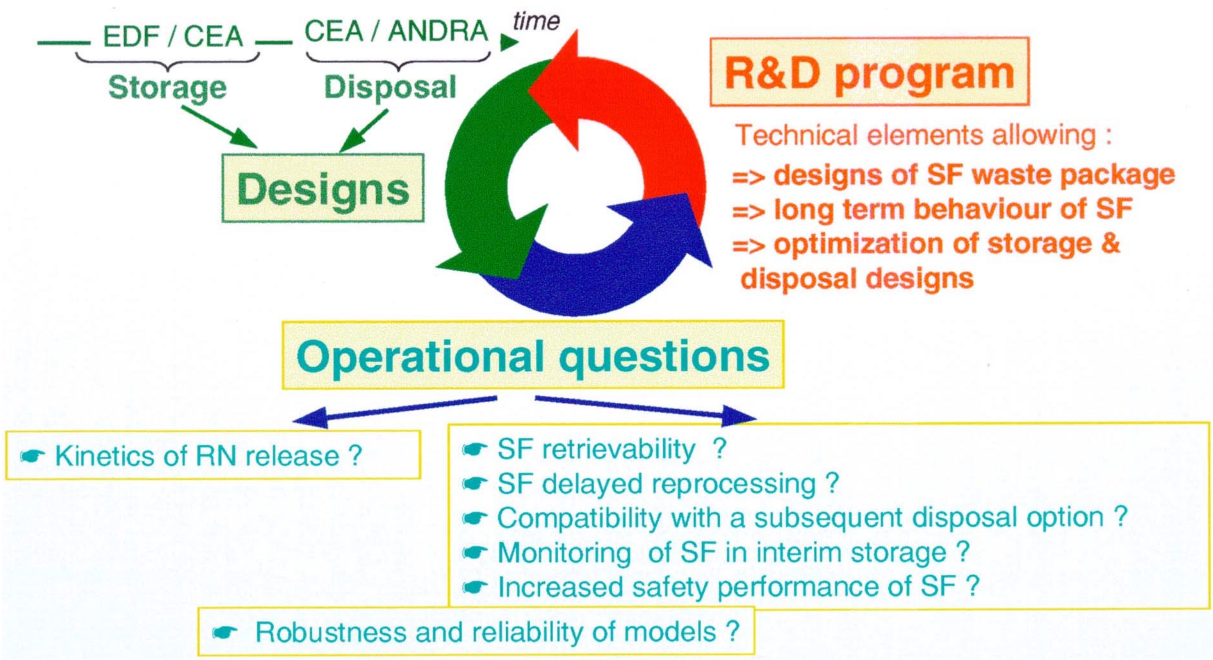


FIG. 21. R&D programme objectives.

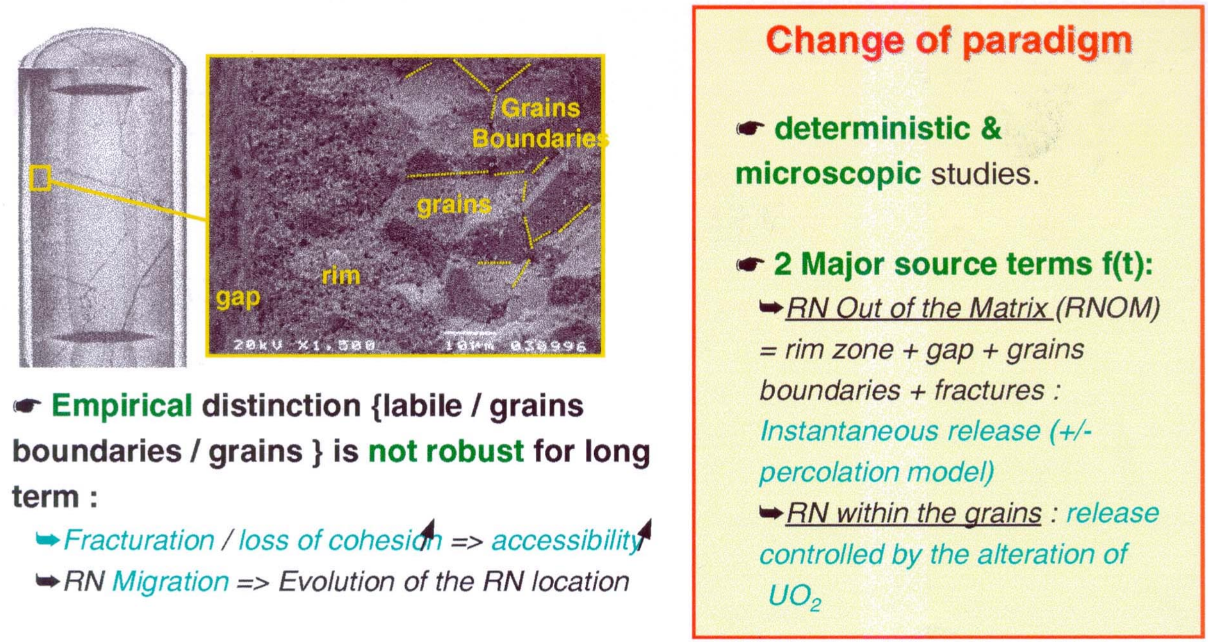


FIG. 22.Radionuclide source terms, French model.

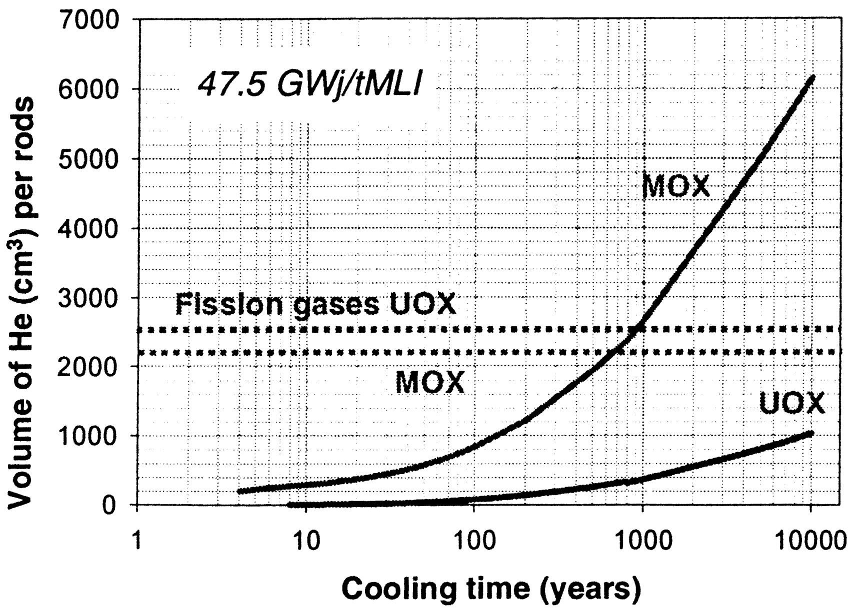


FIG. 23. Estimation of He production during irradiation and storage conditions due to α decay for 47.5 MW·d/t U UOX and MOX.

4.3. GERMANY

The main objective of the R&D-work is to assess and assure the integrity of the fuel under changing boundary conditions such as increasing burnup, improved materials or longer periods of time. Since licensing aspects are closely linked to this topic, the following chapters cover both.

4.3.1. Burnup perspective of MOX and UO_2 fuel

The economics of nuclear power generation have been continuously improved by increasing the discharge burnup of the fuel. From the early 80s the burnup of the peak reload batch of PWR fuel has been steadily increased from about 30 MW·d/kg U to more than 50 MW·d/kg U by 1999. BWR fuel exhibits a similar increase, however, lags about 5 MW·d/kg U behind. By 2005, peak average discharge burnup of 55 and 50 MW·d/kg U, respectively, should be achieved.

Burnup increase has been and will be achieved by increasing the fuel's enrichment and has to be accompanied by improved material properties such as the corrosion resistance of the cladding. Burnup development, however, is generally limited by the enrichment licenced by the authorities for handling unirradiated fuel during manufacturing or storage at the nuclear power plants. At present the max. ^{235}U-enrichment, permitted to be handled worldwide is 5 %. This is equivalent to a maximum FA-burnup of around 70 MW·d/kg U (Table XV.). Precursor UO_2-FAs with a maximum enrichment of 4.6 w/o are being irradiated. They will reach a maximum FA-burnup of about 65 MW·d/kgU after 5 irradiation cycles in 2003.

Table XV. Current high burnup limits and near term development of FA-burnup

Current limits:	UO_2-FA: MOX-FA:	^{235}U = 5 w/o $Pu_{fiss} \approx 6$ w/o	→ 70 MW·d/kg U → 70 MW·d/kg HM
Expected FA-burnup (by ≈ 2003)	UO_2-FA: MOX-FA:	^{235}U = 4.6 w/o Pu_{fiss} = 4.6-4.8 w/o	→ 65 MW·d/kg U → 60-65 MW·d/kg HM

For MOX-FAs the policy of burnup equivalence to UO_2-FAs needs a Pu_{fiss}-content of about 6 w/o to reach 70 MW·d/kg HM. At present, the maximum averaged Pu_{fiss}-content licenced for MOX fuel is 4.8 w/o, which will result in a maximum FA-burnup of 60–65 MW·d/kg HM by 2003.

4.3.2. Aspects associated with high burnup fuel

4.3.2.1. Neutron-physical aspects

To achieve a higher burnup, for UO_2-Fas, a higher ^{235}U-enrichment and for MOX-FAs a higher Pu_{fiss}-content is needed. The consequence is generally a higher thermal load of the fuel to obtain a higher FA-burnup within the same irradiation time. If a certain amount of MOX-FAs replace UO_2-FAs, differences of the reactivity behaviour vs. time have also to be considered. The MOX reactivity curve is flatter than the Uranium one (Figure 24). The point of intersection depends on the relation between ^{235}U-enrichment and Pu_{fiss}-content. A preponderance of Pu_{fiss}-content shifts the point of intersection to the left and leads to a considerably higher thermal load of the MOX-rods at burnup beyond 20 to 30 MW·d/kg HM.

For dry storage, decay heat and neutron activity versus burnup is important, to assess cask temperatures and shielding capacities. A 25% increase in burnup from 55 to 69 MW·d/kg HM increases the decay heat of both UO_2 and MOX-FAs by roughly the same amount (Figure 25). The decay heat of a UO_2-FA is about 2 kW after 5 years cooling, however, is less than a half of a MOX-FA of comparable burnup. With increasing cooling time the decay heat decreases, however, at a much lower rate than UO_2-FA. For example it takes a UO_2-FA about 10 to 20 years to reach 1 kW decay heat compared to a MOX-FA, which takes between 100 to 200 years.

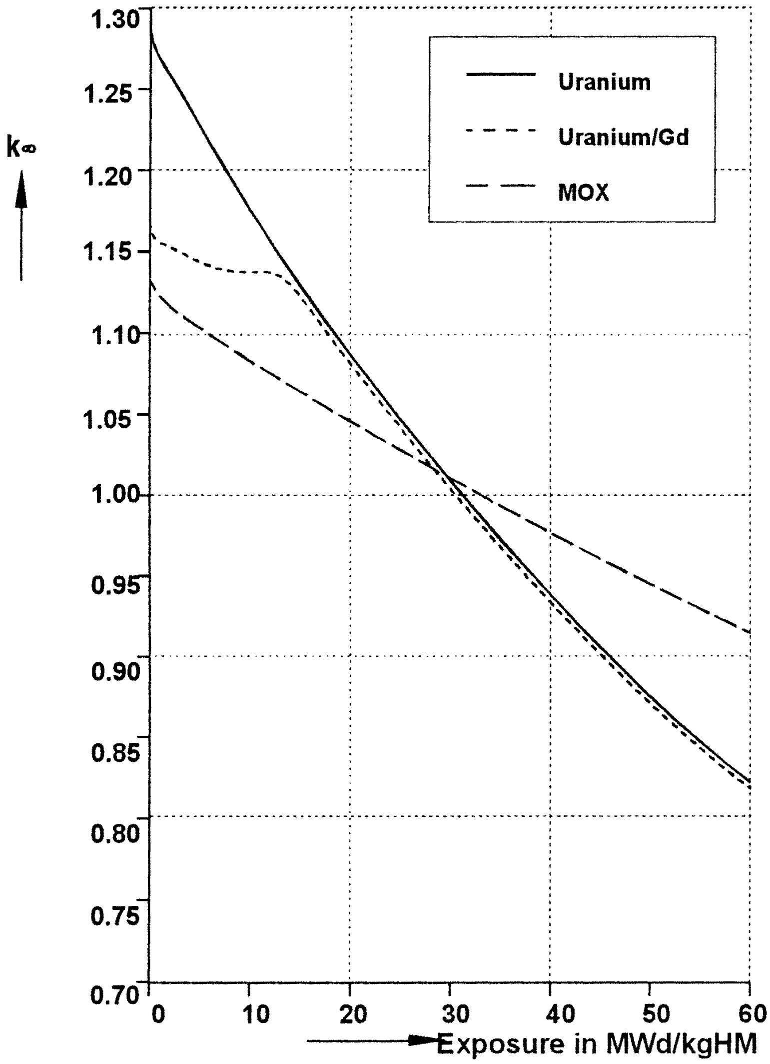


FIG. 24. Reactivity of MOX and UO_2 fuel versus burnup.

The neutron activity exhibits somewhat different features than the decay heat (Figure 26). During 10 and 100 years cooling neutron activity drops by an order of magnitude. Beyond 100 years the neutron activity remains relatively constant. The difference between MOX and UO_2-FAs is approximately a factor of ten.

With respect to an extension to storage duration, the graphs show that it takes about 100 years for MOX fuel to reach the decay heat and neutron activity levels of UO_2 fuel present at the beginning of storage.

4.3.2.2. Behaviour of fuel and cladding materials

The physical properties of PuO_2 and UO_2 such as crystallographic structure and lattice constant are very similar. PuO_2 and UO_2 have a fluoride structure with a simple cubic O^{2-} and a phase centred cubic (fcc) cation sublattice (Table XVI). The lattice constant is 0.55 and 0.54 nm, respectively. The Pu-distribution in the sintered MOX-pellet, however, is not completely homogeneous and depends on the degree of UO_2 and PuO_2 mixing in the course of the manufacturing process. Other physical properties of PuO_2, such as thermal conductivity or mechanical data, have proven to be only somewhat worse (thermal conductivity), comparable or even better, as is the case for thermal creep.

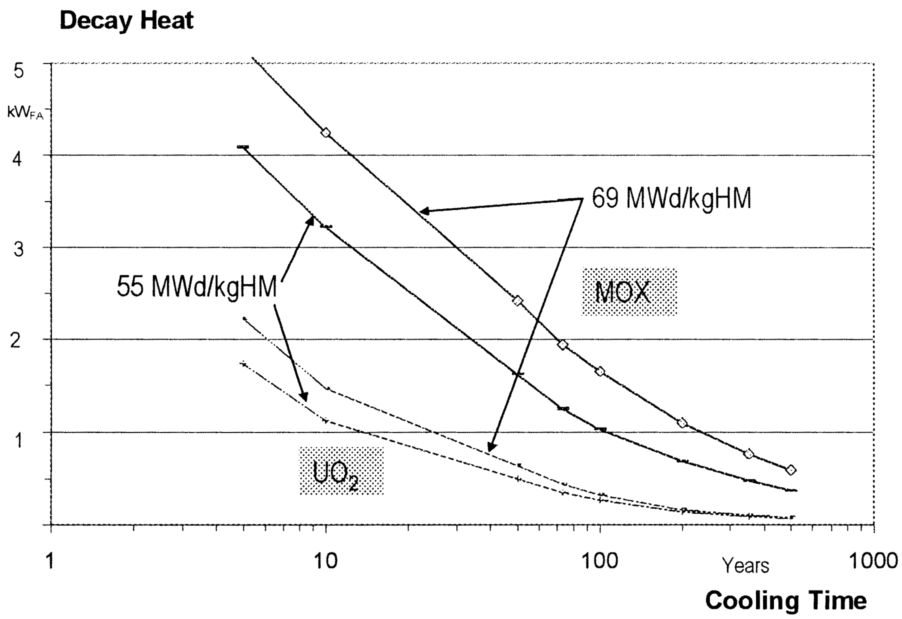


FIG. 25. Decay heat of UO_2 and MOX-FAs versus burnup and cooling time (FA-discharge burnup 55 and 69 MW·d/kg HM).

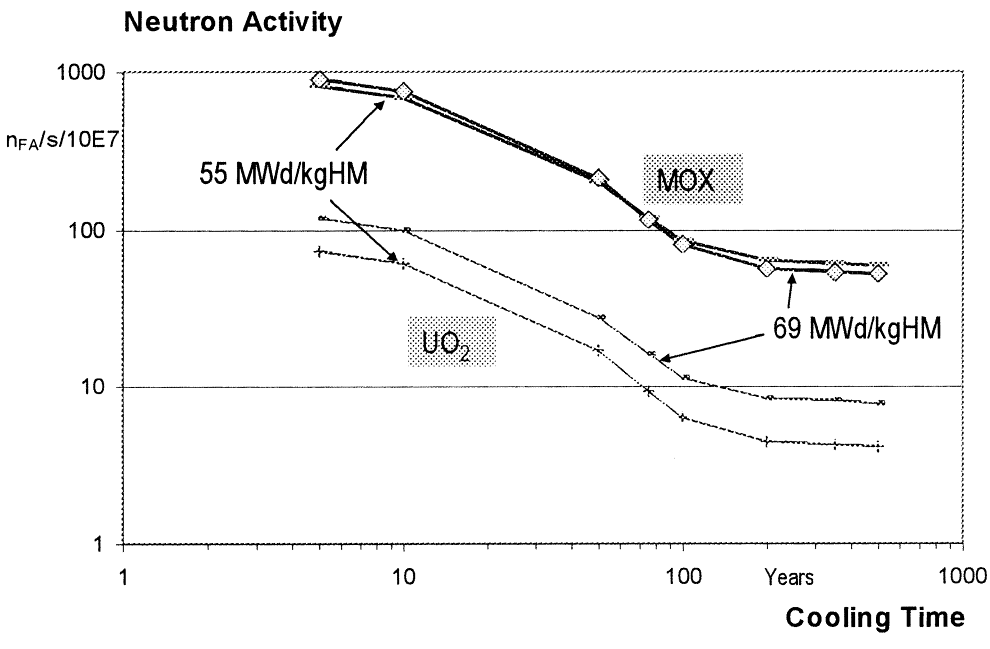


FIG. 26. Neutron activity of UO_2 and MOX-FAs versus burnup and cooling time.

Table XVI. Characteristic data of MOX fuel

Physical:	Chemical:	Nuclear:
Crystallographic structure: Fluoride type, space centred O^{2-}, fcc cation sublattice Lattice constant: 0.55 nm Pu distribution dependent on manufacturing process (MIMAS, SBR)	Gibbs free energy	Isotopic composition: ≈ 5–7 w/o Pu with tails or natural commercial Uranium ^{238}Pu-^{242}Pu: ≈ 1.5 / 60.1 / 24.5 / 8.8 / 5.0 w/o

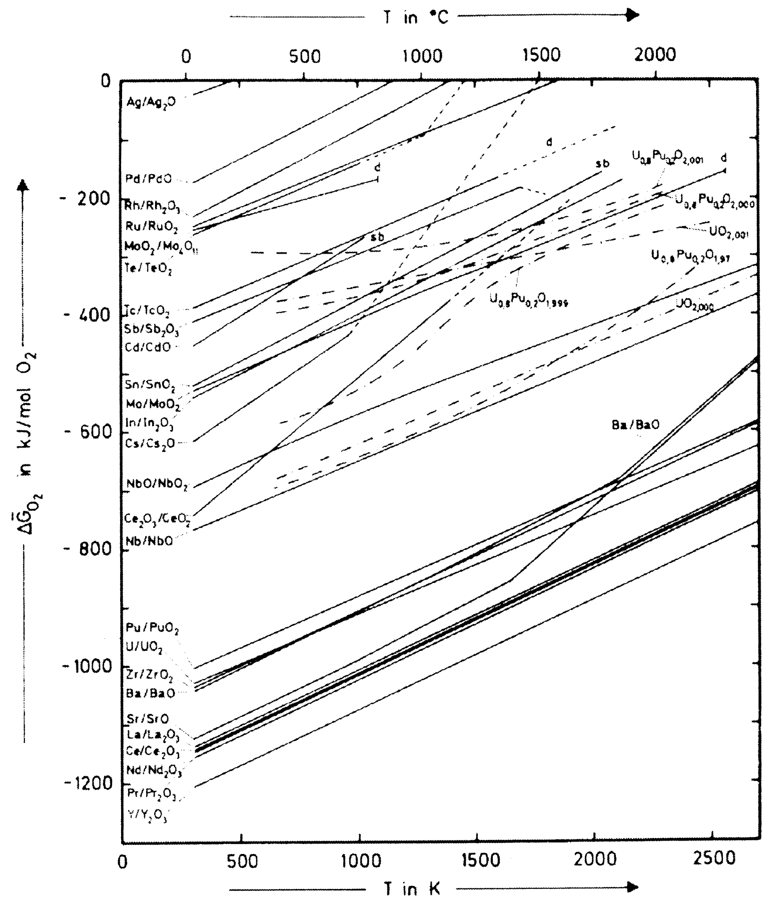


FIG. 27. Gibbs free energy of Oxygen of fission products, UO_2 and MOX.

The chemical behaviour of the fuel can be characterized by the Gibbs free energies of oxygen of the fission product oxides (Figure 27). From Figure 27 it can be seen that a U-Pu-mixture has a somewhat higher free energy than stoichiometric UO_2. The reactivity behaviour, however, such as the formation of stable oxides of Zr, Ba to Y and stable metallic forms of Pd, Rh, Ru and Te, is basically very similar. Therefore, no significantly different behaviour of MOX and UO_2 fuel should occur with regard to all chemistry related effects, like fuel-cladding interaction or stress-corrosion-cracking.

All post irradiation examinations performed up to now on MOX fuel have confirmed that the material behaviour is very similar to UO_2 fuel, e.g. the comparison of the fission gas release correlated to the relevant rod power. In spite of different microscopic structures MOX and UO_2 fuel technically behave the same (Figure 28).

To determine the rod internal gas pressure, the fission gas release of UO_2 and MOX fuel has to be determined as a function of burnup. For UO_2 fuel see Figure 29. Beyond a burnup of 60 MW·d/kg the rod internal pressure increases sharply with increasing burnup. For MOX fuel, the release situation can be assumed to be worst case, due to the higher end of life power, i.e. > 40 MW·d/kg, which results in a much higher internal gas pressure.

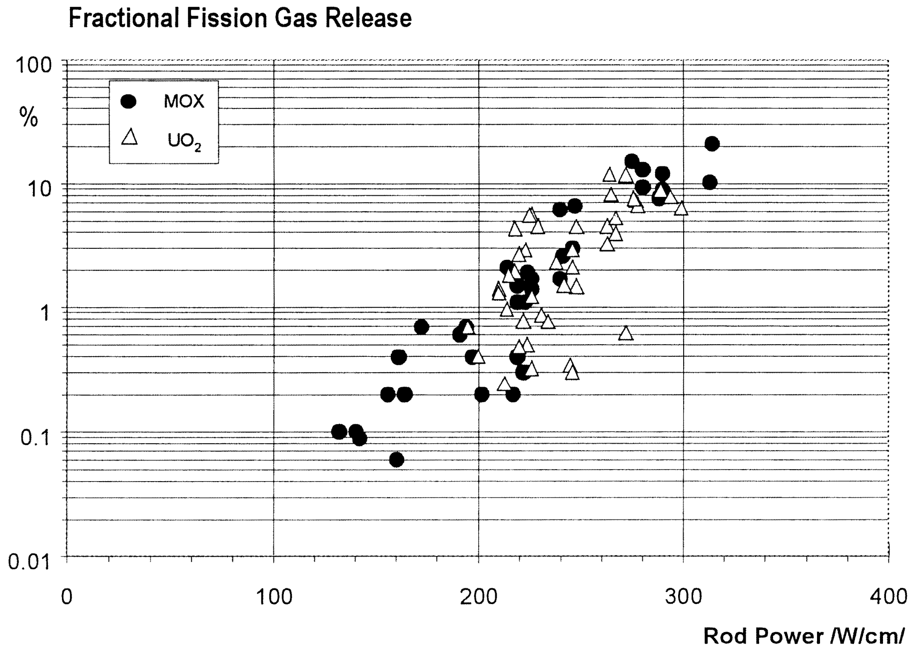


FIG. 28. Fission gas release of MOX and UO_2 fuel referred to the average rod power of the release relevant cycle (rod burnup < 40 MW·d/kg HM).

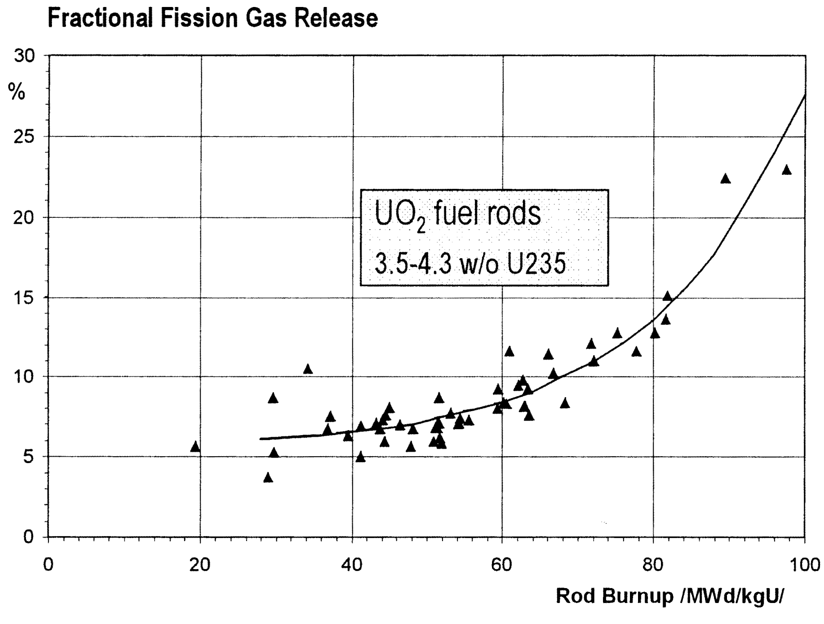


FIG. 29. Fission gas release shows an accelerated increase at high burnup.

Materials with greater corrosion resistance have been developed for high burnup and heavier duty demands. Their inpile behaviour is being tested to decide which materials are best suited under different irradiation conditions up to a burnup of 70 MW·d/kg HM and beyond (Figure 30).

4.3.2.3. In-pile licensing

In the last years, the in-pile licensing methodology has been steadily improved to cope with increasing rod internal pressures due to growing burnup or more demanding power histories.

Statistical methods for fuel rod design allow increasing burnup without changing existing design criteria. This can be done without any loss of safety, by getting a more precise description of the fuel rod behaviour. Such a method describes the rod behaviour in terms of statistically distributed fabrication data and model assumptions. The calculations yield the number of rods having a certain property, e.g. internal pressure. This approach considerably reduces unnecessary margins in the design calculations compared to a deterministic design approach.

Furthermore, the standards for the rod internal pressure have also been made more realistic (Table XVI). The most advanced data were derived from the ROPE experiments. This data is now being used to support new licensing applications.

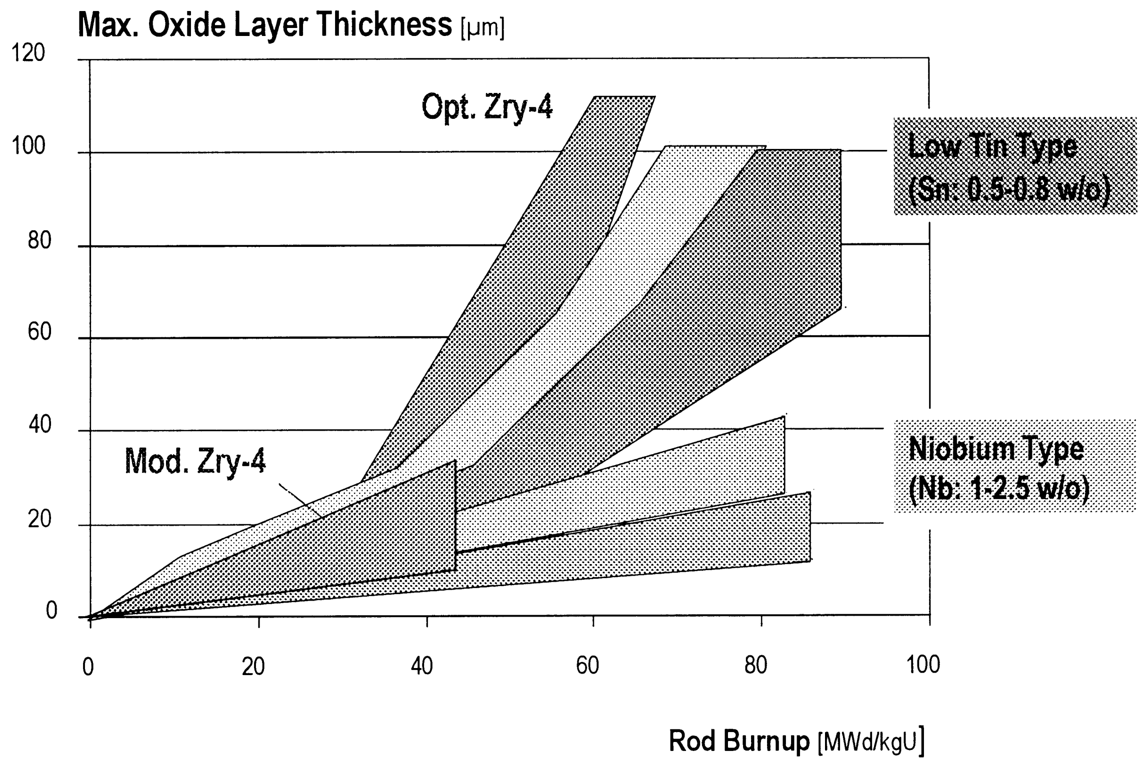


FIG. 30. Better corrosion resistant cladding materials minimize corrosion with increasing burnup.

Table XVI. Standards for the fuel rod internal pressure used by Siemens

1. Fuel rod pressure < coolant pressure

2. Non lift-off (1984)
   The gap between fuel and cladding shall not increase i.e. outward creep of the cladding ≤ pellet swelling

3. Limits for $d\varepsilon/dt$, ε_{max} and σ_{max} (1996) derived from the ROPE experiments

4.3.3. Aspects associated with dry storage of spent fuel

4.3.3.1. Degradation mechanisms

The Zr-based-cladding of spent fuel from LWRs can be affected by stress and strain, fission product fuel chemistry, and environmental conditions.

Stress and strain related effects are affected by e.g. cladding material, cladding wall thickness, fuel rod internal pressure and hydrogen content.

Chemistry related effects deal with the chemical interaction of fuel and cladding or the fission products itself. They can be affected by the type of fuel like UO_2 or MOX, by burnup or stoichiometry.

Environmental conditions can be wet or dry storage, and all kinds of off-normal conditions potentially occurring during the storage period.

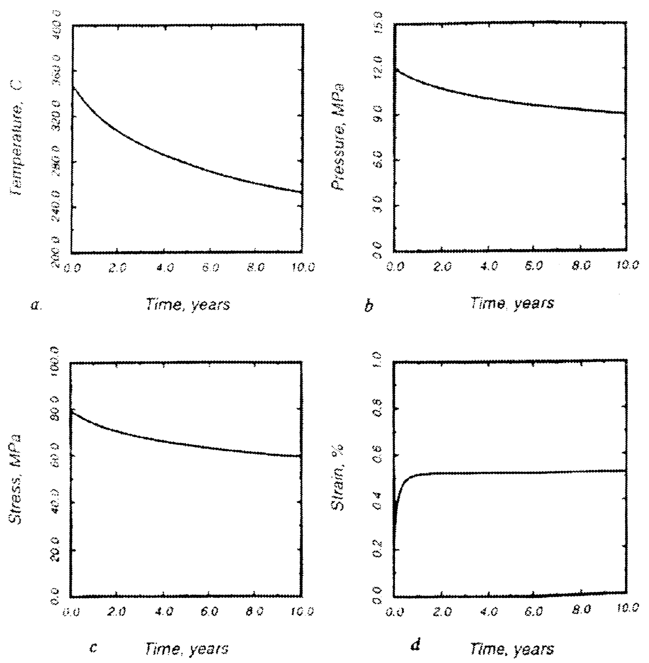
The governing effects depend on the storage type and the environmental conditions, and spent fuel in storage for more than ten years has been analyzed in all countries with nuclear programmes.

In dry storage, initial maximum temperatures of 350 – 370 ^{o}C are licenced under inert gas conditions for the CASTOR V cask. At such temperatures the internal gas pressure is high enough to strain the cladding by thermally induced creep. At these temperatures no fission products are released and therefore, stress-corrosion cracking should not occur. It was also concluded that delayed hydrogen cracking will not occur in dry storage, even at elevated hydrogen contents, because of limited stresses. In summary, stress and strain of the cladding have to be limited to prevent degradation of the cladding (Table XVII). As a basis of current licences an allowable strain level of 1 % was defined together with a stress limit of 120 N/mm^2.

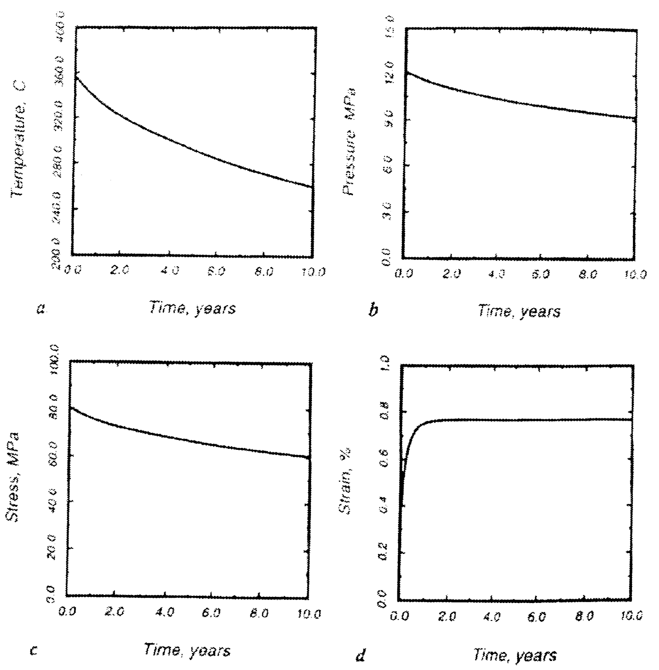
Table XVII. Possible degradation mechanisms of dry storage

1. Stress and strain,
2. Fission product induced stress-corrosion cracking,
3. Delayed hydrogen cracking.

⇒ Stress and strain have to be limited to avoid degradation of the fuel!



UO$_2$-fuel rod with a burnup of 55 GWd/tHM
(DUPLEX cladding)



MOX fuel rod with a burnup of 55 GWd/tHM
(DUPLEX cladding)

FIG. 31. Creep behaviour under dry storage conditions — model calculations.

4.3.3.2. Experiments to support licensing

Two creep investigation programmes were performed in support of cask licensing.

(1) The first programme was carried out on non-irradiated fast creeping cladding types to quantify the creep in an enveloping manner. The data set was chosen with the view of supporting CASTOR V cask licensing for a FA burnup of about 55 MW·d/kg HM. The behaviour of the material with the fastest creep rate was measured and mathematically modelled in order to calculate the total end-of-life (EOL) creep strain for spent uranium or MOX PWR FAs with a fuel rod burnup of 55 MW·d/kg HM. The EOL fission gas pressure determines the hoop stress acting at the beginning of storage. The highest strain after 40 years storage in a CASTOR V/19 cask was calculated to be 0.77 % for a fuel rod with DUPLEX cladding (Figure 31). This value is considerably lower than values at which creep will adversely affect the Zircalloy cladding of LWR spent fuel.

(2) The second programme comprised short time creep and rupture tests on high burnup fuel rods under dry storage conditions. The intention of this programme was to assess the strain potential of irradiated cladding. The tests were performed on corrosion-optimized Zry-4 cladding samples from fuel rods irradiated up to 64 MW·d/kg U (Tab. XVIII). The rod diameters showed a fast irradiation induced creep combined with subsequent slight straining back of the cladding. The oxide layer thickness ranged between 10 to 100 μm. The testing sequence consisted of a creep and a ductility test (Figure 32). The intention of the creep test was to reach a uniform strain of 1.5–2 % within 3–4 days. The tests were performed at two temperature levels of 643 K and 573 K and cladding stresses of about 400 and 600 MPa. The cladding stresses were much higher than those typically found in a stored fuel rod. The creep tests were followed by a low temperature test at 423 K and 100 MPa to assess the cladding ductility with respect to the effect of higher hydrogen content in the cladding (HBU fuel).

The creep tests demonstrated that high burnup rods could reach a uniform plastic strain of 2 % without cladding failure (Figure 33). The low temperature tests at 423 K for up to 5 days revealed no cladding failure under these conditions. Even under the condition of radial hydrogen precipitation (Figure 34) no adverse effect on the cladding performance was found.

Table XVIII. Characteristic data of the irradiated rods for the creep experiment

Manufacturing data:				Irradiation data:		
Cladding outer diameter	10.75	mm		Burnup (rod average)	Rod A 54	Rod B 64 MW·d/kg U
Alloying elements	1.29	w/o	Sn	Neutron fluence (>1 MeV)	9.5	12.1 $\cdot 10^{21}$ n/cm²
	0.22	w/o	Fe			
	0.12	w/o	Cr			
Tensile strength	423	N/mm² (673 K)				

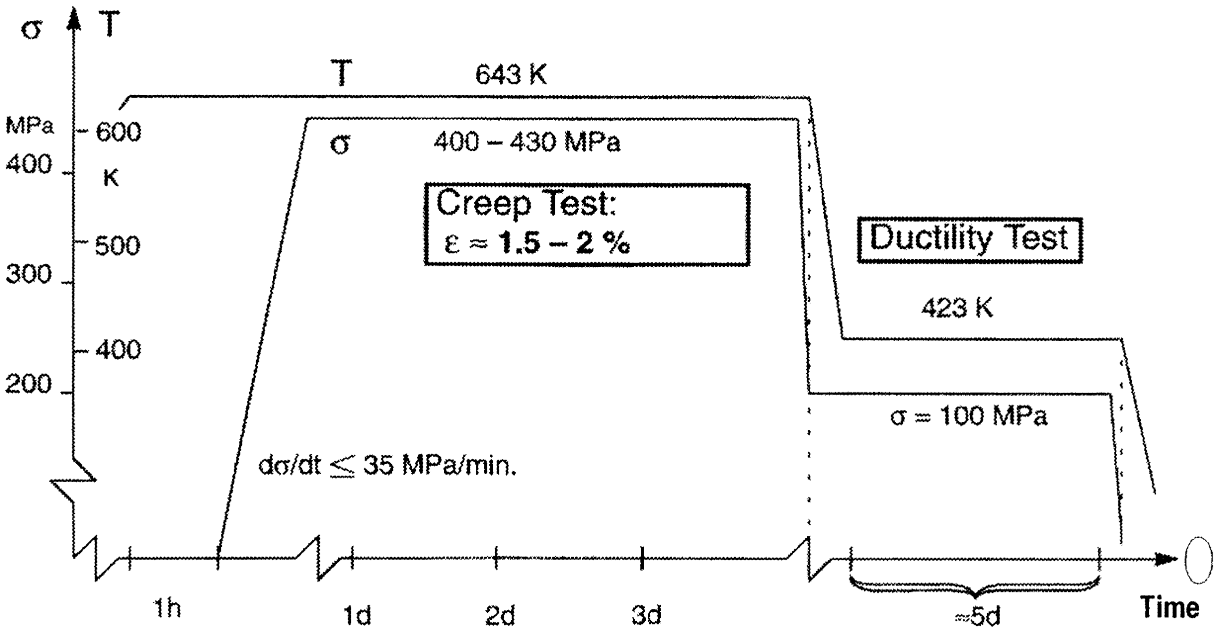


FIG. 32. Testing sequence of short time tests to assess creep and ductile behaviour of high burnup cladding.

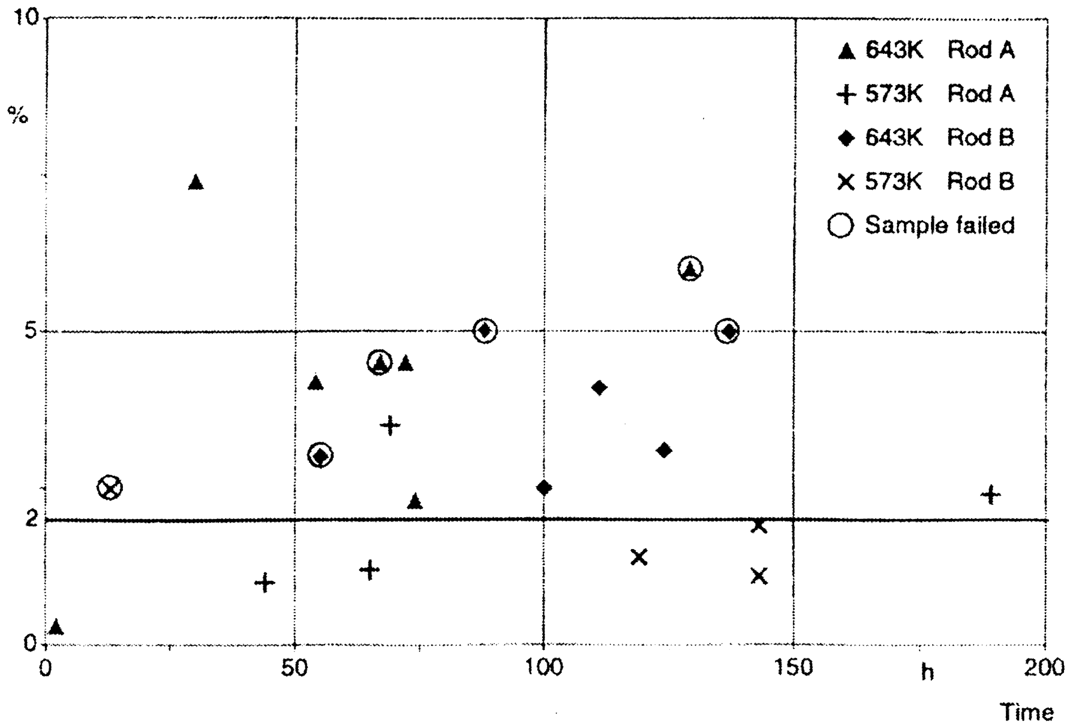


FIG. 33. Creep test results: uniform plastic strain.

4.3.3.3. High burnup fuel and extension of storage periods

The impact of increasing burnup on licensing is shown in Figure 35. At present, a cladding hoop stress of 120 N/mm^2 and a strain of 1 % for storage periods of 40 years or longer were defined as limiting values for cladding integrity. This limits the acceptable maximum EOL pressures to about 18 MPa for a PWR fuel rod and to about 15 MPa for a BWR fuel rod. These pressures have the same order of magnitude as the pressures, which are acceptable during fuel operation, using the so-called non lift-off criterion. Therefore, there are no additional restrictions for dry storage as a consequence of increasing burnup, at the current time.

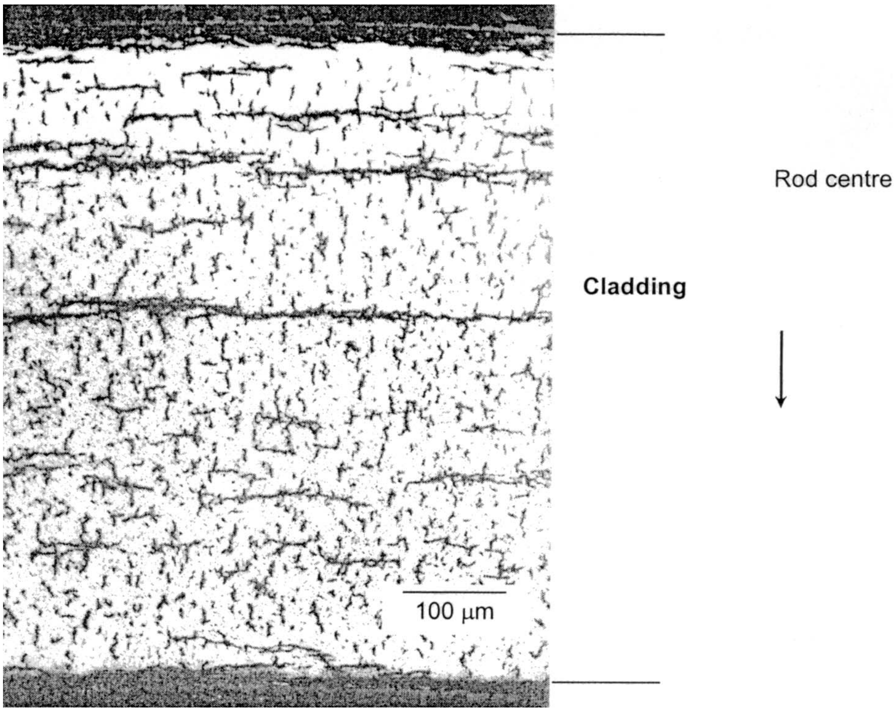


FIG. 34. Metallographic examination of a sample after 573K-creep and 423K-ductility testing.

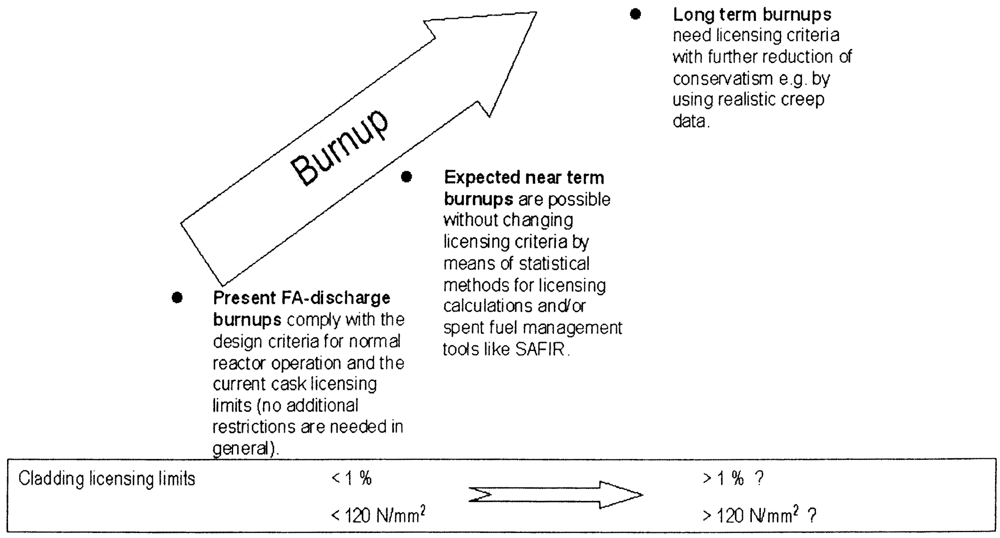


FIG. 35. Impacts of burnup increase on licensing.

Increasing burnup affects both reactor and storage conditions. During reactor operation, the rod internal pressure increases, due to the higher fission gas release. During storage, the higher decay heat increases the cask temperatures, this leads to increased rod internal gas pressure and induces further cladding creep. In short, an increase in burnup would be a driver for greater licensing constraints. Where the increase is moderate, the constraining parameters can be accommodated to some degree by the use of statistical methods and/or optimised cask loading, by using tools such as SAFIR. Under such conditions there is no need to change licensing criteria.

This situation is likely to change by end of this decade, as current stress and especially strain criteria will not be met for some rods if the creep behaviour is based on non-irradiated cladding performance. An example for high duty UO_2 fuel with 5 w/o ^{235}U enrichment is shown in Figure 36. This figure also shows that roughly 10 % of the rods exceed 1 % strain and a few rods are close to 2 %. To remedy this situation a reduction in the degree of conservatism is required, e.g. by using realistic creep data. This means using data obtained from irradiated material placed under temperature and stress conditions of dry storage.

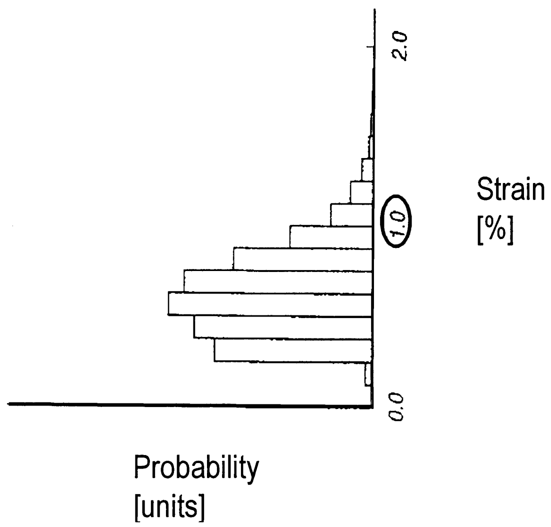


FIG. 36. Probability distribution of post-pile cladding strain after 40 years of storage of high burnup/high duty-rods with 5% ^{235}U (DX, 380 °C).

The most important aspects associated with storage periods of up to 100 years are listed in Table XIX. Storage extension is closely related to the question on the ability to retrieve the fuel assemblies after storage. Presently, the long term behaviour has to be extrapolated from short term reactor operation, under high water temperatures and strong irradiation fields, and from the fuel handling and storage experiences, under cold water and dry conditions.

To assess the behaviour of fuel assemblies for very long storage periods, under dry storage conditions, the experiences have to be combined and attention made to the active processes at moderate storage temperatures.

Table XIX. Aspects associated with storage periods of up to 100 years

Retrievability after storage needs to assess the possible processes in all components of the FA under dry storage conditions.

Possible processes during dry storage:

- Thermally activated processes in the cladding and fuel in relation to very long times, e.g.
  - cladding: creep, hydrogen diffusion, annealing etc.
  - fuel: chemical reactions, gas diffusion etc.
- Irradiation induced processes, e.g. changes of the chemical composition, He-production, etc.

Under dry conditions, residual water is the leading promoter of material changes (corrosion) as hydrogen buildup is absent. Only thermally activated processes are possible. For the cladding creep, hydrogen diffusion and annealing have to be considered.

For storage periods greater than 40 years, creep rate becomes practically zero and the related strain gradation is not an issue, as storage temperatures drops well below 300 ºC. The same is true in principle for cladding annealing, which is most pronounced at the beginning of storage, when temperatures are high.

Under extended storage periods of 100 years, a potential movement of hydrogen is of interest. From reactor operation it is known, that the hydrogen is mostly formed from cladding corrosion, and is taken up to a certain extent by the cladding. In spite of an axial temperature gradient of about 50 K over the active length, the hydrogen distribution remains axially relatively stable during irradiation. This can be understood from the small axial temperature gradient of about 0.1 K/cm compared with the typical depth of diffusion for hydrogen seen in the vicinity of the fuel at 350 ºC. Very long storage periods at temperatures of 200 ºC do not result in much change. The axial gradient becomes even smaller and the depth of diffusion remains in the same order of magnitude, since the decreasing diffusion coefficient is only roughly compensated for over a very long time. A gross change of the axial hydrogen distribution, therefore, is unlikely to occur even during long storage periods, however, a localised diffusion should be possible.

Concerning the fuel inside the cladding, chemical reactions and gas diffusion are dependent on burnup, which determines the amount of new species, and temperature, which determines the reaction rate. During storage the burnup does not change and hence, the production of new species remains small. Additionally the temperatures are relatively low. The conditions for a change of the chemical state are unlikely.

For very long storage periods (hundreds of years) the formation of alphas, i.e. Helium, especially in MOX fuel becomes comparable with the amount of fission gases produced during reactor irradiation. Calculations predict concentrations of up to 0.2 g/kgHM after 200 years of storage. Helium has a smaller atomic radius than the fission gases and therefore diffuses much faster. In literature the diffusion coefficient of He is reported to be far below 10^{-18} cm^2/s under dry storage temperature conditions. Assuming a diffusion coefficient of only 10^{-18} cm^2/s, the typical depth of diffusion after 100 years of storage is around 1/10 of the size of a fuel grain. Helium release via diffusion to the grain surface therefore is practically impossible. On the other hand, in the unlikely event of Helium release there is still a large margin until the cladding reaches stresses that might be of concern with regard to delayed hydrogen cracking.

4.3.3.4. Conclusions

For current and future burnup (in short term) an effective cask licensing methodology, based on the creep behaviour of non-irradiated material, is available. With a move to much higher burnup this methodology becomes restrictive and a new methodology based on the creep behaviour of irradiated material is required.

The extension of the storage periods up to 100 years seems to be possible without impacting on fuel retrievability. For MOX fuel the high Helium production rate in storage is not expected to impact on fuel cladding integrity.

4.4. INDIA

Various types of nuclear fuels including Zircaloy-2 clad natural UO_2 fuels for PHWRs, and enriched UO2 and MOX fuels for BWRs have been examined in hot cells. The fuel elements examined varied in burnup from 2,000 to 29,000 MW·d/t HM and had varied history of wet and dry storage conditions, see Table XX.

Table XX. Irradiated fuel elements subjected to PIE

Type of fuel	Number of Elements		Burnup (MW·d/t HM)	Wet storage (Years)
	Total	Defective		
Experimental Fuel (Natural UO_2)			2,000–7,000	1–10
– PHWR design	8	3+1*		
– BWR design	1	1		
Experimental Fuel (MOX Fuel)			2,000–16,500	1–10
– PHWR design (1.5% Pu)	1	1		
– BWR design (4% Pu)	14	-		
PHWR Fuel (Natural UO_2)	190	8	2,000–14,000	2–10
BWR Fuel (Enriched UO_2)	18	4	5,000–29,000	2–10

* Intentionally defected fuel element

4.4.1. Degradation of storage pool materials

4.4.1.1. Pool liner

Spent fuel storage pools have been in operation at reactor sites and at reprocessing plant sites for more than 30 years. No leakage has been observed in any of the pool liners to date. Ultrasonic technique, however, is currently being developed to enable large scale inspection of pool liner welded regions for surveillance purposes.

4.4.1.2. Examination of spent fuel storage trays

SS 304 trays used for wet storage of PHWR bundles accommodate 11 bundles per tray and are arranged in vertical stacks of 20 trays each. One of the bottom most trays from a central stack was examined to assess the condition after ~10 years of storage under water. Visual examination revealed shallow pitting corrosion on the surface with the pitting density increasing towards the weld regions. The shallow pits were not considered to be performance limiting, as they were not likely to affect the structural integrity of the tray. The trays were considered fit for continued storage of spent fuels. A surveillance programme for corrosion evaluation of the trays is underway.

4.4.1.3. Predictive modelling of Zircaloy-2 corrosion during wet storage

Predictive modelling studies on zircaloy-2 during wet storage showed that the corrosion rates were quite low (Figure 37). In the first 2 years the oxide thickness grows to ~3.7 microns after that the oxidation rate tapers off and no significant corrosion is expected to occur up to a period of several decades.

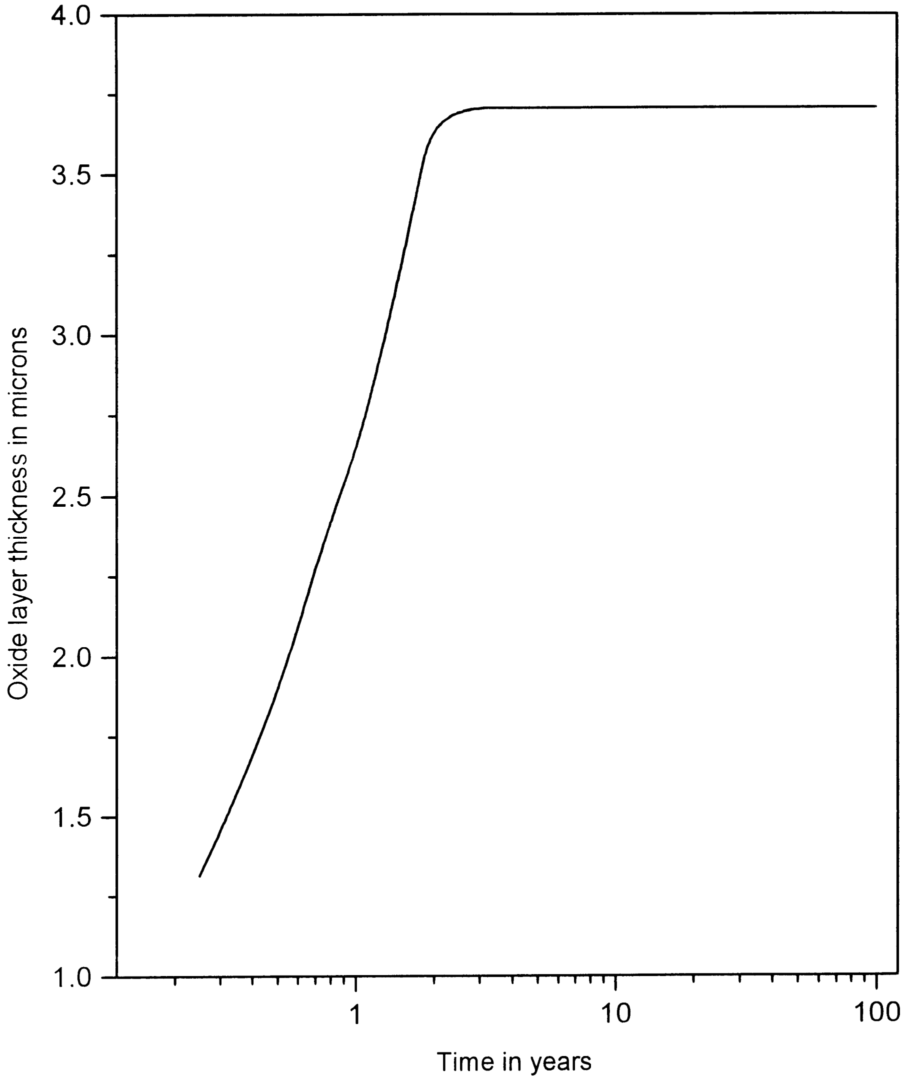


FIG. 37. Zircaloy-2 oxidation rate in spent fuel storage pools, prediction as per model.

4.4.1.4. Microbial fouling

Examination of ion exchange beds of SFSPs has indicated some microbial growth in them. A programme has been initiated to study various aspects of biofouling in spent fuel storage pool materials.

4.4.2. Oxidation of irradiated fuel

Effects of long term air oxidation have been studied on metallographically prepared Zircaloy-2 clad UO_2 fuel sections, stored at ambient temperature for nearly 20 years in hot cell. The temperature dependence of the oxidation rate of UO_2 in air computed from published data is given in Figure 38. As dry storage of spent fuel is carried out after 10 years of wet storage, the oxidation rates are extremely slow at the low temperature levels envisaged in the dry storage conditions. The thin layer of reaction products on the surface was sampled by extraction replica technique. The XRD scan (Figure 39) showed peaks corresponding to $U_{16}O_{37}$.

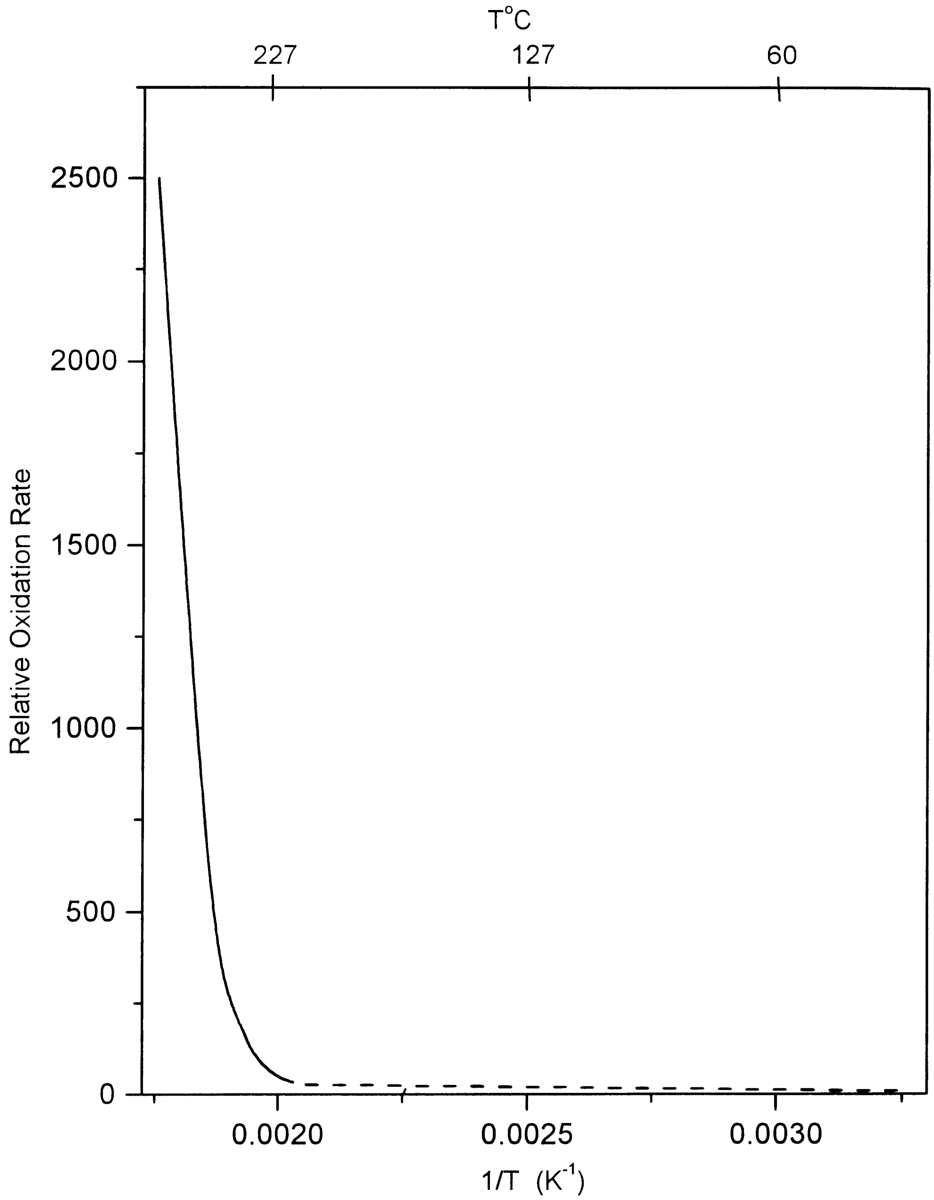


FIG. 38. Variation of oxidation rate of UO_2 in air, expressed on the basis of time taken for clad diameter increase of 2% in defected fuel exposed to air oxidation.

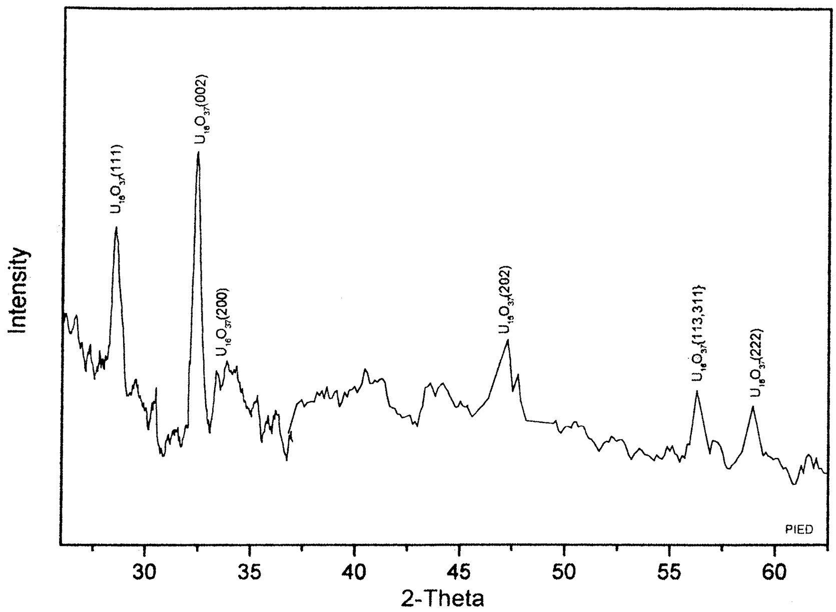


FIG. 39. XRD pattern of oxidation products on the surface of irradiated UO_2 fuel section after exposure to hot cell atmosphere for over 20 years.

4.5. JAPAN

The government and the Electric Power Companies, specifically aimed at the realization of dry storage away from reactor by 2010, have carried out research and development on spent fuel storage.

Two storage programmes were reported:

4.5.1. R&D programme of advanced spent fuel storage technology (1992–1996)

From April 1992 to March 1997, studies of advanced spent fuel storage technology were undertaken at CRIEPI. The 5-year programme related to the dry storage of high-burnup and MOX spent fuels, which are the main objective in the next stage of the storage strategy.

4.5.2. New R&D programme for interim storage of "recycled fuel resources" (1997–2003)

In 1997, a new programme aimed at demonstrating interim storage of "recycle fuel resources" was started. This is mainly based on concrete modular storage technology, such as horizontal concrete silo and a concrete cask.

The results with respect to the studies undertaken on seals, concrete and structural materials are summarised in Chapter 3 — "technical issues".

The new R&D programme also studied high burnup and MOX fuel, the results are summarised below:

4.5.2.1. Characteristics of high burnup and MOX spent fuel

High burnup PWR/BWR and PWR-MOX spent fuel from commercial reactors were used in the study. Table XXI summaries the spent fuel used.

Table XXI. Burnup/enrichment of spent fuel

Fuel type	Rod average burnup	Enrichment/ Pu_{fiss} enrichment
High Burnup PWR	60 GW·d/t HM	3.8
High Burnup BWR	56 GW·d/t HM	3.5
PWR-MOX2	45 GW·d/t HM	3.5

Several post irradiation examinations were undertaken to determine neutron/gamma source data, to support the introduction of burnup credit for interim fuel storage. The following data were obtained:

– Neutron emission and gamma ray source distribution along the fuel rod,
– Gamma ray source distribution in the radial direction of the pellet,
– Chemical isotopic analyses for the nuclide composition of actinides and fission products of the pellet.

The following examinations were also performed to determine fuel characteristics during irradiation and storage:

– SEM/TEM observation,
– Annealing test for fission gas release behaviour,
– EPMA observation of U, Pu, Xe, Cs, and Nd along pellet radius.

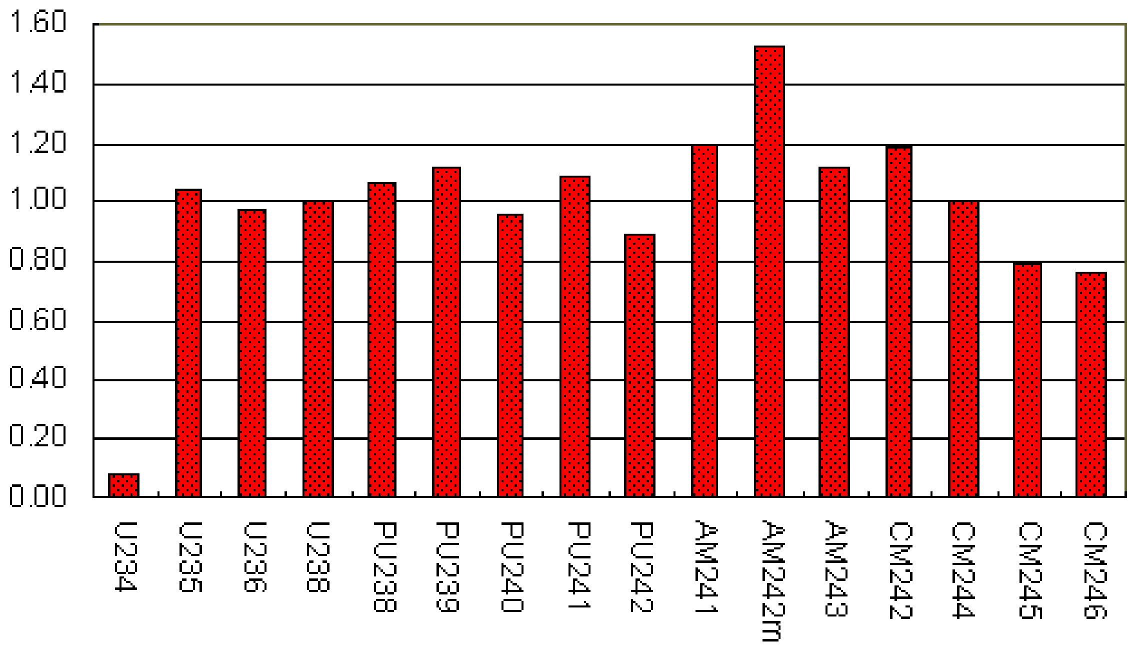


FIG. 40. Chemical analysis and ORIGEN2/82 results for high burnup PWR fuel (65 GW·d/t HM at local point).

Isotopic analyses (chemical) were performed to establish burnup from the amount of heavy metal and ^{148}Nd present. The results were compared with those from ORIGEN2/82 calculations. The C/E (ratio of calculation and experimental value) of actinides is shown in Figure 40 for high burnup PWR fuel. The differences between calculated and experimental isotopic composition were considered in terms of their impact on fuel reactivity when burnup credit is applied.

Pellets from high burnup PWR fuel and PWR-MOX fuel were annealed stepwise to 1700 °C. The fuel temperature under vacuum conditions is expected to rise to 500°C when the cask and/or canister was flooded with helium, during spent fuel-loading operations. The results of testing show that less than 2% of fission gases will be released to the fuel rod interspace during cask loading operations, i.e. up to 500°C. The additional fission gas release has little effect on fuel rod internal pressure.

4.5.2.2. Long term performance of high burnup and MOX spent fuel

BWR spent MOX fuel, burnup ~ 20GW·d/t HM, was used to evaluate the integrity of spent fuel during 20 year wet and dry storage. The following examinations were performed.

– Visual inspection of the fuel pre/post wet storage,
– Puncture test of the fuel pre/post wet storage,
– Atmosphere gas analyses of the fuel pre/post dry storage,
– Ceramographic examinations of the fuel pre/post dry storage.

Preliminary observations show that there is no marked difference before/after storage of these low burnup rods. Further examination, however, may be required to determine their integrity during long term storage.

4.6. REPUBLIC OF KOREA

A computer code for evaluation of spent fuel integrity, SFINEL, has been developed. By using this code the integrity of spent fuels during extremely long term storage periods (up to 100 years) can be predicted.

4.6.1. SFINEL code development

Safe management of spent nuclear fuels is socially, technically, and economically very important in terms of environmental protection and utilization of recyclable resources. One of the most critical parts in the management of spent fuel is to establish a comprehensive monitoring system, which can maintain and confirm the integrity of the spent fuel, at any point in time, until a final treatment/disposal policy can be established. The first step in establishing such a system has been to develop a method for predicting (computing code) fuel cladding integrity based on power history and cladding degradation mechanisms. The scope of the code development project is:

– Literature survey and collection of experimental and in-pile data,
– Domestic and international data: design, operational, and experimental,
– Development of computing code SFINEL,
– Structuring of SFINEL: SPENFIP and SIECO,
– Benchmarking and verification of SFINEL,

- Rod internal pressure,
- Stress rupture, clad oxidation, and stress corrosion cracking,
- Comprehensive benchmarking,
- Development of GUI (Graphical User Interface) of SFINEL,
- GUI with Visual Basics.

4.6.2. Code description

SFINEL (Spent Fuel Integrity EvaLuator) is an integrated computer code for predicting spent fuel rod integrity based on burnup history. This code is the combination of two separately developed programs. One program is SPENFIP (SPENt Fuel rod Internal Pressure evaluator) for determining PWR fuel rod internal pressure, this is based on power histories, and the other is SIECO (Systematic Integrity Evaluation Computer) written by KAERI for estimating the spent fuel cladding integrity during dry storage. The basis of SPENFIP and SIECO programs are GT2R2 (Gapcon Thermal 2 Rev. 2) and DATING (Determining Allowable Temperatures for dry storage of spent fuel in Inert and Nitrogen Gases) respectively, developed by PNL (Pacific Northwest Lab.).

The code requires fuel rod design, power histories, and dry storage-conditions data to be input. The user can then select various options such as fission gas release, clad creep-down, fuel densification, and so on (see Figure 41) to determine their influence.

The output of this code consists of two parts: fuel performance, such as fission gas release and rod internal pressure at EOL (end of cycle) and cumulative damage during 100 years of dry storage (Figure 42).

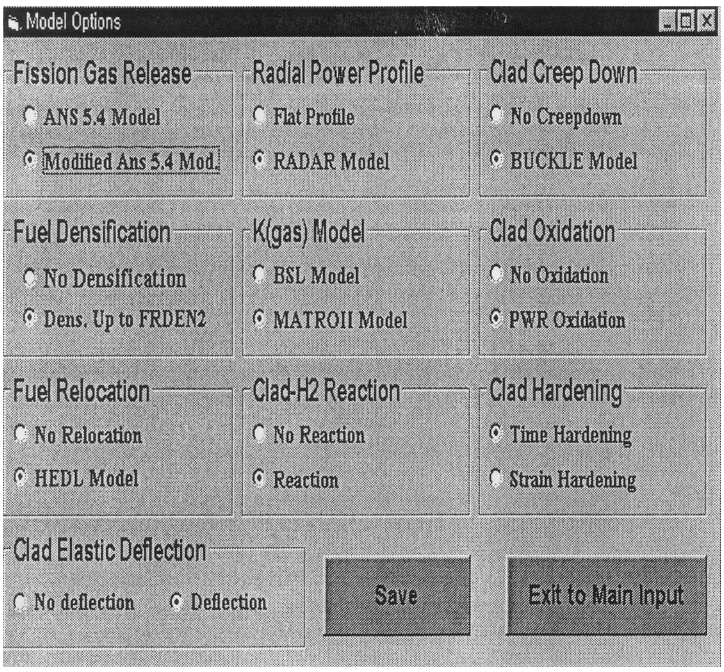


FIG. 41. Model option input.

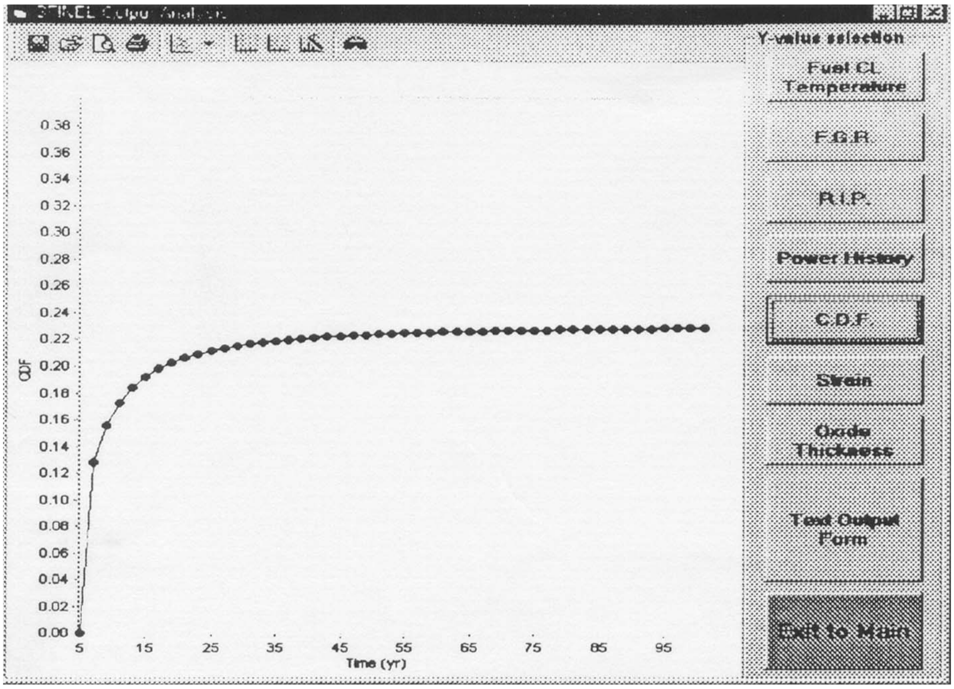


FIG. 42. Cumulative damage fraction vs. storage duration.

4.6.3. Failure analysis of spent fuel during one hundred-year storage

After benchmarking of the SPENFIP and SIECO programs, the SFINEL code was then comprehensively validated.

The computational results from the SPENFIP program were checked against the RISO III data base (M2-2C and PA 29-4) and data from PIE of Korean PWR spent fuel rods. Good agreement between the data was concluded. The output from the SIECO program, written for establishing fuel clad integrity in terms of stress (creep) rupture, cladding oxidation and stress corrosion cracking, was compared with available data. In the case of creep-induced cumulative damage, only a thorough review was carried out because the SIECO code is based on PNL's DATING code, which has already been internationally verified.

The results of cumulative damage from Zircalloy oxidation were compared with available out-of-pile data. Some agreement was found, but it is based on limited experimental data and more data is required for complete verification of the code.

Predicted cumulative damage due to SCC (Stress Corrosion Cracking) was compared with the data currently available from SCC models. Due to uncertainties in the available models and the belief that SCC is unlikely to occur during dry storage of fuel, the SCC formulation was finally dropped from the SFINEL code.

Finally, SFINEL code was comprehensively tested for typical Korean spent fuel rod data. One of the results is shown in Figure 43. In this case the PWR spent fuel rods were dry-stored in the He environment at 380 °C after five years of wet storage at the NPP site. It is predicted that the spent fuel rods can maintain their integrity even after one hundred years of dry storage.

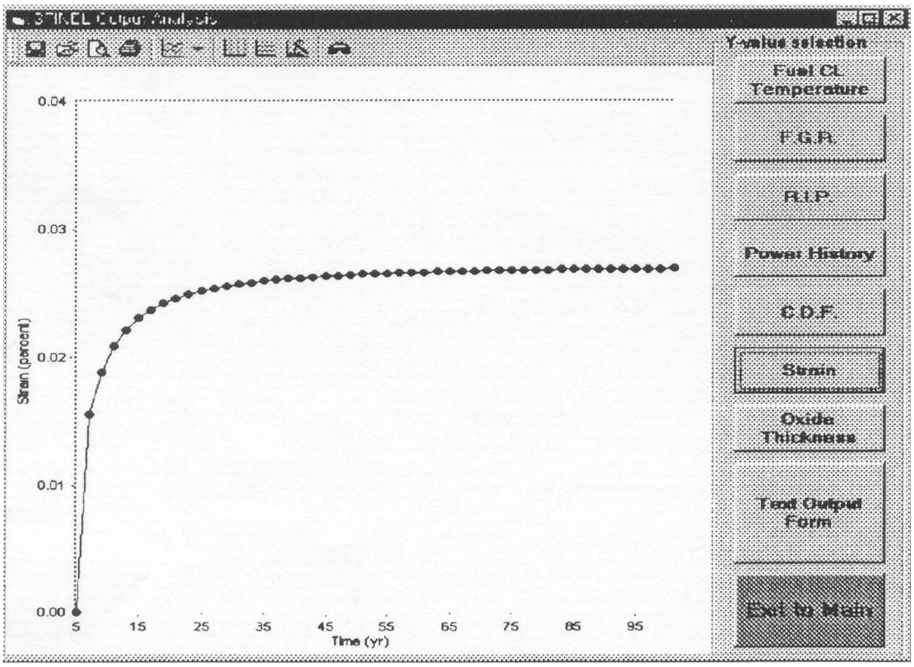


FIG. 43. Strain rate vs. storage duration.

4.7. RUSSIAN FEDERATION

4.7.1. Assessment of RBMK-1000 cladding integrity during dry storage in a dual-purpose metal-concrete cask

The best options for RBMK fuel is to dry store it in a dual-purpose (storage/transportation) metal-concrete cask, MCC. Table XXII shows the characteristics of storage in a MCC.

Table XXII. Terms of storage in MCC

1.	114 halves of RBMK-1000 fuel assemblies
2.	Cooling time – at least 10 years
3.	Residual heat release of each half of FA – 0,04 KW (≈ 5 KW per cask)
4.	Inert gas – argon
5.	Maximum initial temperature of SF - 300°C under normal storage conditions
6.	Temperature of SF during transportation - 350°C
7.	Transient (during 24 hours) temperature increase up to 380°C
8.	Storage period - 50 years

The following criteria is assumed in Russia, as the basis for safe storage (for normal conditions): hoop creep strains of fuel cladding are not to exceed a 1% limit.

On the basis of different initial temperatures (Figure 44), the maximum allowable initial temperature was calculated (Figure 45).

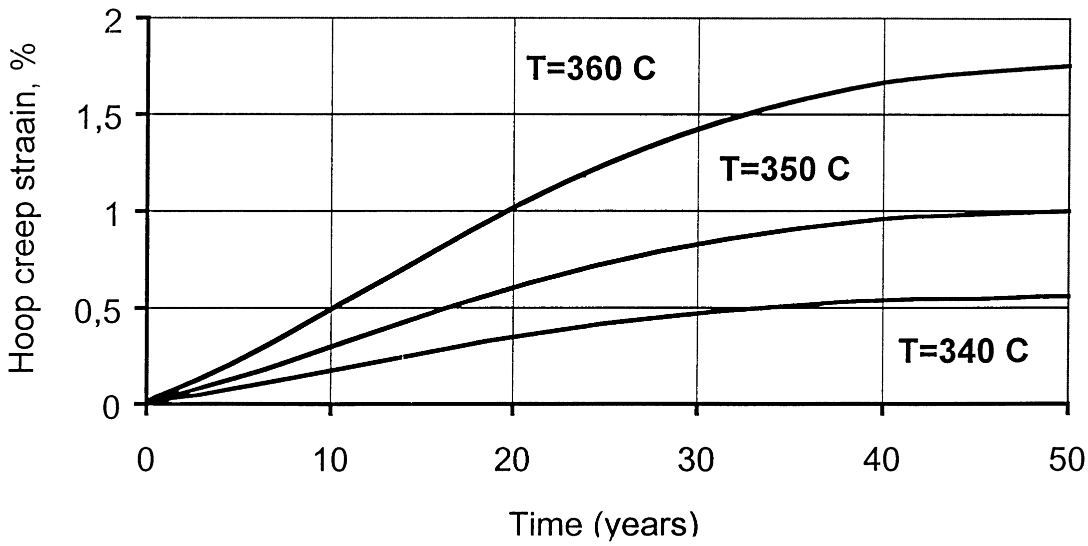


FIG.44. Effect of initial temperature on the accumulation of creep strain.

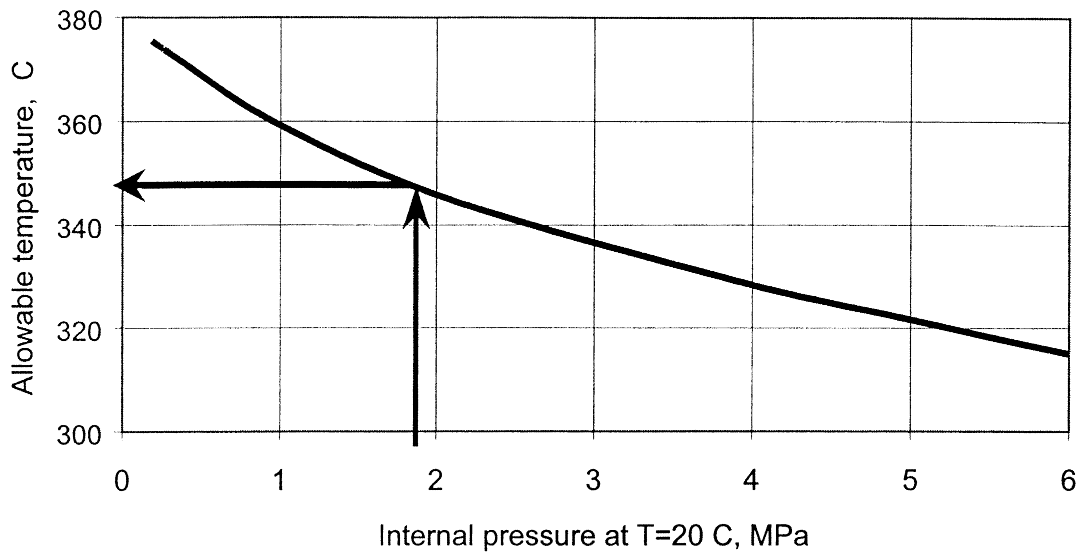


FIG. 45. Maximum allowable temperature of RBMK-1000 spent fuel as a function of internal rod pressure.

4.7.2. Examination results

Over the past 18 years more than 30 RBMK fuel rod fragments, of varying degrees of burnup, have been studied.

The corrosion behaviour of the fuel cladding material was judged from the thickness of uniform oxide coating, nodular corrosion depth and cladding thickness, by optical metallography of sections. This data is summarised in Table XXIII.

The rate of uniform cladding corrosion for wet storage varies between 3–5 μ/year, depending on the burnup and storage time.

The studies show that at a burnup of 1.3 MW·d/kgU the surface of a RBMK-1000 fuel rod can develop nodules up to 40μm deep (outside) and up to 150μm deep (inner) under the spacer grids. As fuel burnup increases to 19.3 MW·d/kg U, these values increase to 130 and 380 μ, respectively. Some fuel claddings, adjacent to the spacer grids, show signs of localised thinning caused by fretting-corrosion. The thinning reaches 400 μm in depth. The appearance of these fuel rods is shown in Figure 46.

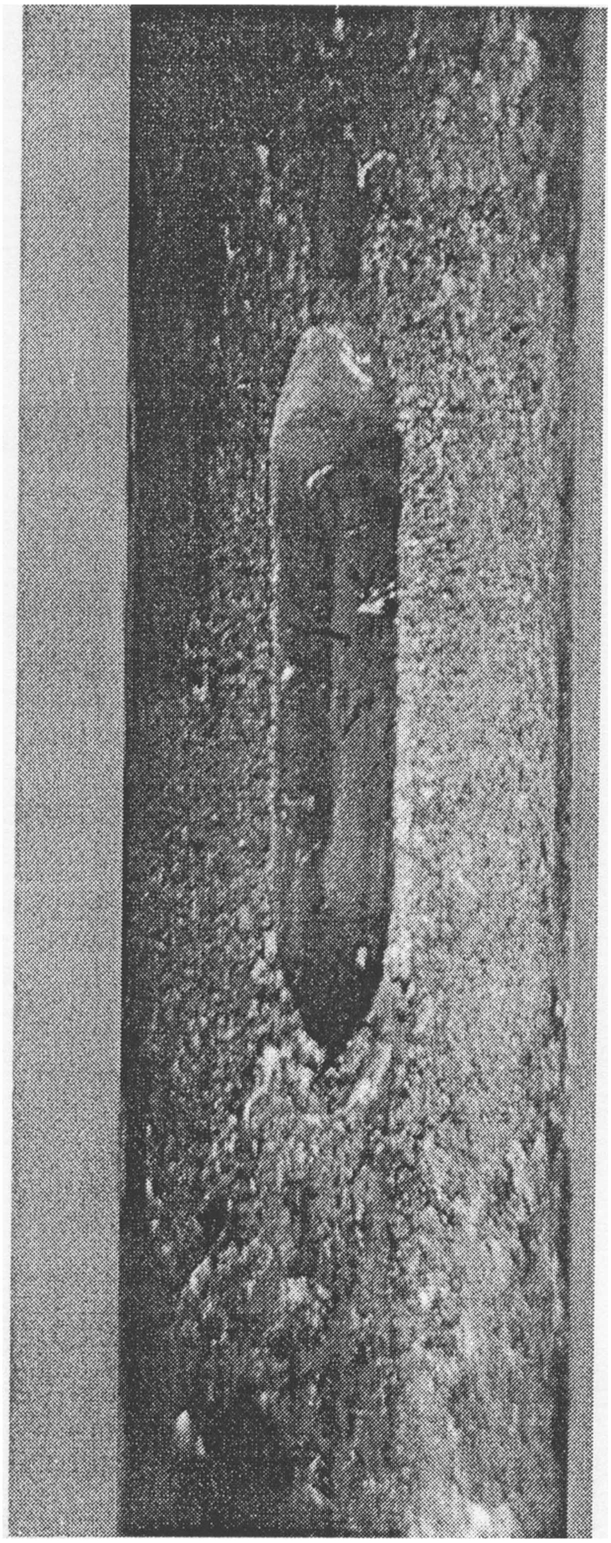


FIG. 46. Appearance of fuel rods in the spacer grid region.

The given nodules' depth and fretting-wear (reaching 400 μm) may substantially limit the mechanical strength of a cladding.

The hydrogen uptake by the fuel cladding material was insignificant, (3–15)*10^{-3} % mass and caused by fuel cladding wear

Table XXIII. Summary of data on corrosion properties of fuel claddings and content of Hydrogen

	Outside SG		Inner SG			
MW·d/kg U	Thickness of oxide coat, μm	Depth of nodules, μm	Thickness of oxide coat, μm	Depth of nodules, μm	Depth of fretting area, μm	Hydrogen content, 10^{-3} % mass
1.3	10–20	0–40	-	0–150	0	3–6
19.3	15–20	0–130	-	0–380	0	3–15
19.5	60–140	-	90–250	0–180	300–400	9–15
22.7	9–65	0–90	30–140	0–190	0	3–5

Fuel rod cladding tensile tests was carriedout at 20 and 3500C on annular samples (3 mm high) and a tensile-testing machine crosshead speed of 3 mm/min. To calculate the strength properties, use was made of the nominal fuel cladding thickness of 0.9 mm. The testing results are summarised in Table XXIV.

The data presented show that in spite of the significant fuel cladding wear, their strength and plastic properties remain at a rather high level.

Table XXIV. Short term tensile properties of fuel claddings

N of SFA	T ^{o}C	σ_B MPa		$\sigma_{0.2}$ MPa		σ_{un} %		σ_{tot} %	
		min	max	min	max	min	max	min	max
1	20	598	608	567	581	1.0	3.5	14	22
1	350	369	429	347	414	1.3	4.9	17	20
2	20	573	608	507	566	2.3	3.2	11.7	12.8
2	350	414	425	350	402	2.2	3.2	14.9	17.6
3	20	563	592	540	562	1.5	4.0	13.1	14.2
3	350	355	397	283	359	2.9	5.0	13.8	16.7

The following degradation mechanisms in fuel rod cladding, SFA and other components in dry storage conditions are to be investigated:

- Interaction of fuel claddings and other components of spent fuel assemblies with the impurities (O, N, H, C etc.) of possible gas environments (argon, nitrogen, air),
- Thermal creep of fuel claddings induced by the internal pressure of transmutation gases and helium, allowing for corrosion damage of the outer cladding surface,
- Delayed hydride cracking (DHC) of fuel rod claddings and other components fabricated from zirconium alloys with a temperature gradient along the length of a product and thermal cycling being present,
- Corrosion cracking of spacer grids and other components of spent fuel assemblies fabricated from austenitic chrome- nickel steels in the gas environment with corrosive impurities (O, I, Cs etc.) and under tensile stresses.

Degradation mechanisms were studied on active SFA structural materials (in hot cells) and on dummy fuel rods.

The initial studies of the impact of internal rod pressure on standard fuel cladding creep rate were made using prefabricated fuel rods heated to 320–600^{0}C in helium for 24 h, or until the loss of tightness by a fuel rod are reported.

The prefabricated fuel rods were manufactured from the full-scale fuel rods of RBMK spent fuel assemblies that had been in operation in Unit 2 of the Leningrad NPP (~ 20 MW·d/kg U) and stored in a cooling pond for 12 years, respectively. The condition of the cladding used in these tests is typical of RBMK fuels that have reached the design burnup.

The prefabricated fuel rods may be hypothetically subdivided into two groups, namely, the ones internally pressurized to 0.5 and 1.0 MPa and the ones pressurised to 3.0 and 5.0 MPa. The value 0.5 MPa is typical for RBMK spent fuel. To conservatively assess the storage safety, fuel rods with high fretting wear (resulting from the interaction between spacer grids) and cladding pressures of 3.0 and 5.0 MPa were tested. These pressures were chosen based on the assumption of the amount of fission gas products released to the fuel cladding interspace, given the chosen storage duration.

The results from short term testing are summarised as follows:

- At 320 and 400^{0}C there is no noticeable evidence of cladding creep, including cladding under spacer grids,
- At 500^{0}C, neither shape change or thermal creep of the fuel rod cladding, internally pressurised to 0.5 MPa, were initially recorded, the fuel rod retained its tightness. After 2 hrs heating at 500^{0}C, significant (~ 16mm in diameter) bulging of the cladding in the top plug area was observed,
- At 600 ^{0}C, the fuel rods, internally pressurised to 0.5 and 1.0 Mpa, demonstrated the classic process of the steady-state creep proceeding at the rates of $(8.9 \pm 0.8)x10^{-4}$ $h^{-1}$and $(1.7 \pm 0.3)x10^{-3}$ h^{-1}, respectively. Figure 47 illustrates the variation in the tangential strain of this fuel rod. During the whole testing time (24 h), the fuel rod remained leak-tight,
- After heating for 13.5 h, the fuel rod, internally pressurized to 1.0 Mpa, lost its tightness, due to a strong local bulging under the spacer grid, where a substantial fretting wear, induced by the interaction between the spacer grid dimples and the cladding was identified,

– The heating of the fuel rods internally pressurised to 3.0 and 5.0 MPa at 500^{0}C resulted in substantial bulges, developed by the cladding in the area of the bottom plug (the first fuel rod) and in the area of the bottom and top plugs (the second fuel rod). After heating for 1.5 h, the fuel rod, pressurised to 5.0 Mpa, lost its tightness.

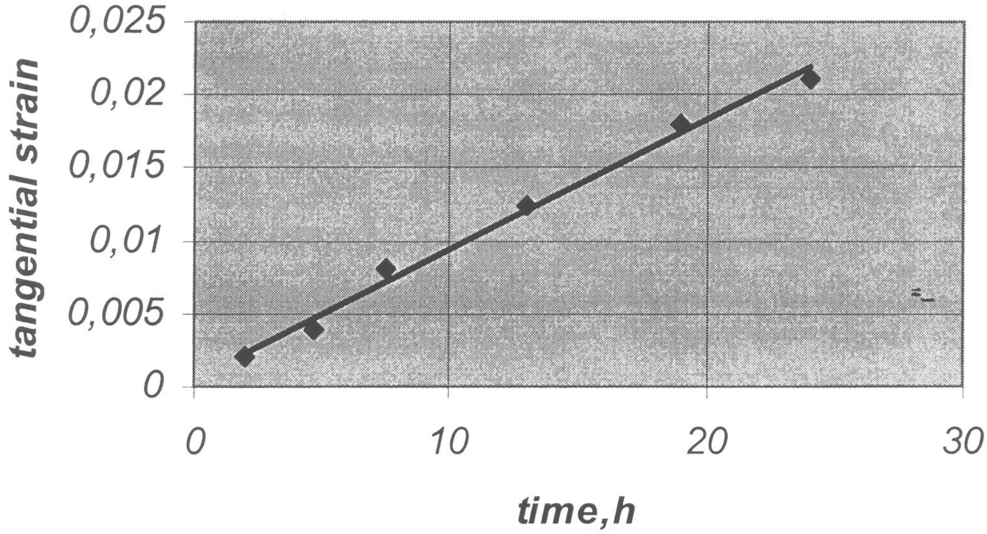


FIG. 47. Variation in cladding tangential strain ε_{φ}.

4.7.3. Conclusion

1. Spent fuel assemblies between 3–15 years cooling have been tested. It allowed conclusions to be drawn on how storage may influence changes in SFA material properties:
   – Long term stored fuel rods were established to be in a similar condition to those prior to storage,
   – During wet storage of spent FAs in cans, their general corrosion is observed to proceed at the rate of 3–5 µm/year.

2. The primary investigations of thermal creep rates of short term tested RBMK-1000 prefabricated fuels reveal the following:
   – The most susceptible sections of RBMK fuel claddings are located under spacer grids,
   – For transient and emergency conditions, the ultimately tolerable temperature of dry inert gas storage is in the range of 450–550^{0}C,
   – To assess the thermal creep rate at lower temperatures (300–400^{0}C), long term (several months) tests have to be implemented.

4.8. UNITED KINGDOM

The pools at Sellafield used for the interim storage of AGR fuel prior to reprocessing have been dosed with sodium hydroxide since 1986 as an inhibitor for intergranular corrosion. The concentration of hydroxide required for inhibition was specified on the basis of electrochemical polarization tests carried out on active AGR brace samples at Harwell Laboratories. Those tests, however, appeared to indicate that hydroxide dosing promoted an additional anodic reaction on the metal, which was interpreted as being the slow transpassive dissolution of the chromium-rich passivating oxide film. Hence, it was concluded that hydroxide dosing, whilst beneficial in preventing perforation by intergranular corrosion, would act to increase the general corrosion rate of the cladding. The issue, however, wasn't investigated further as the envisaged storage durations prior to reprocessing were considered to be relatively short.

With a proposed increase in storage duration to 80 years, both the general corrosion rate of 20/25 Stainless Steel AGR cladding in hydroxide dosed water, and the original Harwell assumptions have been re-examined.

4.8.1. Review of Harwell work

Figure 48 shows a series of anodic polarization curves obtained for a sample of reactor-irradiated brace in 10 ppm Cl^- with various OH^- additions. Similar curves were also obtained in 1 and 5 ppm Cl^- solutions. The test solutions also contained ~ 1 ppm hydrogen peroxide (H_2O_2) added to simulate solution radiolysis. The data shown in Figure 48 were thus taken to indicate that at 10 ppm Cl^-, 400 ppm OH^- (i.e. pH 12.4) was not sufficient to inhibit intergranular corrosion, whereas 500 ppm and above were.

It was speculated that the hydroxide-induced anodic reaction represented transpassive dissolution of the chromium-rich oxide film which passivates stainless steels:

$$Cr_2O_3 + 5H_2O \rightarrow 2CrO_4^{2-} + 10H^+ + 6e^- \quad (1)$$

Such transpassive dissolution is well known in strongly oxidising acidic solutions (e.g. hot nitric acid), but is not normally considered in neutral and alkaline electrolytes, although it is in principle possible from a thermodynamic perspective.

Table XXV lists the reversible potential for reaction (1) as a function of pH in caustic solutions.

Table XXV. Calculated reversible potentials for various reactions in caustic solutions at 25°C

[OH^-] (ppm)	**pH**	**E_{corr} (mV_{SCE})**	**E_o (mV_{SCE})**		
			Cr_2O_3/CrO_4^{2-} a	**H_2O/O_2 b**	**H_2O_2/O_2 c**
34	11.3	-	-162	315	-96
68	11.6	-	-192	297	-114
136	11.9	-	-221	280	-132
200	12.1	-310	-238	270	-144
400	12.4	-160	-268	252	-162
500	12.5	-30	-277	246	-168
600	12.6	-40	-285	241	-174

a E_o (V_{SCE}) = 1.069 – 0.0985pH + 0.0197log[CrO_4^{2-}] [3], assuming 1 μM CrO_4^{2-}

b E_o (V_{SCE}) = 0.983 – 0.0591pH + 0.0147logpO_2 [4], assuming pO_2 = 1 atm

c E_o (V_{SCE}) = 0.437 – 0.0591pH + 0.0295log(pO_2/[H_2O_2]) [5], assuming pO_2 = 1 atm and [H_2O_2] = 30 μM

Also shown are estimates of the metal's free corrosion potential (obtained by rough extrapolation of the curves plotted in Figure 48), along with the reversible potentials for two solution oxidation reactions, oxygen evolution from water (2), and oxygen evolution from hydrogen peroxide (3):

$$2H_2O \rightarrow O_2 + 4H^+ + 4e^- \quad (2)$$

$$H_2O_2 \rightarrow O_2 + 2H^+ + 2e^- \quad (3)$$

Table XXV indicates that the transpassive dissolution of Cr_2O_3 is not in principle possible at the metal's free corrosion potential in 200 ppm OH^- solution, but becomes possible at higher OH^- concentration. Oxygen evolution from water is not thermodynamically possible at the metal's corrosion potential in any of the solutions considered. Oxygen evolution from hydrogen peroxide, however, would appear possible at the metal's corrosion potential (and above) for 400 ppm OH^- and more caustic solutions. Thus, on the basis of these data, it is considered that the appreciable anodic currents shown in Figure 48 could represent the oxidation of hydrogen peroxide (which was included in the test solutions to simulate the effects of solution radiolysis), or the speculated transpassive dissolution reaction. In order to differentiate between the two potentiodynamic reactions, potentiostatic and weight loss measurements were carried out.

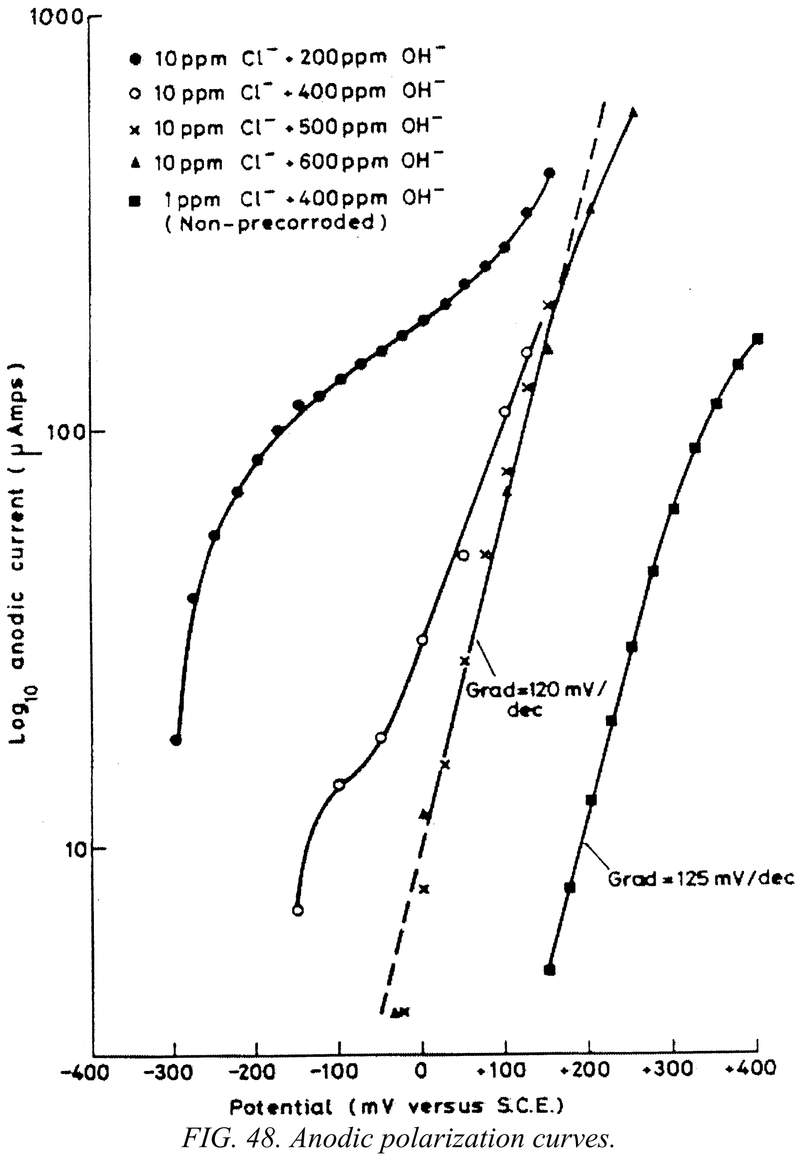


FIG. 48. Anodic polarization curves.

Figure 49 shows the effect of 2 ppm H_2O_2 on the potentiodynamic polarization behaviour of as-received (i.e. ultrasonically cleaned in methanol) 20/25 cladding in 10 ppm Cl^-/pH 12.5 solution at 30 ºC. Figure 49 also includes the relevant Harwell data for comparison. It is clear that H_2O_2 acts not only to increase the metal's corrosion potential, but also has a pronounced effect on the anodic current density.

Extrapolation of the anodic Tafel line, clearly evident in the absence of hydrogen peroxide, to the metal's corrosion potential yields a corrosion current density of ~ 0.01 μA/cm^2. This corresponds to a uniform penetration rate of ~ 0.08 μm/yr. Whilst such measurement of corrosion rate is unreliable owing to the nature of potentiodynamic testing, it does suggest that 20/25 stainless steel is fully passive in pH 12.5 solution, corroding at a negligibly low rate.

In order to determine whether the high current densities measured in the presence of H_2O_2 (see Figure 49) contained an appreciable contribution from the corrosion of the metal (i.e. transpassive dissolution), an electrode made from as-received 20/25 cladding was weighed and then potentiostatically polarized. When the test was terminated, the electrode was then washed in water, rinsed in methanol and dried prior to weighing. A total of four separate weightings were made. All four weights were identical to those recorded prior to the test. Given the cumulative charged passed on the electrode surface area, the charge being wholly attributable to transpassive corrosion, then weight loss amounting to ~ 1.4 mg would have been expected.

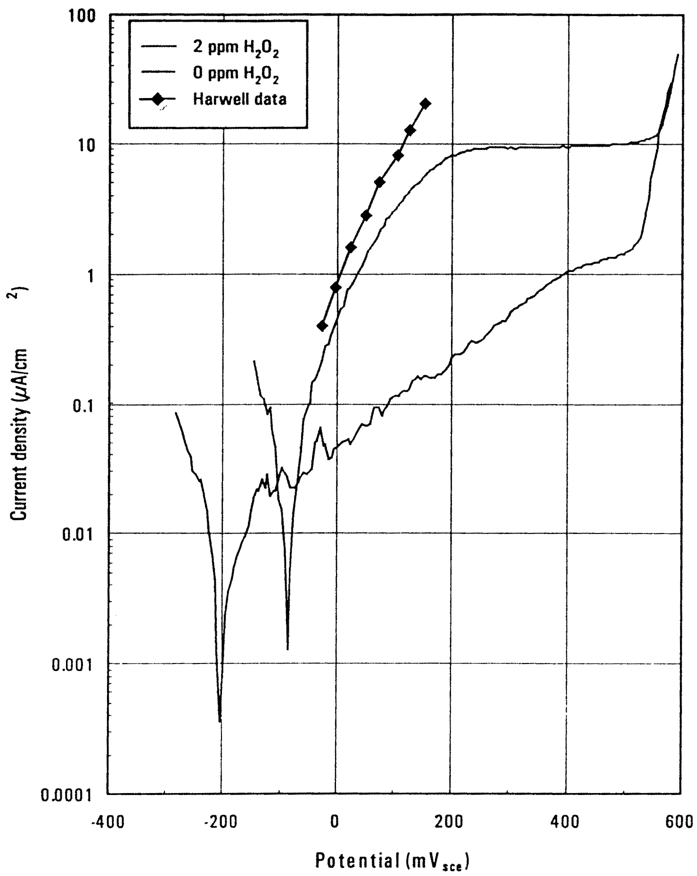


FIG. 49. Polarization curves for 20/25 cladding.

It therefore can be concluded that the vast majority of the recorded current flow shown in Figures 48 and 49 is due to a solution oxidation reaction associated with hydrogen peroxide (most probably its oxidation to oxygen gas), and not metal corrosion.

Additional studies were also undertaken to investigate whether reactor-irradiated material is different from un-irradiated metal, and the influence of galvanic coupling of graphite.

The difference between reactor-irradiated and un-irradiated material was found to be minimal, therefore the use of un-irradiated cladding is justified for all studies. In the case of galvanic coupling of graphite to stainless steel, the graphite was found to act to depress the potential of stainless steel, suggesting that the reduction of hydrogen peroxide proceeds more readily on stainless steel than it does on graphite. It is accordingly considered that a worst case is provided by uncoupled 20/25 stainless steel in hydroxide-dosed water containing hydrogen peroxide.

In summary, it is considered that the evidence presented demonstrates unequivocally that the vast majority of the anodic current passed on polarization of 20/25 stainless steel in OH^- solution containing hydrogen peroxide is not due to corrosion. Rather, it is due to oxidation of the hydrogen peroxide, presumably to oxygen gas. The apparent association of the hydrogen peroxide reaction with OH^- is due to the fact that it becomes possible at the metal's corrosion potential as the pH of the solution is raised owing to the dependence of the former on pH (see Table XXV).

4.8.2. Long term corrosion tests

Long term corrosion tests have been performed using the techniques of Field Signature Method (FSM) and the more traditional technique of Immersion Testing.

4.8.2.1. Field signature method

An instrumented AGR pin was set up with thermostatically controlled solution of caustic being passed through it. After 3 months operation the data collected was interpreted as showing a corrosion rate of 0.5–1 μm/yr. At this point the experiment was terminated and the instrumented pin weighed. The experiment was re-established and run for a further 83 days before being terminated again. Although data interpretation was again in the 0.5–1 μm/yr range re-weighing the pin showed a zero weight loss compared with a potential 0.0074–0.0147g that should have been seen.

A re-evaluation of the technique has been carried out and a problem associated with drift of the reference probe signal identified and rectified. The data now being obtained by this technique is in agreement with immersion test data.

4.8.2.2. Immersion tests

Immersion tests were primarily initiated to support the FSM observations.

Given the results at present, it is not possible to resolve a corrosion rate. It is noted however, that a corrosion rate of 0.1 μm/yr would generate a weight loss of 0.3 mg over the 6-week test period to date. Since no evidence for such a loss has been found, it is tentatively concluded that the corrosion rate is less than this. Such a low corrosion rate is consistent with those rates obtained from investigations carried out into the lifetime of ILW drums, which ranged from 0.03–0.10 μm/yr for 304L and 316L stainless steels in the pH range 9–13.5. This indicates that the chromia-rich film, which protects stainless steel, is stable in alkaline environments, thus ensuring that the metal is fully passivated.

4.8.3. Conclusions

The general corrosion rate of 20/25 stainless steel in 3 ppm Cl^- at pH 11.5 and 12.5 is very low. On the basis of testing carried out to date, it appears to be < 0.1 μm/yr.

The original package of work in this area (Harwell Labs) appears to be flawed. The current package of work has shown that the transpassive corrosion of chromium under current AGR storage conditions does not occur.

4.9. UKRAINE

As interim dry storage facilities will be commissioned until 2010, Ukraine and Russia have agreed a joint programme for the testing of fuel rods and FA fragments at RIAR (Russia).

The research will be undertaken on FAs with maximum and varying burnup. The characteristics and results from three Zaporizhzhya spent FAs (initial enrichment of 4.4 w/o) are shown in Table XXVI.

Table XXVI. Summary of irradiated fuel assemblies 0325, 0328 and 0329 (examination results)

	Parameter	FA 0325	FA 0328, 0329
1.	Average FA - “hottest” fuel rod, burnup MW·d/kg U	48.9/51.3	44.0/46.5
2.	Fuel rod parameters:		
2.1.	Elongation, mm	12.3–18.6	1.0–17.3
2.2.	Decrease in diameter, mm	0.05–0.08	0.04–0.07
2.3.	Fuel-cladding gap, μm	3–43	no data
2.4.	Fuel rod plenum, cm^3	31.5–33.6	31.0–34.0
2.5.	FGR, %	0.19–2.50	0.52–1.94
2.6.	Pressure of gas, MPa	2.46–2.72	2.39–2.64
3.	Maximum thickness of oxide film, μm		
	– outside	7–10	5–7
	– inside	5–10	5–10
	– Hydrogen content in the cladding, %	$(0.88–1.6)10^{-2}$	$(0.3–1.2)10^{-2}$
4.	Fuel pellet parameters		
4.1.	Central hole diameter, mm	2.3–2.4	2.3–2.4
4.2.	Average grain size, μm	3.8–5.8	~ 10
4.3.	Fuel rod density, g/ cm^3	10.21–10.52	10.39–10.47

The following measurements are planned:

(1) After each year of exposure pin-cases containing the FRs are opened. The radiochemical and chemical analysis of the pin-case environment are carried out. FRs are extracted for non-destructive tests (NDTs), this includes:
- Optical inspection including a photography of the cladding defects and anomalies,
- Foucault [eddy] current testing of fuel cladding, with the purpose of analysis,
- Determination of a corrosion layer thickness and detection of fuel claddings cracks,
- On two directions fuel rod measurement profile (shape),
- Fuel rod length and diameter measurement,
- Measurement of ^{85}Kr in a gas collector by gamma ray spectrometry,
- Axial gamma scanning,

(2) Pin-cases containing FA frame fragments are opened annually and visually inspected. This inspection (by photography of the defects), is initially carried out in areas of heterogeneous parts connection:
- Guide tubes — spacer grids,
- Fuel rod — spacer grids.

(3) Destructive tests (DTs) are carried out after 3, 5, 7 and 8,5 years of exposure (storage) and include:
- Metallography of the fuel cladding and welded seams,
- Metallography of the guide tubes and spacer grids fragments,
- Clad hydrogen analysis,
- Mechanical tests of FR and FA structure specimens,
- The radiochemical and chemical analysis of corrosion deposit on FR and FA structure specimen surface,
- Measurement of a corrosion film thickness on both external and internal fuel cladding surfaces,
- Evaluation of fuel pellet condition (ceramografy) for both hermetic (tight) and non-hermetic (failed) fuel rods,
- Measurement of gas pressure under fuel cladding.

4.10. UNITED STATES OF AMERICA

4.10.1. Effect of water on waste form and packages

Considerable progress has been made in characterization of the Yucca Mountain site as a potential site for a geologic repository. A key attribute is the design flexibility to permit the repository to be kept open, with only routine maintenance, from approximately 50 to 125 years from the start of waste emplacement. However, the design will not preclude keeping the repository open for 300 years with appropriate maintenance and monitoring. If licenced as a geologic repository, Yucca Mountain could serve as an extended dry storage facility for several centuries. Research is underway in several fields. For purposes of this TCM, research concerning the effect of water on the waste package is of importance. This research has resulted in an evolution of the waste package design over the past 20 years, see Figs 50 to 52.

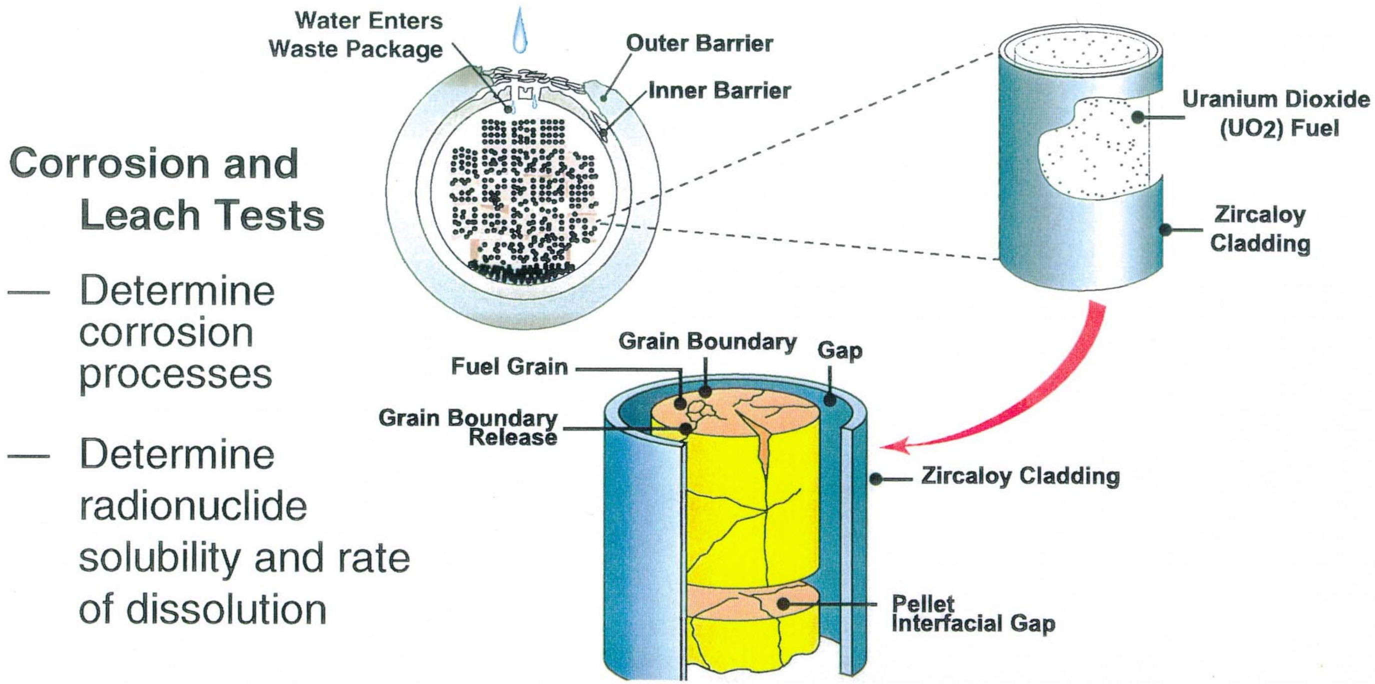


FIG. 50. Effect of water on the waste form and packages.

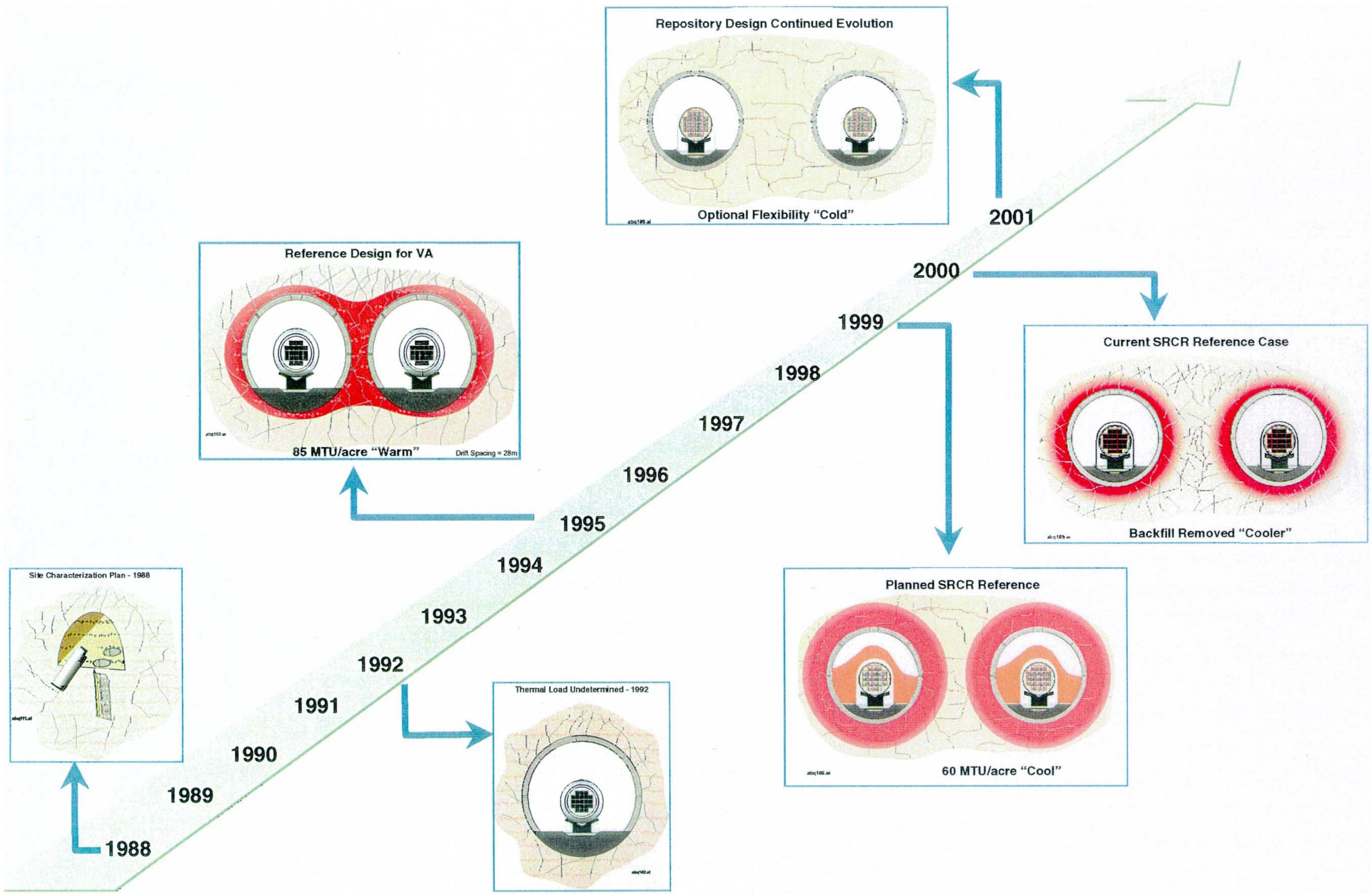


FIG. 51. Evolution of repository and waste package design.

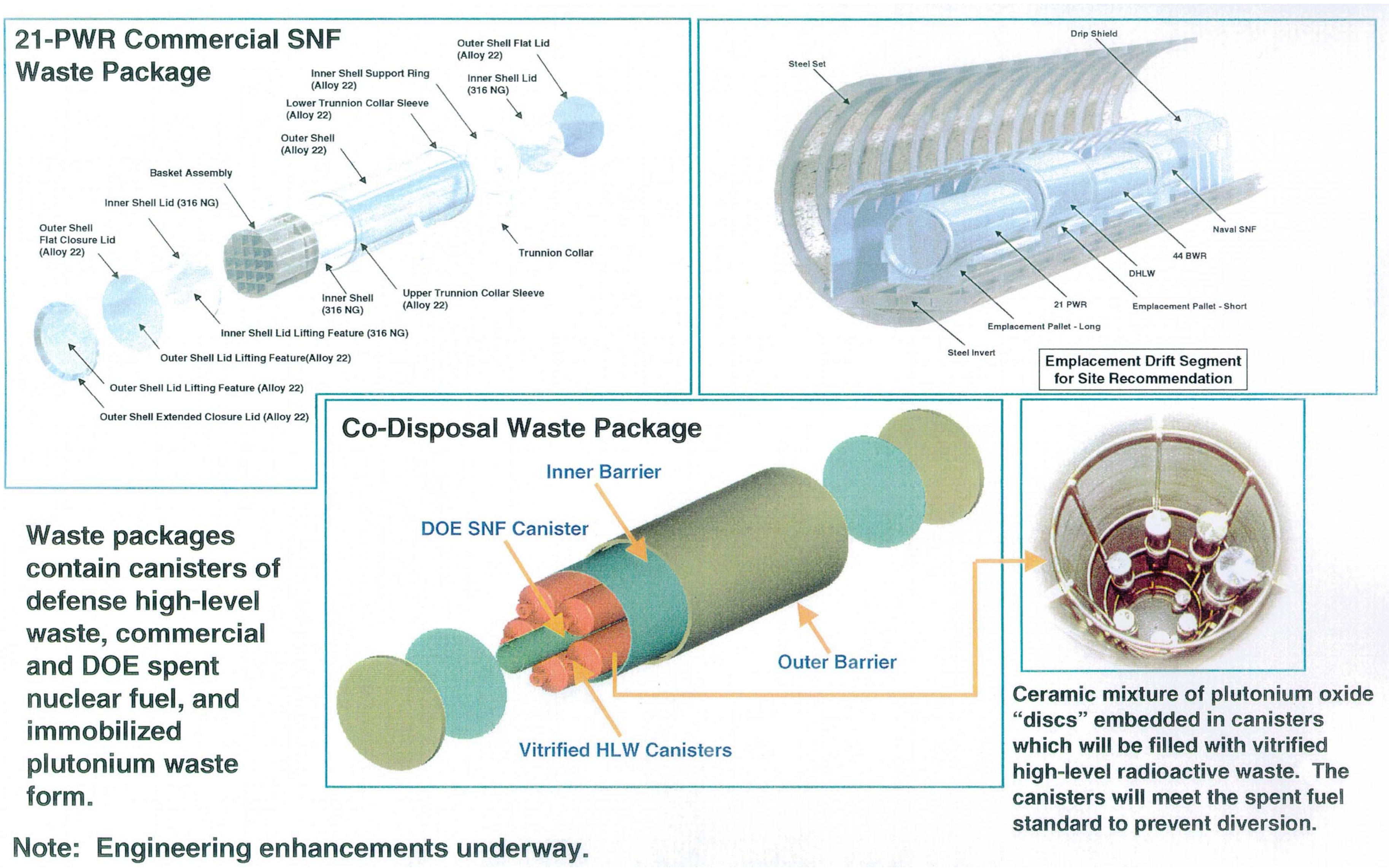


FIG.52. Reference waste package design concept.

4.10.2. Long term storage

Because of near term licence renewal requirements and the potential for extended storage of spent nuclear fuel, there has been a renewed interest by government and industry in research to obtain technical data from dry spent fuel storage cask SSCs that have been in service since the mid-1980s.

DOE, NRC and EPRI have initiated a research programme establish the technical basis to support renewals of licences and Certificates of Compliance for dry spent fuel storage systems and high level radioactive waste at ISFSI sites. These renewals could cover periods of 20 to 100 years. The program, "A Co-Operative Research Program on Dry Cask Storage Characterization", is a logical extension of performance tests and demonstrations that were conducted by DOE between 1984 and 1991 and ongoing dry storage cask monitoring work at the DOE Idaho National Engineering and Environmental Laboratory (INEEL), see Table XXVII.

Table XXVII. Dry cask storage characterization programme

Two phased Research Programme	
Phase one:	Visual inspection of the Castor V/21 & VSC-17 dry storage casks and their contents at the Idaho National Engineering and Environmental Laboratory
Phase two:	Detailed evaluation and characterization of spent nuclear fuel rods at the Argonne National Laboratory-West
	Participants will evaluate data independently.

Spent fuel integrity in four dry storage casks at INEEL was also examined through enhanced monitoring and analysis of cask cover gas samples and the results reported in June 1997.
The current program involves two casks, i.e., the Gesellschaft fuer Nuklear Service Castor - V/21 nodular cast iron PWR spent fuel storage cask and Pacific Sierra Nuclear (BNFL Fuel Solutions) VSC-17 ventilated concrete storage cask configured for PWR spent fuel. The Castor V/21 contains spent fuel assemblies that were discharged between 1981–1983 from the Virginia Power Surry reactor and the spent fuel has been undisturbed since 1985. The VSC-17 contains canisters of consolidated spent fuel that were discharged between 1976–1981 from the Florida Power and Light Turkey Point, and Virginia Power Surry reactors. The cask contains the equivalent of 34 assemblies. The temperature of each cask was above 300°C at the time of loading.

Task 1 (Table XXVIII) of the co-operative program involves: moving the two casks from their storage pad to the INEEL Test Area North Hot Facility, obtaining temperatures of cask exterior surfaces and performing a radiation survey, video and photographic inspection of the casks, seals, selected fuel assemblies or canisters and fuel rods, and, returning the casks to the storage pad. Results from visual inspections already completed of the Castor V/21 cask were very favourable. The spent fuel assemblies and basket were intact and clean and there were no visual signs of degradation.

Table XXVIII. Phase 1 tasks

TASK	CASTOR V/21	VSC-17
Inspection of Cask and Internals	X	X
Exterior temperature readings	X	X
Radiation survey	X	X
Gas sampling of cask interior prior to opening	X	X
Video/photographic inspection of cask support pad	X	X
Video/photographic inspection of cask exterior	X	X
Video/photographic inspection of cask interior	X	X
Video/photographic inspection of cask seals	X	X
Mechanical property evaluation of the cask seal	X	X
General video/photographic inspection of all fuel assemblies, detailed evaluation of up to five fuel assemblies	X	
Video/photographic inspection of 12 fuel rods from the designated assembly	T-11	
Analyze smud and smear samples	X	X
Installation of thermocouple lances and long term monitoring of cask internal temperature	X	X
Installation of gamma and neutron detection devices and long term monitoring of radiation levels	X	X

Phase 2 will involve non-destructive, destructive, and mechanical examinations of spent fuel elements from the casks (Table XXIX). This will provide quantitative and qualitative information concerning the integrity of the spent fuel.

Table XXIX. Phase 2 tasks

Phase Two Items Examinations and Characterizations
Profilometry
Fission gas release
Metallography
Cladding Hydrogen analysis
Creep tests
Cladding stress rupture tests
Cladding tensile tests
Transmission Electron Microscopy

Examples of the types of information that will be obtained include: in situ creep, percentage of fission gas release, internal rod pressure, oxide thickness, hydride morphology and orientation, residual cladding thickness, cladding microstructure, hydrogen content, creep rates, breakaway temperature, tensile strengths, and ductility. Work is underway to load a cask at the INEEL Test Area North complex with spent fuel elements for an onsite shipment to Argonne National Laboratory-West, where examinations will be performed. A report of findings is expected in 2002.

5. REGULATORY ISSUES

The following contributions concerning the regulatory aspects appeared in the presented papers:

5.1. BELGIUM

The spent fuel concept considers either direct disposal in geological repository or reprocessing of spent fuel and recycling as MOX fuel. In between, interim storage is made whatever the chosen solution (if any). Licences for interim storage periods between 10 and 50 years have been obtained.

Interim storage must satisfy four essential conditions:

1. Maintain subcriticality,
2. Ensure that radiation rates and doses do not exceed acceptable limits,
3. Prevent the release of radioactive material above acceptable limits,
4. Maintain retrievability of the stored radioactive material.

Conditions 1 and 2 are satisfied and guaranteed over the storage period by appropriate design. Conditions 3 and 4 can strictly be realized for ”short term” storage, but additional investigations are necessary for extended storage periods (100 years and above). This additional validation is obtained mainly by surveillance programmes and by parallel R and D in relation to fuel rod performance.

Licensing for storage up to 50 years — covering both wet and dry storage — was obtained on the basis of a large number of non-destructive examinations of fuel rods and extended thermo-mechanical calculations. A reduced number of destructive examinations have been made, essentially devoted to defective fuel oxidation in air and performed on low burnup (10 to 30 GW d/t HM) UO_2 fuel with 5 to 30 years storage time. No examination has been made so far on MOX fuel and on high burnup fuels.

Above 50 years (100–250 years) storage time, tentative licensing is based on extrapolations.

5.2. BULGARIA

The Inspectorate of the Safe Use of Atomic Energy (ISUAE) of the Committee for the Use of Atomic Energy for Peaceful Purposes (CUAEPP) performs the state control over the safe use of atomic energy during transportation, storage and accounting of the nuclear material. There are two sites with long term storage facilities in Bulgaria. One site is at Kozloduy NPP and the other is at the research reactor in Sofia. An anti-seismic strengthening programme at Kozloduy wet spent fuel storage facility (SFS) is currently being reviewed by ISUAE. The spent fuel will have to remain in wet storage for an indefinite period of time owing to the Russian Federation’s position, regarding the acceptance of spent fuel.

In connection with the licensing of the activities above, review and assessment have been carried out on the following working projects:

(1) Seismic qualification of I&C systems. Anti-seismic strengthening and replacement, if necessary,

(2) Seismic qualification and anti-seismic strengthening, if necessary, of the SFS safety related mechanical equipment,
(3) Seismic qualification and anti-seismic strengthening, if necessary, of the SFS electrical equipment,

Further assessment of the seismic qualification and anti-seismic strengthening of the SFS safety related electrical and mechanical equipment showed, that work undertaken to date has not resolved the issue. Qualification of the equipment has therefore been proposed for inclusion in the updated programme for SFS safety upgrading.

5.3. CANADA

Legislation is being developed to place into a legal framework the Government of Canada expectations regarding long term management of used nuclear fuel. The following principles are stressed:

(1) Ensure that radioactive waste disposal is carried out in a safe, environmentally sound, comprehensive, cost-effective and integrated manner,
(2) Develop policy, to regulate, and oversee producers and owners to ensure that they comply with legal requirements and meet their funding and operational responsibilities in accordance with approved waste disposal plans,
(3) Waste producers and owners are responsible, in accordance with the principle of "polluter pays", for the funding, organization, management and operation of disposal and other facilities required for their waste.

Currently, the nuclear utilities are taking action to implement this policy, maintaining their responsibility to manage all of their nuclear wastes in an environmentally, socially and financially responsible manner. Part of the tasks undertaken by the Canadian nuclear utilities include the development of conceptual designs for systems to store used fuel for extended periods of time, much longer than the design life of current storage systems. The options considered include extended storage at the reactor sites and a central facility to store the spent fuel from all Canadian producers. Such systems would be designed for a service life of 200 years.

The role of extended storage system would be to allow the time for resolution of technical and social issues in the process of implementing a permanent solution for the isolation of the spent nuclear fuel. A scope of work and technical specifications for the development of conceptual designs and cost estimates for a centralized extended storage (CES) system to accommodate the used fuel from all Canadian reactors. Among other features, the CES system:

(1) Must provide safe storage of all the spent fuel arising from the current Canadian nuclear programme, consisting of approximately 3.5 million bundles of used CANDU fuel,
(2) Shall be housed either in underground or in above ground structures, and
(3) Provide the means for retrieval of the used fuel from the storage area by methods and equipment similar to that used for placing the fuel in storage.

With regard to performance, the maximum fuel sheath temperature during the storage period in the CES facility shall not exceed 200°C. Also, an important feature of regulatory importance is that the CES facility will be designed to meet all of its performance requirements during a service life of 200 years. This includes an active phase of about

30 years followed by a period of approximately 170 years in which the storage facility will operate in a passive mode, i.e. no new fuel will be placed in storage and operations will be limited to monitoring and maintenance.

The reference fuel to be used for developing a conceptual design of the CES system has the following characteristics:

Average Burnup:	220 MWh/kg U
Bundle Power:	455 kW/bundle
Fuel Age:	30 years

A burnup value of 280 MWh/kgU will be used for shielding calculations.

5.4. CZECH REPUBLIC

The dry interim spent fuel storage facility (ISFSF) at Dukovany was commissioned in December 1995, after more than one year of trial operations. It was licenced in January 1977 for a 10-year period. Spent nuclear fuel will be stored in dual purpose CASTOR - 440/84 casks with the total capacity of the facility approximately 600 Mg HM.

5.4.1. Recent situation

(1) Existing Czech legal requirements do not specify the time period for licensing a SF storage facility,
(2) Existing safety criteria (e.g. Shielding, sub-criticality, residual heat removal, retrievability…), shall be met for any time period.

Expected development: system shall be open to accept, for example:

(1) Higher enrichment, higher burnup fuel, implementation of burnup credit,
(2) New storage component materials.

5.4.2. Probable scenarios of future regulatory evaluations

(1) Safety Analysis Report (SAR) evaluations — It is difficult to analyse/predict precisely the behaviour of the spent fuel and storage components for a period of 100 years or longer. A living SAR with a periodical updating (10–20 yrs) will be required. The initial SAR shall explain the concept and principal basis for the lifetime.

(2) Continuity of information/knowledge will have to be assured:
   - Database describing spent fuel history — from the core via wet storage to dry storage — will be necessary,
   - Detailed description of the storage technology will have to be kept for the whole lifetime of the facility,
   - Operational records (especially the off-normal events or accidents) will have to be kept for the whole lifetime of the facility.

   Strict measures will be required to keep relevant information and records complete for the life of the facility (including backup and redundancy).

(3) Maintenance and inspections:
   - Long term programmes of monitoring, maintenance, in-service inspections will have to be performed by the operator,
   - Long term programmes of independent inspections will have to be implemented by the appropriate regulatory body,
   - The ISFSF operator will have to perform a self-assessment.

   The operator must have the ability to maintain required safety features of the storage system, which means to be able to evaluate the influence of time, temperature, radiation and surrounding environment, both on fuel and storage components:

   Strict requirements on monitoring and maintenance will be necessary, including validated and reliable methods of non-destructive testing.

   It was concluded that:
   - Existing knowledge can support the storage of spent fuel for periods up to100 years,
   - Periodic re-evaluations of storage by the appropriate regulatory body shall be required,
   - Systematic and reliable monitoring of all safety related features of the storage systems should be the priority of the operator.

5.5. ITALY

Societa Gestione Impianti Nucleari pa (Sogin), formerly ENEL, has submitted to the Italian National Agency for Environmental Protection (ANPA) a document "Proposal of Design Requirements for Casks and Storage Facilities", to be used for the procurement of metallic casks and asked ANPA for a written "Technical Opinion" on the most relevant safety aspects before issuing a formal bid. Significant safety aspects identified by ANPA included:

- Storage of MOX, U-Th and/or defected fuel,
- Storage of fuel with other highly radioactive materials (controls rods, structural materials, etc.),
- Burnup credit,
- 1 lid (2 gaskets) solution versus 2 lids,

The following include some of ANPA's conclusions:

- Storage of MOX fuel is acceptable, provided special care is exercised in evaluating the peculiar properties of such fuel (e.g., in criticality and shielding analyses), U-Th fuel is a special concern, due to lack of experience,
- The number of MOX elements per cask might have to be limited,
- Visual examination or sipping tests are not a valid means for assessing fuel integrity and the presence of water inside the rods,
- Special care will be dedicated to the review of the cask design, if heterogeneous loading (fuel of different characteristics in the same) is used. Special qualification might be needed,
- ANPA is strongly against the storage of spent fuel with other highly radioactive materials,
- ANPA is currently against taking advantage of "Burnup Credit" for cask design,

– There are special materials in the cask (e.g., neutron absorbers, organic shielding materials, gaskets, etc.) that require careful review regarding their qualification for the entire projected design life (50 years) of the casks, highest degree of reliability and flexibility (e.g., detection of leaking gasket and changing one gasket without the need for a hot cell),
– The 2-lid/2-gaskets solution appears to fulfil the above requirements.

The bid was awarded to GNB in 2000, although the official licensing procedure will begin later.

5.6. LITHUANIA

The State Nuclear Power Safety Inspectorate (VATESI) licences and regulates radioactive waste management activities. Wet and newer on-site dry storage facilities are located at and operated by the Ignalina Nuclear Power Plant (INPP). INPP transmitted a Preliminary Safety Analysis Report to VATESI in April 1995, to operate a dry storage facility at the plant and transfer a portion of the spent fuel from wet to dry storage in CASTOR RBMK-1500 and CONSTOR RBMK casks. The operating licence for 20 years was granted in February 2000, with a requirement for periodic safety reviews. Licensing delays were attributable to technical questions by VATESI that had to be resolved, as well as unexpected legal and social issues. The dry storage facility design life is 50 years, however, an analysis by cask manufacturers has enabled INPP to take a position that the cask life can be extended to 100 years. The existing licence must be renewed after 20 years.

5.7. NETHERLANDS

In 1984, the government of the Netherlands issued a policy paper on radioactive waste that was discussed and accepted by Parliament. This policy for radioactive waste was based on the general environmental management philosophy, which stipulated that all hazardous waste must be isolated, controlled and surveyed. For radioactive waste this means more specifically:

Isolation of radioactive waste from the biosphere:

(1) Control of waste production by means of restrictive use of radioactive materials,
(2) Surveillance of the waste-management system.

In practical terms, this has been translated into the following:

(1) All types and categories of radioactive waste generated in the next 100 years will be stored above ground in engineered structures, which allow retrieval at all times,
(2) This long term storage, together with a centralized treatment facility, is considered as a normal industrial activity and will be located on one single site,
(3) Research will be performed on final disposal possibilities within the Netherlands, or within an international framework.

Essential in this policy and deviating from the policy in many other countries is the long term delay on final disposal. However, this is a logical step for the situation in the Netherlands. Creating a final repository in the Netherlands within a few decades is neither practical nor economically feasible. The volume of waste that has to be disposed of is still very small, public acceptance for final disposal is lacking and, if a repository will be created, the waste has to be fully retrievable.

Research has been conducted for the possibility to store all types of radioactive waste for periods longer than 100 years in engineered structures above the ground. A storage period of 300 years, before final disposal in a repository, has also been investigated.

5.8. SLOVAKIA

Following political and economical changes at the end of 1980s, the nuclear power plant operator in Slovakia decided to change the original scheme of the back end of the nuclear fuel cycle. Instead of reprocessing spent fuel, it would be stored at an ISFSF until final disposition. As a result, reconstruction of the existing ISFSF began in 1997, and the Nuclear Regulatory Authority of Slovakia (ÚJD SR) is expected to issue an operating licence.

The original Slovak spent fuel management plan was based on the assumption that all spent fuel from Russian designed WWER reactors would be transported to the Russian Federation for reprocessing or permanent disposal. Only about 700 spent fuel assemblies were actually transported, because of political changes in the Russian Federation. Currently, spent fuel is stored in at-reactor pools, and in the ISFSF of a wet type, located at the Jaslovské Bohunice site. The ISFSF was commissioned in 1986. After the decision to store spent fuel in Slovakia, Bohunice decided to extend the capacity and lifetime of the ISFSF from 5040 to 14,112 assemblies. With regard to compliance with the ÚJD SR safety functions for the reconstructed ISFSF, the following items had to be demonstrated by the operator:

(1) Assurance of sub-criticality of spent fuel, stored in compacted form in baskets and of baskets in pools, under all nominal conditions and operational events of the ISFSF,
(2) Reliable residual heat removal from spent fuel,
(3) Reliable confinement to avoid release of fission products into the environment.

The operator had to also demonstrate, that all the above mentioned safety functions would not be adversely influenced by variations in the characteristics of important materials during the operational life of the ISFSF, which is assumed to be about 50 years.

5.8.1. Current status

According to the current ISFSF operating licence, the operator has to report yearly to ÚJD SR on the status of materials, construction and equipment that impacts nuclear safety. Thus far, no significant problems have appeared during operation of the facility. Neither are there any symptoms of fuel cladding deterioration or storage component corrosion. Visual observation of the outer surface of selected spent fuel assemblies has been performed using underwater cameras. This inspection did not reveal any significant differences between the newest and oldest stored assemblies and there was no corrosion identified.

5.8.2. Programme for long term monitoring

A long term monitoring program has been prepared by the operator as part of the safety documentation required by the ÚJD SR for issuing a construction (reconstruction) licence. The operator had to demonstrate an ability to control the variance in characteristics of some materials, construction and equipment caused by an ageing process. Results acquired on the base of the monitoring program will allow the ÚJD SR to compare information submitted by the operator in safety reports with the real status and, if necessary, direct the operator to the operator in safety reports with the real status and, if necessary, direct the operator to prepare some corrective measures in advance. The program concentrates on monitoring of:

(1) Stability and degradation of building constructions which is focused on:
  - Basement constructions,
  - Concrete structures of spent fuel pools,
  - Supporting metal elements and constructions,
  - Outer wall panels of the main building,
(2) Pressure vessel and piping system lifetime which is focused on systems of cooling and cleaning of cooling water, and on decontamination, this part includes also programme of non-destructive tests of selected equipment,
(3) Corrosion destruction of selected buildings, machinery and transport technology focused mainly on technology in contact with pool water,
(4) Mechanical status of rotating machines focused on selected pumps and ventilators,
(5) Selected power supply systems and components (transformers, electro-motors and cables),
(6) Status of spent fuel assemblies.

Long term monitoring will be incorporated into the operator's operational program for current maintenance and controls of the ISFSF components.

5.8.3. Monitoring the stability and degradation of building construction

The stability of the ISFSF building basement and status of spent fuel pools will be monitored. According to measurements ongoing since 1987, the basement of the building has moved ("settling-down") differently from calculated values. Because its behaviour could have an adverse impact on the stability of the building, the operator declared that this effect could be eliminated by proper distribution of baskets containing spent fuel.

Because the flow of underground water can also influence the stability of the building, underground water levels will be monitored in drainage pits and wells on the outer perimeter of the building.

Regarding the spent fuel pool walls, monitoring will be concentrated on:

(1) Measurement of vertical movement,
(2) Measurement of relative deformations,
(3) Mapping of cracks,
(4) Wet spots and leaks,
(5) Corrosion of concrete and reinforcements,
(6) Concrete hardness and carbonatization,
(7) Thermal impact.

Monitoring will be carried out in the form of four inspection types:

(1) Preventive inspection: visual inspection twice a year,
(2) Standard inspection: preventive inspection once a year completed by measurement of vertical movements, cracks, and measurement of relative deformations in selected positions,
(3) General inspection: standard inspection once in four years by other monitoring methods,
(4) Extraordinary inspection: following extraordinary events.

Monitoring the corrosion of spent fuel pool components provides important information about the status of components that guarantee basic safety functions of the ISFSF and allows the operator to prepare possible corrective measures in a timely manner.

Spent fuel pool components will be monitored by means of test samples and by acoustic emission. The following materials will be used as samples:

(1) Basic spent fuel pool lining material,
(2) Weld joints made of the same material,
(3) Selected components removed during the ISFSF reconstruction,
(4) Samples of hexagonal tubes made form ATABOR steel used in new compact baskets.

A variety of samples will be prepared:

(1) To determine loss of weight — monitoring of common corrosion, intercrystalline corrosion and overall corrosion status of the sample surface,
(2) CBB (Crevice Bent Beam) type — complex monitoring of influence of deformation, residual stress and crevice effect on corrosion cracking, pit and crevice corrosion,
(3) Circular shape weld — simulation of real weld joint,
(4) U-bend — monitoring of materials with elastic and plastic deformation,
(5) Metallographic examination — exact monitoring of surface effects.

Several types of samples will be placed in every spent fuel pool below and above the water level and into the position of water level fluctuation. Initially, the samples will be analyzed each year. This period can be changed according to acquired results.

Measuring acoustic emissions will monitor corrosion of spent fuel pool lining. Each pool will be equipped with five sensors located in the area of fluctuating water level. Measured signals will be compared with the spectrum measured on samples. Measurement will be carried out over a two-month period. This period can be changed based on results acquired and experience gained.

Additional features of the long term monitoring program include visual inspections, leak tests, destructive inspections, and gamma spectrometry.

The ÚJD SR is going to issue a permit for operation of reconstructed ISFSF only for a limited period (approximately 10 years). The important condition for prolonging the operational licence is to achieve positive results from the long term monitoring program.

5.9. UNITED KINGDOM

BNFL has been contracted to manage the lifetime irradiated Advanced Gas Reactor fuel arising from British Energy reactors. The agreement formulated is a mixture of reprocessing (covering the planned life of the Thorp Reprocessing Plant) and interim storage for up to 80 years for the remainder of the fuel arising. Given the storage duration involved, one of the main regulatory concerns from the outset has been the need to maintain a clear auditable quality assurance trail. The quality assurance based history of the project should detail the decisions taken, justifications and supporting technical/safety cases leading to the final storage method adopted.

A number of engineering and technical issues must be resolved, such as:

(1) Identifying and evaluating the degradation mechanisms affecting the long term integrity of storage pools,
(2) Establishing methods and procedures to monitor the long term condition of the pond, and
(3) Ensuring that the long term corrosion behaviour of the fuel being stored is established and storage conditions are optimized to inhibit corrosion to the fuel cladding and components.

These global issues mirror international regulatory concerns with respect to the duration of storage. In response, BNFL has set in-place plans to resolve, as far as is practical, each issue. Given the storage duration, it is difficult to project to the end of life and provision of contingency plans are necessary, e.g., guaranteeing a civil structure that is already operational will maintain its condition for another 80 years. The fall back position in this case is reliance on routine condition monitoring which enables a look ahead for set periods (five to ten years) and the capability to predict that the building will be structurally sound for that duration. If for some reason a failure occurs, there is sufficient lead-time to build an alternative facility (or use casks) and empty the current facility before safety is compromised.

5.9.1. Changes to plant safety case methodology

Since January 1995, BNFL has been working with the Nuclear Installations Inspectorate (NII) to implement a major revision to the methodology used in preparation of plant safety cases. The result of this collaboration has been the ‘Continued Operation Safety Case’ (COSC), which replaces the ‘Fully Developed Safety Case’ (FDSC), such a safety case has been recently prepared for the LWR Fuel Storage Pond. The main differences between the two approaches are outlined below:

(1) Greater joint ownership of the safety case by the operators, designers and safety assessors,
(2) Whilst the FDSC was mainly based on probabilistic assessments, the new approach puts greater emphasis on a deterministic aspects, the later being achieved by a combination of desktop and walk-down studies,
(3) Against each hazard the supporting structures, systems and components are identified and assessed against a safety function classification on a 1–3 scale, 1 being a major contribution to radiological safety, 2 a significant contribution to radiological safety and 3 a minor contribution to radiological safety,
(4) Detailed Design Assessment Reports are produced for each major plant item/process area based on:
    - Engineering Schedule, including description, safety function etc.,
    - Assessment against modern standards,
    - Achieving ALARP,
    - Assessment against ageing (historic and future),
(5) For new plants there would also be Design Justification Reports,
(6) Review period is continual with a major appraisal every ten years, compared to a five to seven year review previously.

5.10. THE UNITED STATES OF AMERICA

Spent fuel cladding experiences a more harsh environment in dry storage than it does during reactor operation. The most likely failure mechanism in dry cask storage, long term cladding creep, is not a problem in reactor operations. In the reactor, the cladding temperature is near the saturation temperature of the coolant water, which is about 345 °C, and the pressure differential across the cladding are one or two MPa. At reactor shutdown and spent fuel pool cooling conditions, the cold temperature renders creep as a negligible contributor to degradation. However, during dry storage, the maximum allowed cladding temperature can be as high as 570 °C during vacuum drying operations and 340 to 420 °C for long term storage. The differential pressure across the cladding may vary between 11–12.4 MPa. These operating conditions could significantly increase the cladding creep phenomenon, including material properties that can influence fuel behaviour under postulated storage conditions and postulated accident conditions for transportation.

There is a growing scientific consensus that hoop strain is the dominant failure mechanism for dry storage of spent fuel in an inert environment. The high temperature and pressure under dry storage conditions enhances the rate at which cladding creeps. In addition, long term exposure to the reactor environment can result in the build up of zirconium oxide on the cladding and thereby lead to a reduction in ductility of the cladding. New, un-irradiated cladding may fail from a creep strain greater than 10%. However, irradiated Zircalloy cladding may fail from creep at strains less than 1% because of the high oxidation and high hydrogen concentrations that may follow reactor operating conditions.

Based on experience, data, and theoretical modelling of material properties, there is reasonable assurance that for fuel burnup of less than 45,000 MW·d/t HMthe cladding would not fail from creep rupture under dry storage conditions for a period greater than 20 years. However, there is limited data for US approved Zircalloy cladding with burnup in excess of 45,000 MW·d/t HM. The data that does exist support safe storage for oxide thickness of irradiated fuel rods whose oxide thickness is less than 80 micrometers. However, depending on the plant's operating conditions, some Zircalloy clad fuel could build up an oxide thickness that is greater than 80 micrometers. Therefore, the United States Nuclear Regulatory Commission (USNRC) has placed analytic and storage constraints on high-burnup fuel with an oxide thickness greater than 80 micrometers. However, the USNRC does not have data under postulated conditions of dry cask storage for other improved cladding materials at high-burnup conditions. Available literature indicates that cladding will experience increased oxidation, hydrogen concentrations, and stress with increasing burnup. These changes, coupled with decreased ductility for high-burnup cladding, could lead to cladding failure by creep rupture over the period of dry storage conditions.

Under US regulations, "... the spent fuel cladding must be protected from degradation that leads to gross ruptures" [10 CFR 72.122 (h) (1)]. However, cladding failure from creep would most likely result in a slow fission gas leakage through a pinhole crack. This failure mode could depressurize the fuel pin and thereby prevent a gross rupture from occurring. However, the fuel will eventually be transported to a final repository and consideration must be given to the embrittlement of the cladding that results from in-reactor exposure to high fluence.

Currently, the NRC staff believes that creep rupture is the dominant failure mechanism under dry storage conditions. Further, cladding oxide thickness and hydrogen concentration are functions of material and several reactor operating variables (i.e., reactor power history, temperature, water chemistry, etc.). Mechanical properties, such as creep rates and strain limits, are related to oxide thickness and hydrogen concentration, but the correlations will vary with cladding material and reactor operational history. Furthermore, it is uncertain the impact embrittled cladding will have under transport conditions.

The US regulations for storage of spent fuel (10 CFR Part 72) require that the fuel be readily retrievable from a storage cask. The spent fuel cladding must be protected from degradation that leads to gross rupture during dry storage conditions. Further, transportation regulations (10 CFR Part 71) require that the geometry of the contents of a spent fuel package should not substantially change under the conditions specified for normal and hypothetical accident conditions of transport. Without data to support continued integrity of the cladding throughout the storage period and subsequent transportation to a disposal site, it is difficult to demonstrate that the regulations are met for high-burnup fuel.

An alternative approach to demonstrate retrievability of the cladding is to store the fuel in a can within the dry storage canister. This approach assures that criticality will not occur and assures thermal margins. However, this approach is more costly because fewer fuel bundles could be stored in one cask.

To establish a basis for storing and transporting high-burnup fuel as structurally intact fuel, and thus avoid the added costs of canning, data are needed to establish the mechanical properties, long term creep, and characteristics of cladding oxide and hydrogen concentrations.

For existing fuel designs using other than Zircalloy cladding and with burnup exceeding 45,000 MW·d/t U, a database must be developed that can be used to establish acceptance criteria. The data may already exist within the international community or domestically as proprietary data but are not readily available to the entire industry. Until such data are obtained, the USNRC will continue to assume that fuel with other than Zircalloy cladding is failed or damaged for dry storage and transportation purposes. For new fuel designs, the data to demonstrate cladding integrity under storage and transportation conditions could be obtained under the lead test assembly program for the new design. This would assure that the data are available early and avoid undue costs associated with non-optimal cask usage.

An alternative approach may be to analytically demonstrate for a range of fuel conditions ranging from uniformly distributed rubble bed to rubble concentrated in a pile to intermediate failure conditions that criticality would not occur, unacceptable hot spots would not occur, and external doses remain within acceptance limits. This may not be possible. However, if it is then one would need to redefine retrievability to include non-intact fuel geometries. These conditions would also have to be acceptable for final disposal.

From the perspective of the USNRC:

(1) Characterization of high-burnup fuel under normal, off-normal and accident conditions is needed for licensing evaluations,
(2) Characterization of normal-burnup fuel (less than 45 MW·d/t U) is adequately understood for licensing needs, with confirmatory activities ongoing at INEEL.

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