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Potential Interface Issues in Spent Fuel Management

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POTENTIAL INTERFACE ISSUES IN SPENT FUEL MANAGEMENT

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POTENTIAL INTERFACE ISSUES IN SPENT FUEL MANAGEMENT

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FOREWORD

All phases of a fuel cycle raise particular challenges. They include not only maintaining flexibility to accommodate the range of potential future spent fuel disposition options, but also defining and addressing the relevant issues in storage and transport, given the uncertainties regarding storage duration, the availability of future technologies and future financial, regulatory and political conditions.

Taking a holistic view of the nuclear fuel cycle ensures that influences from, and impacts on, all phases of the cycle are clearly understood. This general view facilitates effective decision making in the back end of the fuel cycle (BEFC). It is therefore important to establish interfaces and any potential issues which can impact on the BEFC.

This publication is an output of a series of meetings to identify and evaluate issues and opportunities associated with interfaces in the BEFC and to describe effective management approaches based on the experience of Member States. During the meetings, participants from Member States and other international organizations shared and evaluated the main interfaces and potential interface issues among the spent fuel storage, transport, reprocessing and disposal phases of the BEFC, and also reviewed the national approaches to addressing these issues.

The aim of this publication is to provide an approach to identify the interfaces in the BEFC as well as the potential issues that should be addressed. It also aims at responding to the solutions Member States most often find to be effective and, in some cases, were adjusted or revisited to reach the fixed target. Most of the interfaces and issues are country specific, as evidenced by the variety and diversity of examples provided in this publication.

The IAEA gratefully acknowledges the contributions of J. Van Aarle (Switzerland), B. Carlsen (United States of America), G. Demazy (Belgium), R. Einziger (United States of America) and T. Saegusa (Japan), the technical meeting participants and the assistance from other experts. The IAEA officers responsible for this publication were P. Standring and X. Zou of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

1. INTRODUCTION

Safe, secure, reliable and economic technology for the management of spent fuel arising from nuclear power generation is a key issue for sustainable utilization of nuclear energy. Most of the spent fuel from current nuclear power production is stored while awaiting national and international decisions on final disposition.

Unresolved back end of the fuel cycle (BEFC) issues and the lack of a disposal solution often generate public resistance to existing or new nuclear power plants and would be highly detrimental for the viability of nuclear energy. Rejection of a proven BEFC solution or planned project, facility site, or transportation route generates doubts, and re-enforces the idea that there may be no acceptable BEFC solution.

To address a number of potential issues and opportunities, careful consideration of the challenges associated with spent fuel management, including extended storage and subsequent transport is appropriate. As an example, IAEA Member States opting for the 'wait and see' approach to the BEFC have indicated that one benefit of waiting is to learn from experience of other countries. Some Member States have noted that, with final disposition potentially being postponed for several decades, the evolution of storage technologies may result in many different package types that could have long term consequences associated with deployment of the potential treatment or disposition options for the fuel.

In the absence of a defined endpoint, such as reprocessing or direct disposal with a commitment to construct the necessary facilities, an effective programme to manage spent fuel has become both increasingly important, due to the potential for significantly extended storage durations, and increasingly complex due to the uncertainties and potential ramifications on future options that must be considered when making decisions in each phase of the fuel cycle. A systems approach is needed to ensure that influences from and impacts on all phases of fuel cycle are taken into account when making decisions. Particular challenges include maintaining flexibility to accommodate the range of potential future spent fuel disposition options as well as defining and addressing the relevant issues in storage and transportation given the uncertainties regarding the storage duration the availability of future technologies and also of future financial, regulatory and political conditions.

The evaluation of potential interface issues reflects a growing awareness of Member States of the need to be informed by full consideration of interdependencies among phases of the nuclear fuel life cycle, policy considerations, and various stakeholders that influence management options and decisions associated with spent fuel. Because many issues affect multiple stakeholders and may require lead times to resolve, it is important to identify interface issues early and solve them in a timely manner.

Sharing information on interfaces and requirements will also help keep 'cross border and/or multi-lateral' options open. Given that no country has yet fully resolved BEFC issues, strategies that enhance the possibility of international options will appeal to many states. Sharing information on interfaces and requirements will enhance the potential for international collaboration. Several Member States and IAEA sponsored activities have begun to develop standard guides and recommendations consistent with a more holistic approach to the management of the nuclear fuel cycle. IAEA initiatives have included:

- Development of a performance based definition of damaged fuel [1];
- Development of a regulatory framework that considers the needs of both storage and transportation of spent fuel (Nuclear Safety and Security Department/IAEA – International Working Group for an Integrated Safety Demonstration for the Dual Use Cask for Spent Nuclear Fuel);
- A consultancy working to develop a guidance publication to identify issues and opportunities to be considered when extending the storage period for spent fuel;
- A coordinated research project (CRP) focusing on specific demonstrations of spent fuel performance in dry storage over periods of time longer than addressed by existing regulatory licensing precedent and subsequent transportability of the fuel.

1.1. OBJECTIVE

Opportunities to efficiently manage the BEFC are lost if interfaces are not identified and addressed in the early planning stages.

The objective of this report is to suggest and illustrate a process for systematically identifying and evaluating the potential interface issues in spent fuel management, and to give examples of effective management approaches based on the experience of Member States.

1.2. SCOPE

The conclusions made in this report are based on the experience and opinions of the members of the consultancy meetings and Member States that participated in the technical meeting. Many of the interfaces and recommendations reflect good practices identified in existing national and international standards and guides. Nonetheless, the rigorous process for identification of interfaces as outlined in this report helps to systematically identify issues and opportunities associated with management of the interfaces within the BEFC.

In order to make this report tractable, limits on its scope have been made. This evaluation deals primarily with the back-end of the light water reactor (LWR) and heavy water reactor (HWR) fuel cycles. Although many of the interface considerations are common to spent fuel from all reactor types, interfaces specific to non-LWR and HWR fuel cycles are not explicitly considered.

Within this publication the nuclear fuel cycle is broken into two parts; the front-end comprising the facilities that manufacture and irradiate (reactor operations) the fuel thus producing spent fuel, and the back-end comprising storage, transportation, reprocessing and/or disposal of the spent fuel. Historically, much of the R&D emphasis has been on ensuring safety and profitability of the reactor operations phase. This approach has resulted in a policy, regulatory framework, financial structure, equipment designs, and operational strategies that may not be most efficient when the impacts of the entire fuel cycle are considered; especially the management of spent fuel and other waste materials generated during the operational phase.

The BEFC was divided into storage, transport, reprocessing, and disposal. This effort focuses on the interfaces that exist between these phases and the associated management entities and affected stakeholders, referred to herein as ‘participants’. Interfaces that occur within a given phase are normally addressed in facility specific operations procedures and were considered outside the scope of this effort. No attempt was made to determine which phase was the driver for the interface, only that an interface existed. The front end of the fuel cycle (i.e. fuel design and manufacturing, and reactor operations) is considered only to the extent that they influence the characteristics of the fuel and requirements for handling within the BEFC.

Transfer and transport of spent fuel are differentiated. Those activities that move fuel within a site boundary are considered transfers and are not included in the scope of this publication. Examples are fuel movements from the reactor core to the spent fuel pool or from the spent fuel pool to dry storage at the same site. Transport is considered to be the movement of fuel between two distinct facilities over public roads, rail or waterways. Transport is considered as a distinct phase of BEFC operations and is addressed within this publication.

Safeguards are also a major concern in any activity related to the BEFC. It is assumed that safeguards obligations are duly considered by any entity involved in spent fuel management. Hence, safeguards and security are not directly addressed in this report. However, effective communication and integration in the BEFC will facilitate effective implementation of safeguards and security.

A complete treatment of the interfaces at the BEFC would include other waste streams resulting from the potential treatment of spent fuel (i.e. uranium, plutonium, low level waste, intermediate level waste). Although there are interface issues associated with the process steps that generate these streams, they are considered out of scope with respect to this report.

This report describes a process for identifying and managing interfaces to achieve spent fuel management objectives. The phases and interfaces identified in this report are not intended to produce a conclusive or fully developed list of issues. Rather, they are intended to serve as an example of the process and as aids to help trigger thought, questions, and discussion as individual countries work to identify and manage their own interface issues.

The report does not specifically look at or address safety issues related to interfaces. The reader is referred to the IAEA safety standards series for example:

- Predisposal Management of Radioactive Waste, Safety Standards General Safety Requirements Part 5 No. GSR Part 5, IAEA, Vienna (2009);
- Storage of Spent Nuclear Fuel, Safety Standards Specific Safety Guide No. SSG–15, IAEA, Vienna (2012).

1.3. AUDIENCE

This report will be useful to:

- Managers and workers in the BEFC that would like the benefit of previous international experience dealing with interface issues;
- Planners in countries with existing nuclear generation programmes who want to improve the efficiency of spent fuel management and minimize interface related issues;
- Countries who are contemplating starting a nuclear generation programme and want to inform decisions at the front-end of the fuel cycle with consideration of the effects on the BEFC;
- Any stakeholders who want to understand the inter-relationships among phases of the BEFC and to effectively participate in decisions related to the management of spent nuclear fuel.

1.4. CRITERIA FOR AN INTERFACE TO EXIST

An interface exists when an output or product from a phase or a participant is needed as a prerequisite or input to another. An interface issue exists when there is a potential for an unintended consequence, inefficiency or missed opportunity if not properly coordinated. Interface issues may be operational, regulatory, economic, societal, political, or safety and security related considerations. For example:

1.4.1. Operational

- Availability and timing of shared resources such as commercial casks and transport equipment;
- Equipment incompatibilities such as a cask arriving at facility with insufficient crane capacity or clearances;
- Failure to properly specify or to demonstrate compliance with criteria;
- A reprocessing facility requiring a specified mix of spent fuel burnups and a shipper with an incompatible inventory of spent fuel;
- Availability of necessary equipment and properly trained personnel to support schedule commitments;
- Unanticipated intervention from various stakeholders;
- Availability of financing when needed;
- Cost of including a step in the process;
- Transfer of key information and responsibilities for record keeping.

1.4.2. Regulatory, safety and security

- Shipments that cross jurisdictional borders;
- Intermodal transport;
- Long term storage of spent fuel on a site licensed for reactor operations;
- Long term storage at centralized interim storage sites;
- Regulatory bodies with overlapping jurisdictions;
- Waste forms without a clearly defined regulatory framework;
- Activities that influence policy;
- Decisions and actions likely to affect public acceptance;
- Susceptibility to changes in the political climate;
- Costs imposed on other parts of the process;
- Transfer of physical protection responsibilities.

2. PROCESS

2.1. APPROACH

Figure 1 is a flow diagram showing the major steps of the BEFC[1]. A TM was held 3-6 November, 2009 to identify needs, interests, and experiences from Member States regarding interface issues associated with the various paths associated with the BEFC. The chairman's summary and papers presented at the TM are provided in ANNEX I and II. The output from the meeting was used as input to address the interfaces of interest.

2.2. DEVELOPMENT OF THE MATRICES

Matrices were developed to facilitate determination of the interfaces associated with the BEFC. Table 1 (Section 3.1) and Table 2 (Section 4.1) show, respectively, potential interfaces among the phases and among the participants. The phases were identified from the flow diagram (Fig. 1).

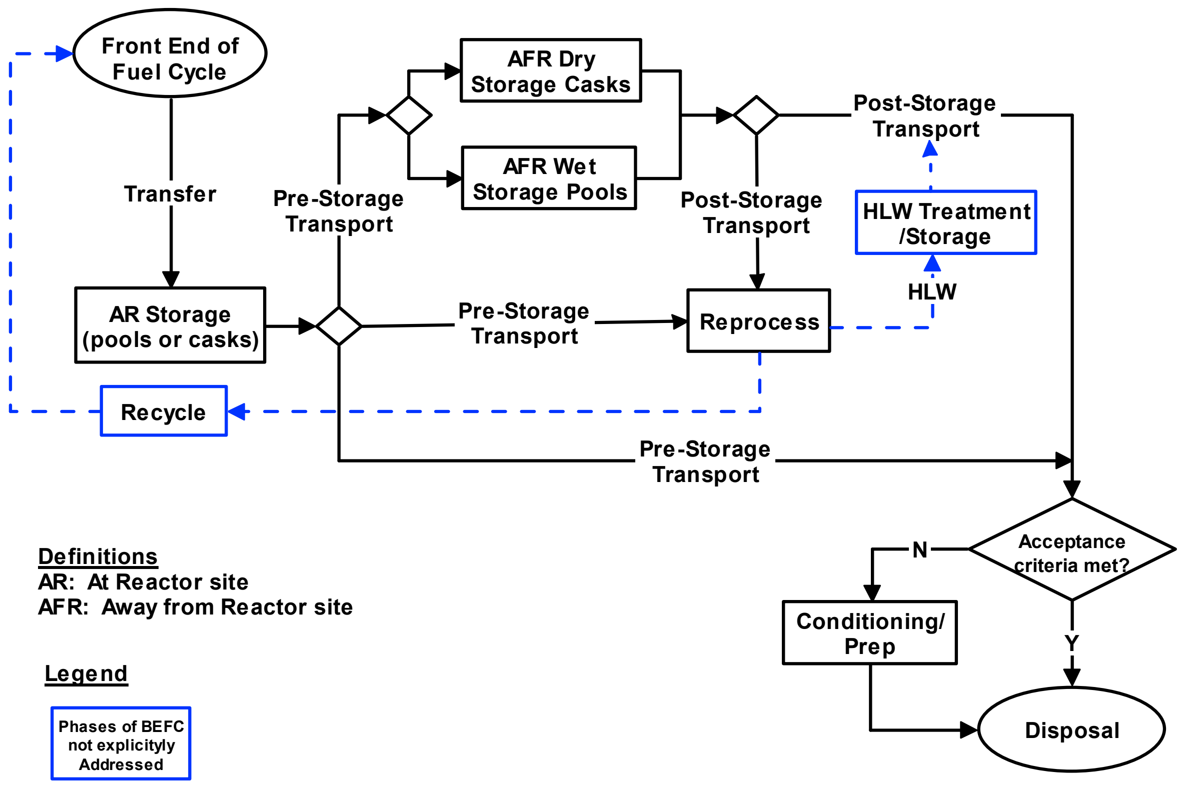

FIG.1. Flow diagram of interfaces with and within the back-end of the LWR fuel cycle.

[1] A number of flow paths can be used depending on the number of transfers between pools and dry storage that are contemplated. For simplicity, only the more commonly anticipated flow paths are shown. Other flow paths can be constructed from the steps in the flow paths shown.

Identification and classification of the participants were more difficult for several reasons:

- The regulatory and political framework varies within different countries. Roles and authorities associated with the industry, regulator, and policy maker as well as the effects of public involvement vary widely;
- There are many types of stakeholders with significantly different roles such as those having political or financial interests and others who are concerned about the radiological impacts of a nearby facility;
- Many countries have to interface with entities in other countries as a result of the need to transport spent fuel or high level waste across borders to a storage, treatment, or disposal facility; or to reprocess and transport recycled materials (mixed oxide fuel (MOX), reprocessed uranium, etc.);
- The endpoint for spent fuel affects the issues and the importance of different choices in spent fuel management. Country specific policies for the endpoint include "wait-and-see", disposal, and reprocessing. Regardless of a country's stated policy, uncertainty remains until the policy is fully implemented. Under any policy scenario, ultimate disposal will eventually be required;
- The effects of many interfaces and interactions are not confined to just two parties and thus are not confined to a single cell in the participant to participant matrix.

Two distinct groups of participants were identified. The first includes national government and nuclear industry participants. The second includes local entities that influence and/or establish local policy (i.e. local governments, tribal entities, members of the public, political organizations, and other local interests). Interfaces among the national government and industry participants are addressed in Section 4. Section 5 addresses key interfaces with local entities.

There may be additional phases and/or participants specific to a country or a BEFC management strategy. Further, phases such as transport may have many sub-phases such as sea, ground or intermodal. The methodology provided in this publication can easily be expanded to address these additional considerations. Although several key participant to participant interfaces and issues are identified, the relative importance and applicability of each, as well as impacts on other participants, should be carefully considered with respect to each individual country.

In the matrices (Tables 1 and 2), a cell entry refers to the section in the publication that discusses associated interface issues. Examples (positive and negative) for addressing the issue that have been employed by Member States are provided in the Appendix when available.

3. THE PHASE – PHASE MATRIX

This section illustrates a process for identifying interactions among phases within the BEFC and also with other phases of front-end of the nuclear fuel cycle.

3.1. DESCRIPTION OF THE MATRIX

The matrix given in Table 1 denotes the different interfaces among the phases of the BEFC. It also describes the strong dependence of the back-end on fuel design and reactor operation. Although the consultancy meeting acknowledged the existence of different modes of transport (i.e. road, rail, sea, and air), from a practical view, the interfaces are sufficiently similar not to make a distinction within the matrix.

3.2. CROSS CUTTING ISSUES

A number of cross cutting safety and technical issues were identified. These were placed into the following four categories:

- Fuel design ramifications;
- Compatibility of fuel with acceptance criteria for present and future phases;
- Compatibility of packaging and equipment between shipping and receiving facilities;
- Knowledge management/adequacy of records.

3.2.1. Interfaces between front-end of fuel cycle and BEFC

Fuel design requirements should include BEFC performance considerations. Fuel design and irradiation conditions affect the spent fuel characteristics which in turn affect individual safety assessments, risks, available options and costs of handling, transportation, storage, reprocessing, and disposal. For example, although a move to higher fuel burnup typically reduces front-end costs, it adds cost on the back-end by increasing storage times and the potential for personnel exposure necessary storage, handling, reprocessing, and disposal. High burnup has been found to exacerbate all phases of the management of spent fuel, particularly through increased heat loads and increased susceptibility to degradation [2]. Another example is the effect of fuel design and burnup on cladding condition and its susceptibility to failure during storage and transportation.

Although fuel design decisions have significant impacts on the available options and costs in spent fuel management, BEFC considerations are often not explicitly included in design criteria of nuclear fuels. Fuel design and reactor operations decisions should ensure compatibility with all steps of the BEFC, i.e. storage, transport, reprocessing, and final disposal. Similarly, BEFC facility operators and facility/equipment designers should consider a range of fuel design parameters with the objective of accommodating evolving fuel designs. Discussion and negotiation are necessary in order to evaluate how specification changes could affect BEFC safety assessments, risks, costs or available options, and to reach a consensus that satisfies the requirements of the fuel vendor, nuclear power plant (NPP) operator and of BEFC management.

A successful BEFC is a key to growing the industry. Hence, it is in the interest of fuel vendors, nuclear power plant (NPP) operators, and BEFC facility owners/operators to work together to ensure that decisions are made in view of their impacts on the entire fuel cycle rather than to optimize any particular portion.

TABLE 1. THE PHASE – PHASE MATRIX

	Wet storage (AR and AFR)	Transport	Reprocess and disposal direct disposal
Front end of fuel cycle and reactor operations	See section 3.2		See section 3.2
Wet storage (AR and AFR)	See section 3.3.1	See section 3.3.2	See section 3.2
Dry storage (AR and AFR)	See section 3.3.1	See section 3.3.2	See section 3.2
Transport		See section 3.3.3	See section 3.2
Reprocess and disposal direct disposal		See section 3.2	

3.2.2. Maintaining compatibility of fuel with acceptance criteria for present and future phases

Receiving facilities should specify acceptance criteria at the earliest opportunity in order to establish requirements for fuel condition and equipment compatibility as well as the information and records needed to demonstrate compliance.

The fuel owner is responsible for ensuring that the spent fuel and any associated packaging fulfil all relevant requirements such as:

- Physical compatibility (size, weight, structural integrity, etc.);
- Compatibility with safety basis (thermal, radiological, criticality, etc.);
- Compatibility with handling, transport and storage requirements, including suitability for retrieval and transport after the anticipated storage period;
- Known or likely requirements for subsequent disposal or other management aspects included in the owner's spent fuel management strategy, such as the need for further treatment or conditioning of spent fuel;
- Damaged fuel must be identified and placed into a condition that meets acceptance criteria.

The following are major considerations relative to ensuring compatibility with acceptance criteria.

3.2.2.1. Thermal load and dose limits

Cooling pool

Cooling pool design should ensure adequate removal of decay heat likely to be generated by the maximum inventory and heat load of spent fuel anticipated during operation. The design of heat removal systems should include an additional margin of heat removal capability to account for processes likely to degrade or impair the system over time. In determining the necessary heat removal capability, the post irradiation cooling interval and the burnup of the fuel to be stored should be a major consideration. Redundant or diverse heat removal systems should be provided.

Timing

Thermal load and dose limits for the transport cask will determine the length of time that the fuel has to cool before it can be transported. In some cases, cask thermal limits for storage may be more restrictive than for transport. Thermal loading often establishes the timing when fuels may be moved between subsequent phases. Thermal loads are affected by fuel burnup and fuel type (i.e. MOX fuel). Higher heat loadings and/or cask loading constraints may require longer cooling times. As a result, additional cooling pool and heat rejection capacity may be required to maintain sufficient cooling pool capacity for defueling. Alternatively, additional storage casks and/or storage casks with additional heat rejection capacity may allow shorter cooling periods and thus offset a need for additional cooling pool capacity. Higher heat loads will also affect timing, throughput, and dose during transportation and subsequent reprocessing and disposal.

Issues associated with cask thermal loading have become increasingly important due to higher burnup fuels and larger capacity casks.

MOX fuel

The increased source term for MOX fuel from reprocessing increases both heat loading dose during fuel fabrication and handling. This affects reactor safety analyses, cooling time, cask loading limitations, etc. In addition, the presence of increased quantities of plutonium in the fuel may necessitate more rigorous safeguards requirements during transport.

3.2.2.2. Cladding integrity

Events during storage or transport may change the spent fuel integrity so that it might require remediation, repackaging, or other design solutions to ensure acceptability for post storage transportation, reprocessing, or disposal.

Fuel design and burnup history affect fission gas production and resultant stresses in fuel and cladding. Fuel cladding creep and rupture temperature is related to cladding stress state, which may be a factor in cask and facility safety analyses. Stress state of the fuel may also influence the tendency for hydride reorientation in the cladding. Some current fuel designs result in excess hydrogen in cladding that, at high temperatures, may result in hydride reorientation. The resulting loss of ductility may limit the fuel's ability to withstand transportation accident scenarios.

Failed fuel should be identified and documented to ensure that proper accommodations are made during subsequent phases. Assessment of spent fuel integrity may be necessary prior to acceptance and placement in storage. Damaged fuel or spent fuel debris must be identified and placed into a condition that meets acceptance criteria. Suitable equipment shall be provided to identify failed fuel bundles for the purpose of segregated storage and subsequent reprocessing. These include equipment such as sniffing facility, periscope, etc. for direct or remote inspection of fuel bundles and other irradiated components.

3.2.2.3. Dryness

When fuel is loaded into a cask, characterization may be needed to confirm compliance with acceptance criteria (dryness, thermal, etc.) before fuel is accepted. Drying and demonstration of dryness is normally done at the wet storage facility where fuel is loaded. This may impose requirements on the receiving and/or the shipping facility to demonstrate or evaluate that criteria are satisfied and/or to accommodate any discrepancies. The evaluation of the moisture content is an important step in the demonstration of the absence of hydrogen build-up in the cask, and degradation of the fuel due to corrosion and galvanic action. It therefore requires adequate equipment and tools, accurate procedures, maybe adapted facilities, and exacting quality control.

3.2.2.4. Undefined or changing disposal criteria

Acceptance criteria need to be known in order to enable storage conditions, packaging, and other decisions that assure criteria can be reasonably met. However, acceptance criteria for the disposal facility are often not available and/or may change after storage decisions are made. In many countries, disposal time is far in the future and the criteria for acceptance of spent fuel on the disposal site are not well defined. In some cases development of different final disposal concepts for spent fuel could lead to different types of casks or canisters. The fuel owner may have to demonstrate that storage conditions did not jeopardize the compatibility of his fuel with final disposal criteria (e.g. cladding may or may not be credited in repository safety basis). Casks loaded with the objective of optimizing dry storage may not meet disposal criteria. Repackaging, which increases cost and exposure, and/or the need to select a repository site and design disposal facilities for existing sub-optimum packages could be avoided by thoroughly identifying the interfaces and addressing the issues in advance.

Until final disposition is defined and implemented, it is not possible to determine if spent fuel can meet the acceptance criteria without further conditioning. Hence, efforts should be made to preserve flexibility for future options.

3.2.3. Ensuring compatibility of packaging, equipment, and operational controls between shipping and receiving facilities

3.2.3.1. Compatibility of packaging for damaged fuel

If damaged fuel is present, special equipment, packaging, or processes may be required. Storage packaging may or may not be designed for transportation (and vice versa). Likewise, the packaging may or may not permit monitoring/inspection of fuel condition, which may affect transportation decisions. For damaged fuel, perform additional safety analyses is required to determine the need to modify the transport package and/or its safety basis and authorization.

3.2.3.2. Compatibility of equipment and facility

Cask and fuel handling and other equipment interfaces must be designed and managed to ensure the cask can be handled at both shipping and receiving facility. Similarly, other facility constraints such as overhead clearances and crane capacities at the receiving facility, shielding, etc. must be compatible with the package being shipped.

3.2.3.3. Contamination controls

Contamination controls on the cask and transport conveyance must be effectively understood and addressed by the shipper and receiver.

3.2.4. Knowledge management and adequacy of records

The safe and cautious management of the spent fuel records and information is necessary to demonstrate compliance with facility acceptance criteria and to enable transfer of ownership. Records from manufacturing and reactor operation must include sufficient information on handling and developments during storage to demonstrate that the acceptance requirements for subsequent phases of the BEFC are satisfied (preferably without the need for re-opening the cask) and records management must have sufficient controls to ensure their acceptability. Adequate records are also necessary to support the decommissioning of storage and other BEFC facilities. Some examples of good practices are summarized below:

- If the acceptance criteria for the spent fuel endpoint are not clearly specified, preserve comprehensive information in order to preserve the ability to demonstrate compliance with future acceptance criteria;
- To support future transfer of ownership, maintain a complete information chain of the spent fuel history (fabrication, operations, and storage);
- Establish and undertake an operating experience feedback (OEF) programme to collect, screen, analyse, and document operating experience and events at the storage facility in a systematic way. Consider also relevant operational experience and events reported by other facilities;
- Gather, safeguard, and periodically update digitalized media containing any information considered useful for the future management of the spent fuel.

3.3. EVALUATION OF THE PHASE TO PHASE MATRIX

3.3.1. Wet storage and dry storage

3.3.1.1. Degradation during storage

Storage conditions and duration may result in chemical and/or physical changes to the fuel and casks or canisters. For example, older fuel could become brittle and require additional fuel handling and/or packaging requirements. The condition of the fuel and its packaging affects its transportability and ability to meet acceptance criteria for the next phase of the BEFC (e.g. transport, dry storage, reprocessing, or disposal). Below are a few common practices for controlling and monitoring to detect degradation:

- In wet storage, establish and control limits on water chemistry, temperature, etc. to ensure that potential degradation by corrosion does not result in the need for special handling or does not limit available options for treatment and disposal of the spent fuel;
- In dry storage, control and maintain the casks to assure that their safety functions will not be compromised during the expected storage period (up to 100 years or even more). As only outer parts of casks can be a subject of operational controls (cask body, top lid, trunnions) the conditions inside the cask may need to be monitored. For example, degradation of gaskets is indicated as a change of gas pressure inside the cask or in control space e.g. between two lids. Degradation of

neutron shielding could be detected by an increase of the intensity of neutron fields from the cask;
- Degradation during dry storage is a potential concern. At present, there has been little need for moving fuel from one dry storage to another dry storage facility. However, this is likely to become a much more important interface in the future. As storage periods are extended, it is likely that fuels will be consolidated to regional facilities and/or moved to newer storage facilities.

3.3.2. Transport and storage

3.3.2.1. Degradation during storage

See Section 3.3.1.1

3.3.2.2. Damage during transport

Transport could change the material characteristics of the package and affect its ability to meet subsequent storage requirements. Vibration/impact during transport could affect metal seals and/or fuel condition. Fuel damaged during transport may require repackaging and/or storage in designated packages or pool areas.

Mechanical vibration during pre-storage transport may compromise the containment performance of the storage cask. If seals degrade or become compromised during transport or storage, the storage facility may not have the capability to correct the deficiency. One may consider replacing the secondary lid seal or installing an additional lid with an appropriate gasket for post-storage transport.

3.3.2.3. Re-flooding of dry storage casks

Re-flooding of storage casks may be necessary if they must be opened for unloading, repackaging, inspection, or repair/replacement of components. Pools may be needed for planned or unplanned maintenance or inspection of casks and their inventory (e.g. gaskets on primary lids, conditions of spent fuel, etc.). The framework for using the pools is set up either by the storage facility license or by the reactor facility license. As the lifetime of spent fuel storage facility may exceed the operational lifetime of a reactor, a means to perform cask inspections and maintenance must be maintained if the reactor and pools are decommissioned.

3.3.2.4. Transport licenses

Transport licenses may during storage period and requirements for relicensing may change.

3.3.3. Transport and transport (Intermodal)

Changing transport carriers and/or mode of transportation add complexity. Hence all aspects (technical, political, safety, etc.) should be carefully considered, including but not limited to:

- Equipment compatibility (operational);
- Licensing authorities and regulations (regulatory);
- Secure transfer site (security);
- Public acceptance associated with different modes of transport.

4. THE PARTICIPANT – PARTICIPANT MATRIX

The former section of the report presented a methodology for identifying the interfaces and issues among phases of the BEFC. Resolution of these issues requires interaction among various participants. This section follows a similar process for identifying interfaces among key participants in the BEFC. Potential issues to be addressed at these interfaces are provided along with potential solutions. When available, examples as to how these issues have been addressed in different countries are given in the Appendix.

Addressing BEFC issues is often a costly activity that may require years of persistent effort and investment. Responsibilities for the BEFC may lie with various industry and/or government organizations. The time periods for planning and implementing BEFC strategies may be longer than the tenure of responsible personnel, organizations, and political leaders. Consequently, a clear designation of responsibilities is essential.

4.1. DESCRIPTION OF THE MATRIX

The matrix given in Table 2 illustrates potential interfaces among key industry and national governments participants in the BEFC. Interfaces between these groups and local entities are addressed in Section 5.

In initial drafts of this report the various entities encompassing 'Industry' were separately considered and individual interfaces were identified. It became apparent that many of these interfaces were the same and thus repeated, and, depending on the particular country, the responsibilities may be different for the same title. Therefore, several participants were combined into one column of the participants–participants matrix entitled 'Industry', which includes:

- The fuel vendor and the fuel owner;
- The cask vendor and the cask owner;
- The NPP operator and the storage facility owner/operator;
- The transport carrier/consignor;
- Reprocessing plant owner/operator;
- Disposal facility owner/operator.

Hence, the interfaces shown with 'Industry' may include interfaces with and among any of the above. It is left to the reader to ensure interfaces within the industry participants are properly identified and to determine which individual industry participants manage interfaces with non-industry participants.

It has been observed that different participants can correspond to different departments of one entity or company. The interfaces are therefore, to be regarded as interactions to be managed to facilitate smooth implementation of activities rather than contractual relationships between commercial companies. It is up to the individual country and commercial company to define who in their organization is responsible for addressing the interface issue.

TABLE 2. THE PARTICIPANT – PARTICIPANT MATRIX

	Regulator	Bordering/transit countries	Industry
Policy maker	See section 4.3.1		See section 4.3.2
Regulator	See section 4.3.3	See section 4.3.4	See section 4.3.5
Bordering/transit states			See section 4.3.6
Industry			See section 4.3.7

4.2. CROSS CUTTING ISSUES

The chief cross cutting issue with respect to the participants in the BEFC is the need for clearly defined ownership and accountability for management and disposition of spent fuel. This is addressed differently in each country. Regardless of how these responsibilities are divided between government and industry, a long term vision and commitment to implementation is essential for success.

At some point in time, transfer of ownership from the spent fuel owner to the state or to a public body responsible for the long term disposal has to be organized. That transfer of ownership requires all fuel data to ensure the safe handling and storage of the fuel is provided to the receiving entity. Such data will document the mechanical, physical, chemical, radiological fuel characteristics as well as the irradiation history of the fuel. That transfer does not preclude that the fuel owner remains responsible, in part or in total, of the technical, safety, ecological, financial consequences of any defect not detectable when the ownership has been transferred to the state or the designated public body. No feedback presently exists about the organization of such transfer of ownership. However, technical, jurisdictional, legal and financial obligations will have to be thoroughly analysed in order to clearly identify the scope of responsibilities of each party.

4.3. EVALUATION OF THE PARTICIPANT–PARTICIPANT MATRIX

4.3.1. Regulator and policy maker

The role of the regulator is to translate national policy into workable regulations. The regulator is responsible to advise and inform the policy maker on technical matters and on the effects of policies with respect to assuring the safety of the public and the environment. Despite political and industrial influence, the regulator must take care to carry out its mandates with impartiality.

4.3.2. Industry and policy maker

Clearly defined and stable policy is necessary in order for the industry to effectively plan for meeting its responsibilities with respect to managing the fuel cycle. With a national policy that is consistently held, a strategy can be followed and optimized, enhancing confidence in the availability of a national solution for the BEFC. The back-end policy defined by the policy maker determines the waste and spent fuel management paths available. Though the need for facilities is established by industry, political considerations affect the location, capacity, schedule, and financing approach for disposal facilities.

Although consistent policy is essential for effective investment, periodic review and update of policies to account for current and foreseeable energy and environmental needs is a necessary part of the process for maintaining sound policy. For example, the interim storage capacity must be based on the production from the running NPPs and the availability of an endpoint.

Availability of an endpoint (reprocessing and/or disposal) affects the need for storage capacity and its duration. Duration of storage period affects condition of fuel and fuel package – which may limit and/or add costs/constraints to future options. The length of the storage period and associated requirements depends upon the availability and actual implementation of national policy relative to closure of the nuclear fuel cycle.

In countries where a closed fuel cycle is a well-defined national policy, indefinite storage of the spent fuel may be avoided. In some cases, the policy maker has recognized the importance of a firm policy and has taken action to ensure policies are developed. For example, a directive establishing a European Community framework for the responsible and safe management of spent fuel and radioactive waste management was recently issued. This directive (DIR2011/70/Euratom) requires European Union (EU) Member States to develop their national policy and programme plan by 2015.

Changing or uncertain policies present difficulties. Siting and planning are long lead activities that require continuity in policy to obtain financial investment. Unclear or uncertain policy, policy changes, and associated legal challenges result in schedule and financial uncertainty. To provide some buffer from the effects of uncertain or changing policies, the following practices should be considered:

- Develop and implement flexible strategies that can accommodate policy changes;
- Plan spent fuel storage facilities with sufficient capacity and longevity to accommodate uncertainty in national policy.

If there is no policy, the resulting uncertainty impedes public confidence and industry growth. For example, the inability to reprocess or dispose of spent fuel requires expanding storage capacity and duration. Yet the lack of an endpoint results in an inability to make commitments regarding the duration and needed lifetime of storage facilities; thus increasing the difficulty of siting and licensing nuclear facilities.

In some cases, the relevant policy maker may include policies from other countries or affected territories. For example:

- New build of nuclear facilities in the vicinity of a border, sending waste or spent fuel abroad for treatment, or cross border transportation would likely involve dealing with different policy making bodies;
- Change in the policy for shipping spent fuel to the receiving state or through transit states could result in delay or cancellation of shipments; thus necessitating the shipping country to build additional storage or to change their policies accordingly.

4.3.3. Regulator and regulator

It is the responsibility of the regulator(s) to ensure that the regulations for the various stages of the BEFC are compatible in order to provide the industry with a clear and well integrated regulatory framework.

Regulators should ensure that licensing policy accounts for the future phases of the fuel cycle in order to ensure that no irreversible actions are taken that could make the subsequent phases difficult or impossible to achieve.

4.3.4. Regulator and bordering/transit countries

Intergovernmental Agreements are necessary for coordination of cross border transport. Shipping, receiving, and transit countries may have policies and/or regulations that must be coordinated in order to ensure that cross border transports can be completed. For example these agreements often include verification of the existence in that country of adequate equipment, facilities, and procedures corresponding to international standards and confirmation that the country of destination has ratified the 'Convention on Nuclear Safety' and the IAEA 'Joint Convention'.

For international shipments of MOX fuels or spent fuel for reprocessing or disposal, intergovernmental agreements between the country of origin, the destination country, and any transit countries are required. Border/transit states (also known as coastal states) should be engaged diplomatically in order to demonstrate the safety of transport and to secure free passage.

4.3.5. Industry and regulator

Regulations that ensure safety and security are essential for effective operation of the BEFC. Transport regulations [3] are available for cross-border transportation and are applied for domestic transport in many countries as well. Several IAEA guidance publications are available for use by countries developing storage regulations [4 – 6].

Industry may seek exceptions and/or propose revisions to the regulations or regulatory guidance when they believe regulatory requirements impose costs without commensurate public benefit or when more effective methods are identified to achieve the intent of the regulations. As an example [3] and [7] are reviewed on a periodic basis to address proposals from industry and Member States. As regulations continue to evolve, it is of primary importance to evaluate impacts of regulatory changes to assure that licensed equipment continues to meet any new requirements or to specify any associated operational restrictions.

Regular interaction between the regulator and industry participants occurs to ensure necessary licenses are in place and followed. A number of interactions regularly occur to maintain the licenses, obtain interpretation and regulatory guidance, and evaluate changing regulatory and plant conditions. The Regulator interacts at every step of the operating license granting process for BEFC operations and facilities, including the issuance of the acceptance criteria. Vendors/licensees often interface with the regulator to evaluate and provide technical justification for new technologies and solutions that may enable more effective methods to comply with regulatory requirements.

Licenses are usually valid for a limited period of time (e.g. three to five years is typical for spent fuel transport casks while storage facilities may be licensed for 40 years or more), after which the license must introduce an application for extension or renewal of the license as a condition of continued operation. The regulator reviews the safety analysis report that specifies form and content, and sets constraints on receipt/handling of damaged fuel. Each renewal or extension can introduce the opportunity of re-visiting agreements granted on past justifications. Relicensing activities should be undertaken well in advance of the expiration date to demonstrate that the safety evaluation remains in compliance with applicable regulations and/or to identify a relicensing approach.

In the framework of license renewals and new projects,, additional justifications or practical measures, as a consequence of reported incidents experienced in a country or in similar storage facilities located abroad or the revised interpretation of regulations, could be required by the regulator either as a prerequisite to the approval of a project or during its execution. Additional requirements, tests, and studies may be required and relicensing requirements are uncertain due to potentially changing conditions resulting from aging and/or the evolution of regulations. It is beneficial for the industry to respond to regulator requests for information and analysis in a timely fashion. Legally binding time limits must be observed. This could ultimately lead to lapsed licenses.

Relicensing methods

The regulator approves the licensing bases prepared and submitted by a licensee. Relicensing considerations include:

- Establishing a path forward for relicensing that assures that nuclear facilities are either relicensed or spent fuel is relocated to an appropriately licensed facility prior to current license expiration;
- Identifying and addressing technical issues that could compromise safety during extended storage period or subsequent transport;
- Developing ageing management plans to address material degradation, required documentation, records storage, and other institutional controls, etc.;
- Collecting data and conducting research to support license extension applications that provide needed information well in advance of the need to extend the license.

4.3.6. Industry and bordering/transit countries

Industry may need to interact, in a direct or indirect way, with officials from bordering and/or transit countries in order to ensure successful transport and minimize potential for additional constraints, costs, and delays.

Clear task sharing between industry participants long in advance of transport reduces the risks of application process disruption due to omitted steps or incomplete documentation. For example, in the EU, transit through a state or state-to-state transport requires agreement of all competent authorities involved prior to the shipment [8].

Industry may initiate and facilitate inter-governmental agreements between the concerned states to ensure issues of common concern are identified and addressed (see Section 4.3.4).

4.3.7. Industry and industry

The various phases and operational steps in the BEFC may be performed by a number of different entities that require close coordination. To assure safe and effective operations, interactions among industry participants are necessary to address issues related to contractual arrangements, schedule coordination, records management, transfer of ownership, assuring compatibility (tooling, physical geometry, acceptance criteria, etc.), specifying performance requirements, as well as numerous other transfers of information and materials needed.

Interfaces are both direct and indirect because spent fuel along with associated equipment and information are propagated through the various phases of the BEFC. Examples of these industry/industry issues include:

- Coordination between fuel owner and cask vendor concerning cask timing and delivery issues, specification of fuel characteristics for cask design, specification of cask performance requirements , etc.;
- Communication between NPP operator and fuel vendor concerning fuel performance specifications and feedback of fuel performance information;
- Ensuring that thermal loads and limits are clearly understood by both storage facility operator and cask vendor;
- Ensuring that burnup history and records of fuel condition are clearly documented by both NPP operator and storage facility operator and within the storage cask/facility parameters;
- Interfacing of transport carrier and NPP operator with storage, reprocessing, and disposal facility operators as well as many other participants such as governments and municipalities, equipment vendors, other transport industry regulators, and others to ensure:
    - Schedule coordination with shipping and receipt facilities;
    - Equipment compatibility with shipping and receiving facilities;
    - transport routes and associated approvals are properly prepared;
    - Personnel training and qualifications.

To illustrate the need for close coordination among industry participants, consider the case where the reactor cooling pool is nearing its capacity. In this case, the NPP operator must manage very stringent time limits for loading casks to free up storage positions in the cooling pool. Hence, storage casks must be delivered and loaded at, or above, the same pace that spent fuel is removed from the reactor. Consequently, delays in cask availability could result in plant shutdown. To avoid this consequence, careful planning and coordination must be done as well as provision for backup solutions such as maintaining some excess cooling pool and/or storage cask capacity.

Several good practices have been successfully applied to facilitate interactions among various industry participants and phases of the BEFC. These practices can be generally grouped into the following categories. A few examples of good practices are given with each.

Effective planning and preparation:

- Participate in user groups to address common concerns and to share technical information, lessons learned, and solutions;
- To help assure successful transportation, a leading organization responsible for the success of the operation may be designated to coordinate the many participants involved. The industry participants are accountable for the progress of their individual task towards the leading organization; similarly the leading organization

will solely be accountable to the authorities for the progress and success of the transport. Transport coordination services:

- Assure availability of authorized transport routes;
- Confirm compatibility of interfaces between casks, transport conveyances, tools, and equipment;
- Verify operator and carrier organization qualifications, insurance contracts, equipment maintenance, etc.;
- Assign clear responsibilities and time limits to each Industry participant;
- Follow up the implementation of each individual task;
- Organize coordination meetings on a regular basis and keep in close contact with the authorities.

- Evaluate the potential risks and, if necessary, develop procedures or casks tailored to address specific needs of degraded fuels (fuel owner and the NPP operator). For example, in Belgium, fuel owners are working with the transport carrier to reduce structural stresses on casks and cask contents during fuel transfers (i.e. administrative controls during transfer, shock absorbers on casks, accelerometers on cask, etc.);
- Precautions have to be taken in order to avoid the potential degradation of fuel leakers due to extended storage in wet conditions. The licenses of storage casks may, depending on the country, require that stored spent fuel elements must be leak-tight.

Contingency planning:

- Perform contingency planning to mitigate the consequences of potential errors or oversights and to avoid unnecessary costs, delays, and technical difficulties;
- Review past events and other relevant facilities and operations to identify areas where additional effort and/or redundancy as appropriate (e.g. consider multiple vendors to reduce risks associated with cask availability);
- Maintain some reserve capacity in cooling pool and/or on-site storage casks;
- Plan for possible delays in receiving approval;
- Plan alternative routes for transport of spent fuel and high-level waste in order to offer the flexibility to the transport.

Careful coordination and contractual arrangements:

- Hold regular coordination meetings to ensure customer requirements and vendor constraints are clearly understood, to status progress, address emerging issues, and to assure early identification and resolution of issues that may affect the success of the transaction;
- Clearly identify points of contact and decision making authority within each organization;
- Ensure contractual arrangements clearly specify tasks, responsibilities, interfaces, products, quality assurance (QA) requirements, schedules, and all other conditions necessary for a successful transaction;
- Transfer of ownership is a significant activity involving many parties representing the industry, the regulator, and other affected stakeholders. A clearly defined

process and approval authority helps to ensure any change of ownership is properly executed. Transfer of ownership addresses a number of issues such as:

  - Identification and segregation of financial, regulatory, and other legal obligations and liabilities;
  - A complete records package that includes fabrication, operations, and storage history.

- Contractually specify delivery times with sufficient margin to ensure operations are not adversely impacted;
- Fuel characteristics define the cask to be supplied. Jointly developed suitable casks for difficult fuels in order to define the best compromise between the requirements of the fuel owner and the design constraints of the cask vendor.

Execution and follow-up:

- Hold regular debriefing and feedback meetings;
- Thorough follow-up by applying appropriate quality assurance and process controls to ensure contract performance;
- Report off-normal events and other lessons learned that can benefit other industry participants;
- Ensure timely access to relevant industry information and act promptly to take appropriate corrective actions and/or to implement improved practices.

5. PUBLIC CONFIDENCE

Interfaces associated with public interaction were not included directly in the participant–participant matrix. The nature of this interface is primarily social and political rather than technical. Nonetheless, managing interfaces that affect public confidence in safe and effective management of spent fuel and associated radioactive wastes is a key issue that must be proactively addressed. Past experience has shown that failure to effectively identify and address public concerns has resulted in opposition that has caused cost and schedule delays and has hindered the operation of nuclear facilities. Lack of public support and/or active resistance from political organizations and local populations have prevented siting of new spent fuel or waste storage facilities and have even resulted in closure of existing facilities. Conversely, public support can encourage favourable conditions/politics for BEFC nuclear facilities.

Local populations may have traditions and interests that present unique concerns. Perceptions of nuclear facilities vary widely in different areas as does the local influence on policy and industry. Nuclear facilities are perceived as a liability by many due to safety concerns, potential effects on surrounding land values, and increased traffic and industrial activity. In other cases they are perceived as an asset due to their potential for economic benefit, increased employment, energy availability, clean energy production, etc. Additionally, some groups such as native nations or tribal entities may have legal status and rights that differ from other public entities. For example: in United States of America (USA), tribal entities are considered as independent nations, and must be respected as such. Similarly, in Sweden local governments must approve siting of nuclear facilities.

Because of the different local perceptions, political and regulatory frameworks that exist in various Member States, it is difficult to fully identify issues and prescribe solutions to the range of potential public issues. Public concerns and effective methods for their resolution are very much local issues that need to be addressed in context of specific national, regional and local culture and value systems. There are, however, a number of common principles that should be taken into consideration when interacting with public and political organizations.

The timely disclosure of information on any development regarding waste management in general, and also in connection with the project, is a key method in confidence building. Proactive communication of accurate, understandable information is much more effective than attempting to refute erroneous messages.

5.1. PUBLIC INTERACTION WITH THE POLICY MAKER

Availability of an endpoint (direct disposal or reprocessing and disposal) affects the necessary storage capacity and duration. Further, commitment to a clear and achievable path to an endpoint gives the public confidence that interim storage facilities will not become permanent and that spent fuel can be successfully managed. Achieving an acceptable endpoint to the BEFC is of primary importance for the viability of nuclear energy. Rejection of a proven BEFC solution or planned project, facility site, or transportation route generates doubts, and re-enforces the idea that there may be no acceptable BEFC solution. For example, in Canada, Germany, Switzerland, United Kingdom (UK) and the USA new site selection processes have been initiated for high level waste disposal which has a negative impact on public confidence in the ability to achieve an acceptable endpoint. Conversely, identifying and successfully implementing an endpoint is likely to reduce the perceived risk associated with spent fuel and thus facilitate siting of new facilities.

5.2. PUBLIC INTERACTION WITH THE REGULATOR

Public trust in the competency and effectiveness of the regulatory process is essential for achieving and maintaining stability in a nuclear programme. Not in my backyard (NIMBY) and/or not in my term of office (NIMTO) syndrome has resulted in abandonment of many projects even though all regulatory and licensing requirements were satisfied. Trust is easily lost. It is therefore, essential that the nuclear industry and its regulatory infrastructure maintain high standards in all their dealings and that any errors are identified and addressed and communicated in a candid and open manner. Good practices include:

- Maintaining an inspection programme that ensures operational compliance;
- Hiring and retaining staff with direct nuclear experience;
- Ensuring the basis for regulatory decisions is well justified and available to constituents;
- Use of effective methods for sharing information with public and obtaining public input are essential for the regulator to obtain public confidence; For example:

  — Inviting and educating stakeholders in how the regulator works through open meetings;
  — Utilize internet media to communicate easily understandable, timely, and relevant information (e.g. user-friendly web site, Twitter, etc.);
  — Address questions and concerns from the public in a timely and respectful fashion;
  — Select experts considered credible by the public (scientists, technicians, representatives of nuclear control organizations, etc.) and prepare communications such that they are clear and understandable and relate directly to the questions and concerns.

- Aligning national standards and practices with internationally endorsed standards and procedures as well as independent oversight from IAEA or other recognized bodies;
- Explaining and discussing all changes in regulations or guidance with the affected public prior to implementation;
- Establishing a regulatory framework where public concerns are carefully considered for activities that involve new risks or other local impacts but does not needlessly exercise the public for routine activities or other accepted activities (e.g. on-site transfers);
- Balancing transparency/openness with the need to maintain independence and to protect security considerations and proprietary information.

5.3. PUBLIC INTERACTION WITH THE INDUSTRY

Stakeholder acceptance and not necessarily technical considerations will be the basis for decisions regarding site locations. The public is often involved through referendums or other processes to determine how the BEFC operations and funding will be addressed. A lack of public acceptance, a way forward or issues with NIMBY may prevent new facilities being built and thus results in de facto extended storage at existing sites.

Political organizations often try to influence industry decisions through interaction with the policy maker. Political organizations whose purpose is to actively oppose the industry pose a particularly difficult challenge. They may disagree with operations or strategies proposed for the BEFC management. For example, they might debate the safeguards considerations due to plutonium separation in the case of reprocessing or the detrimental consequences for the local population from very long-term storage.

Because there is nearly always some active resistance to nuclear build or transport, mere acceptance by even a large majority of the public is often not sufficient. Providing access to information and facilities is key to effectively communicating with concerned members of the public. Below are several examples that have proven effective:

- At the very early stage of a project development, communicate with the public with the objective of taking into account the needs of the local authorities and population. Assessing and addressing public acceptance issues early in the siting/licensing process helps to avoid unanticipated schedule impacts;
- Consult with the public and local officials to select a site, if possible, in areas with favourable demography; capitalize on economic development/infrastructure attending the facility, etc. Actively engage with the native and local populations to identify and address the impacts the new build or future expansion of nuclear facilities will have on them;
- Methods tailored for addressing each group help to ensure that their core concerns/interests are fully understood and addressed;
- Encourage beneficial development within the community and region:
  - Increased employment and local development opportunities offer a benefit to communities willing to host BEFC facilities;
  - Citizens association and local partnerships may help to encourage mutually beneficial development.

- Financial and/or social incentives aimed at providing long term economic benefit to the community, compensating for the loss of value of neighbouring pieces of land, addressing the increased traffic and noise, etc. have helped to reduce the perceived liability of new build and to provided compensation and lasting benefit for affected communities;
- Negotiate issues and concerns to develop a win-win situation. Both sides should be prepared to make concessions and trade-offs;
- Plan facilities and fuel cycle processes to minimize risk and maximize security;
- Make roles played by industry, government, academia, labour, and local/tribal organizations clear and binding.

5.4. SPECIAL CASES

5.4.1. Policy maker, regulator, industry and regional governments

In many cases, the local public and government are supportive of existing nuclear facilities in or near their community. Local populations receive direct economic benefit and also, often due to their proximity and familiarity with the operation, have a better understanding of the risks. Citizens and organizations in bordering areas, however, often perceive risks without commensurate benefit. This situation may result in regional governments and/or coalitions of surrounding local governments uniting to oppose siting facilities in adjacent or nearby regions. Consequently, it is important for both the regulator and the industry to consider and interact with outlying communities and regional governments who can influence siting and operational requirements.

5.4.2. Policy maker, regulator, industry and independent nations or tribal entities

New facilities for storing or disposing of high level waste may be in competition with the interests of native populations. At issue might be ancestral culture, sacred sites such as ancestral burial grounds, and in some cases land use on reservations.

5.4.3. Transport entities and the public

Public demonstration on transport routes could result in unplanned conditions with the potential for compromising safety and/or security controls. To ensure safe and secure transport, planning should take potential public actions into consideration. Below are some examples of practices that are considered prudent. The participants involved in these interfaces may vary from country to country and even from transport to transport within a country; however, the transport documentation should clearly define the entities responsible for the various transport interfaces:

- Ensure that transport documentation clearly defines the entities responsible for the various transport activities and interfaces;
- Allow time for public in route interactions with the local population;
- Set up the appropriate infrastructure for effective trans-shipment in a secured area. This needs to be prepared a long time in advance of the beginning of transports;
- Have programmes and infrastructure in place to assure that active resistance can be safely managed; without impacting essential operations. For example, security measures may include physical barriers, guard forces, etc.;
- Designate 'Safe Places' along the transport route in order to be sure that whenever a delay is necessary to solve a blockade issue, the materials to be transported are safely guarded;
- Use alternative transport routes when planned transports are jeopardized;
- In some cases, public intervention may be terminated by legal action;
- On-site transfers are relatively unaffected by public interference in many countries because transport in the public sector is not required. However, due to the regulatory framework in Korea and Spain, the public is active even in on-site transfers to AR storage.

Note that many countries keep transport movements confidential, while in other countries transparency is an obligation except for the transport of materials listed in [9] which must remain secret for security reasons.

6. CONCLUSIONS

The process outlined in this report identified many potential issues from the interactions between the phases and among the participants responsible for the phases of the BEFC. When available, solutions based on Member State experience are provided. In addition, examples of approaches employed to resolve the issues are given in the Appendix. This is not intended to represent a complete list.

Key conclusions that can be drawn from the integration tasks are:

- Assuring compatibility of schedules, equipment, and acceptance criteria are key interface issues;
- The biggest uncertainty in successfully integrating the BEFC is the uncertainty relative to the endpoint of the fuel cycle. Stable policy over the long time frames, needed to enable efficient BEFC implementation, is unlikely to occur. Hence, industry should work not only to encourage stable policy but also to develop strategies that include flexibility and contingency to enable success amidst uncertainty;
- The existence and importance of interfaces depend upon the national energy policy, objectives of the nuclear programme, maturity of the nuclear programme, size of the programme, the regulatory framework, the fuel cycle employed, and other country-specific considerations. The issues become different if the whole BEFC is within the control of a single country;
- There are a number of approaches to address an interface issue;
- Effective integration begins early in the planning process. Opportunities are lost if interfaces are not identified and addressed in the early stages of each of the BEFC phases;
- Successful integration requires an understanding of the full life cycle and the willingness to make trade-offs that assure success over all phases of the BEFC rather than optimizing for particular interfaces of the BEFC;
- Record keeping is an important issue for each interface. Without proper records, interface issues might not be able to be addressed or conservative and costly alternative approaches might need to be developed;
- At present, there is limited experience transporting fuels from a dry storage facility to a subsequent dry storage facility. However, as storage periods are extended and countries contemplate consolidation into regional or centralized dry storage facilities, this interface will take on increasing importance; particularly if inspections and/or repackaging are needed to prepare fuels for long term storage;
- Additional pro-active efforts are needed from every participating organization in the BEFC to ensure early attention to public acceptance in the siting, safety, operation, duration, oversight, and path forward. Accurate information must be provided in a user-friendly format.

Lastly, effective management and integration of the various phases of the BEFC is simply sound management. The principles presented in this report do not identify new ideas, rather, they emphasize the importance of systematically identifying and managing interfaces within the BEFC. Because of the complexity of the issues and interfaces, a process is provided to help ensure a rigorous identification of applicable interfaces and consideration of the associated issues and opportunities.

Appendix

INTERFACE ISSUES – MEMBER STATE EXAMPLES

The following examples illustrate practices that have been employed for addressing identified issues. Some examples are also provided to show situations resulting from issues that have not been adequately addressed. In many cases, examples are linked to a specific country. This is not intended to imply that the practice is unique to the country.

1. THE PHASE – PHASE MATRIX (SECTION 3)

1.1. CROSS CUTTING ISSUES (SECTION 3.2)

1.1.1. Interfaces between front-end of the fuel cycle and BEFC (Section 3.2.1)

Changes in the fuel design may limit and/or adversely affect storage, transport and disposal systems and operations:

- In Hungary when applying for new or modified fuel license, one must show that the fuel can be transported, handled at the receiving facility, and stored in the interim storage facility;
- In the UK changes to Advanced Gas Reactor (AGR) fuel design or manufacture are reviewed to ensure compatibility with BEFC handling [Annex II., Section 13.3.4];
- In Switzerland there is a requirement for every new fuel assembly design to demonstrate that the fuel can be either reprocessed (statement from reprocessing company), interim stored (statement of sufficient capacity from interim storage facility) or disposed of (statement from disposal organization) [10];
- In France changes to LWR fuel design were made in order to improve drying, thus mitigating the risks of radiolysis and fuel degradation in storage;
- In Finland new build reactors must have sufficient wet storage capacity to accommodate the projected lifetime spent fuel inventory;
- In France the application of burnup credit, justified by experimental results, has allowed for greater flexibility in spent fuel management;
- In Switzerland the disposal methodology leads to a thermal limit ($\leq$ 1500 W) for spent fuel containers entering a geological repository. Use of high burnup UO_2 as well as MOX spent fuel may lead to some spent fuel containers not being fully loaded.

1.2. MAINTAINING COMPATIBILITY OF FUEL WITH ACCEPTANCE CRITERIA FOR PRESENT AND FUTURE PHASES (SECTION 3.2.2)

Cooling times must be managed to ensure thermal load limits are not exceeded:

- In Japan the wet cooling period has been extended to ensure heat loads during dry storage are below maximum permitted storage temperature limits;
- In the Czech Republic at least one reactor pool must have sufficient reserve capacity to accommodate not only a full reactor core inventory, but also the inventory of one cask;
- In Switzerland the dry storage cask capacity has, in some cases, been reduced to ensure heat loading limits are satisfied;

- In the USA materials and cask designs are being developed with the objective of accommodating the higher heat loads;
- In the United States of America it was expected that utilities would elect to ship fuel to the proposed repository prior to sufficient ageing to meet repository heat loading requirements. As a result, repository plans included an ageing (i.e. temporary storage) facility to accommodate these fuels;
- In Switzerland to maintain the concrete storage pad below temperature limits, cask storage limits may be more restrictive than transport limits. In this case, cask loads are down rated to ensure that they meet the storage requirements;
- In Belgium when heat dissipation from the cask through the gap between cask and transport canopy result in temperature limits being exceeded, solutions include delaying shipment until heat loads are compatible with existing requirements or to redesign transport railcar and/or canopy to improve heat transfer when needed.

1.2.1. Cladding integrity (Section 3.2.2.2)

Cladding integrity is relied upon to maintain confinement and geometry control. Higher burnup fuels and/or longer storage periods may affect cladding performance:

- In Switzerland the regulatory guideline G05 requires that the integrity of the spent fuel cladding is ensured during dry storage. This assurance is provided for each type of cladding based on the determination of maximum allowable temperatures during dry storage to ensure certain stress (< 120 MPa at beginning of storage) & creep (<1% after 60 years of storage) criteria are respected. This information is obtained from the fuel vendors;
- In Japan dual purpose casks are licensed for storage with lower cladding temperature (in order to preclude the possibility of hydride reorientation) limits so they can be licensed for transport (Annex II, Section 4.);
- In many Member States because bare damaged fuel is not allowed to be placed into dry storage, the wet storage facility monitors water quality to confirm that breached fuel rods are identified and packaged into appropriate damaged fuel canisters prior to placement into a dry storage cask;
- IAEA [1] includes a number of techniques for detecting and managing damaged fuel.

1.3. ENSURING COMPATIBILITY OF PACKAGING, EQUIPMENT AND OPERATIONAL CONTROLS BETWEEN SHIPPING AND RECEIVING FACILITIES (SECTION 3.2.3)

When moving spent fuel, packaging and equipment must be compatible with and within the authorized safety basis of both shipping and receiving facilities. In addition, operational controls (e.g. radiological controls):

- In the United States of Americas a lightweight cask was developed to accommodate the crane capacity at a reactor pool. This necessitated the removal of shielding that created radiological dose concerns resulting in an amendment to the operating procedures and license;
- Some reactor facilities were not originally designed to handle current transport casks. As a result, many reactor facilities have been modified;
- Western European countries, spent fuel transportation to reprocessing facilities in France and in the UK were suspended in 1998 because contamination values

exceeding regulatory limits up to several hundred times were measured frequently on both casks and/or trailers. The problem originated from pool water and CRUD being trapped behind trunnions and threaded holes which were not easy to decontaminate. The occurrence of such contaminations has been realized by the industry involved. However, the industry failed to apply and implement corrective measures to prevent such contamination. After thorough analysis of the root cause for the contamination, the Competent Authorities (i.e. the Regulator) required, amongst others, protective coverage to all surfaces during fuel loading/unloading as well as extensive radiation controls; mainly contamination measurements. After implementation of all measures required, inter-governmental agreements were reached to restart transportation in March 2001. Additional measures introduced also included the fitting of trunnions at sub-zero temperatures so that they would further seal the void spaces behind them as they heated to ambient and the operating temperatures after fitting.

1.4. KNOWLEDGE MANAGEMENT AND ADEQUACY OF RECORDS (SECTION 3.2.4)

Records must remain accessible, usable, undamaged, beyond the life of any particular storage system. They must be maintained and transferred with the fuel to its final end state. Records storage technologies continue to evolve. Databases and digital storage poses particular concerns.

Due to the growing complexity of digital objects, a publication now scarcely contains just text, the distinction between library material and archival material is fading. Since the rise of Internet, publications are no longer strictly limited to one specific location. This is also leading to a blurring of the borders between national and international publications. Therefore cooperation in the library sector, at both a national and international level is continuing to grow. International cooperation and knowledge dissemination structures have been set up and are already proving their worth. International organizations (European Commission, United Nations Educational Scientific and Cultural Organization, International Federation of Library Associations and Institutions etc.) have influenced or guided the growth of collaboration among interested stakeholders by providing funds and displaying commitment and determination. Some initiatives are:

- Alliance for Permanent Access (2008–present).The Alliance for Permanent Access (APA) is developing a shared vision and framework for a sustainable virtual infrastructure for permanent access to scientific information (including all disciplines from physical, biological, or environmental sciences, to social sciences and humanities) in Europe [11];
- Open Planets Foundation (2010–present). The Open Planets Foundation (OPF), a consortium of European National Libraries and Vendors, has been established to provide practical solutions and expertise in digital preservation, building on the research and development outputs of a previous Planets project [12];
- E-Depot, National Library of Netherlands, (2000–present). A digital archiving environment that ensures long term access to digital objects (catering mainly for electronic journals) [13];
- DRIVER I (2003–2007) and DRIVER II (2008–present). DRIVER is a multi-phase effort whose vision and primary objective is to establish a cohesive, pan-European infrastructure of Digital Repositories, offering sophisticated functionality services to both researchers and the general public [14];

- KEEP (2009–2012). KEEP (Keeping Emulation Environments Portable) will develop an Emulation Access Platform to enable accurate rendering of both static and dynamic digital objects: text, sound, and image files; multimedia publications, websites, databases, videogames etc. [15];
- CASPAR (2006–2010). Cultural, Artistic and Scientific knowledge for Preservation, Access and Retrieval (CASPAR) was an integrated project co-financed by the European Union [16]. It orchestrated the implementation of Open Archival Information System (OAIS);
- PARSE Insight (2008–2010). A two-year project co-funded by the European Union, concerned with the preservation of digital information in science, from primary data through analysis to the final publications resulting from the research [17].

1.5. TRANSPORT AND STORAGE (SECTION 3.3.2)

- In many countries following transport, it is confirmed at the receiving facility either by direct measurement or by indirect evaluation, that the containment function was not compromised during transport. Extensive experience at La Hague shows that, with appropriate packaging, spent fuel shipments have been received showing the absence of fuel damage due to transport. This applies to shipments by rail, road, and sea over long distances;
- In most countries the licensing periods for storage and transportation casks varies. Japan and Czech Republic for example require their transport casks to maintain a license and be renewed at 5 and 10 year intervals respectively. While in Switzerland the period of storage license is not limited and the transport license is not required to be maintained (renewed) during the storage period. Hence, transport licenses may lapse while casks are in storage and, prior to future transport, licenses may be renewed and/or fuel may be repackaged if needed;
- In Japan the radiological release limits for the storage facility are more restrictive than for transport. Spent fuel is loaded into the dual-purpose metal cask at reactors and transported to an AFR dry storage facility. The cask is sealed with a metal gasket providing containment that may be compromised by mechanical vibration during transport. Although the dual purpose metal cask could conform to the transport regulations with respect to the radiological release (containment function), the metal cask might not conform to the storage requirement with respect to a radiological release limit for storage.

2. THE PARTICIPANT – PARTICIPANT MATRIX (SECTION 4)

2.1. REGULATOR AND POLICY MAKER (SECTION 4.3.1)

- In Sweden policy decisions are based on technical studies often performed by or under the direction of the regulator.

2.2. INDUSTRY AND POLICY MAKER (SECTION 4.3.2)

Several examples of the impacts of changed or failed policies exist (see below). Examples include:

- In the United States of America a change in government has led to a delay with the plans to provide a repository, this will have a significant impact on spent fuel storage capacity and duration at storage sites. As a result of failure to implement a policy for spent fuel reprocessing and/or disposal, many countries have created and/or plan to expand on-site storage facilities to accommodate the need for additional storage;

- In Switzerland, due to a change in the nuclear energy law, spent fuel elements may not be exported for reprocessing for a period of 10 years, commencing 1 July, 2006. As a consequence, additional interim dry storage and additional wet storage was required. In Switzerland, license for storage facilities is given by government (i.e. policy maker) not the regulator;
- In France in compliance with a 2006 French law, an inter-governmental agreement must be reached before reprocessing may be implemented. The agreement must address the overall time schedule of spent fuel reprocessing, transportation of related waste, and the projected use of recovered fissile materials. Experience shows that negotiation of such agreements is a long process which, in some cases, leads to an impasse that precludes acceptance of the spent fuel for reprocessing;
- In Eastern European countries, as a consequence of policy changes resulting from the breakup of the former Soviet Union, additional spent fuel storage facilities have been required to manage spent fuel stocks either as an interim measure or to remove the reliance on third parties:
  - In Hungary a decision to build dry storage capacity in the 1990s was necessary because at that time the Russian Federation no longer accepted Hungarian fuel for reprocessing;
  - In Slovakia up to 1987 all the WWER-440 spent fuel was stored for 3 years in the deactivation pools of the NPPs prior to transport to the Soviet Union for reprocessing. This acceptance criterion was later changed to a ten years cooling requirement prior to transport. As a result an intermediate wet storage facility had therefore, to be built on the site of Bohunice power plant for the needs of the WWER-440 units;
  - In the Czech Republic storage capacities had to be developed in the early 1990s as a result of the former Czechoslovakia being split. The original strategy in the former Czechoslovakia for the BEFC was based on a contract covering the transport of spent fuel back to the former Soviet Union where the fresh fuel had been purchased. To comply with a five year cooling requirement for spent WWER-440 fuel being transported to the Soviet Union a centralized ISFSF at Jaslovské Bohunice NPP site was used for the whole of Czechoslovakia. With the split of Czechoslovakia (31 December 1992) this facility became under the governance of the newly formed Slovakia Republic. Spent fuel that had been irradiated in the now Czech Republic was later returned to the country of ownership once a new interim storage facility at Dukovany NPP had been built.

3. REGULATOR AND REGULATOR (SECTION 4.3.3)

All Member States should strive for harmonization of requirements for transport and for storage. This is essential for a shared and understandable approach of spent fuel management and of safety issues, for smooth cross-border transport, and for public acceptance.

- The IAEA is coordinating integration of regulations and guidance concerning the re-licensing of casks with lapsed certificates of approval B(U)F after several years of storage (e.g. definition of requirements for records on casks history, non-destructive testing, destructive investigation of components of a duly selected representative cask model, etc.);

- In the Unites States of America, the US Nuclear Regulatory Commission (NRC) undertook a task of evaluating the current state of integration of the regulations for the BEFC. This included ties of the BEFC to reactor operations, and fuel manufacturing since these phases, although outside the BEFC, may provide necessary input:
  - The various phases of the BEFC were identified;
  - The regulations for each phase were evaluated to look for inputs required from previous phases. Examples might include fuel parameters, records, fuel condition, and fuel and container degradation;
  - Regulations from previous phases were examined to determine if the required information was available if the regulation was implemented. If not a recommendation to modify or clarify the regulation was made;
  - To date it has been determined that the US regulations are flexible and complete enough to meet any BEFC path determined by the policy makers. There are some regulations that do appear to need clarification. The process of evaluating whether the guidance is adequate to implement the regulations in an integrated fashion, especially in light of the uncertainty of the ultimate disposition path is just beginning.

- In the Czech Republic, a holistic approach is followed for integrating issues with transport and storage licenses (design approvals) for spent fuel casks. This approach, developed in the mid-1990s, together with storage facility and transportation licensing procedures, represents a comprehensive system for licensing all spent fuel management activities that includes the full life cycle of the facility. At the end of the facility lifetime the regulatory body issues a decommissioning license for either dismantling of the whole facility or its reuse for other purposes;
- In Japan, a holistic approach has been developed that integrates transport and storage [18], [19], [20], [21]. Traditionally, spent fuel transport and storage of dual purpose casks have been examined and licensed separately for different licensing periods. However, both the transport and storage could be examined and licensed in a holistic manner, by mutually using transport and storage safety records. The holistic approach assumes that spent fuel is firstly transported from the power plant to a storage facility, stored for 40 to 60 years, and then transported to a reprocessing plant. Thus, the integrated operation from transport before storage until the transport after the storage could be licensed in a holistic manner as illustrated in Fig. 2.

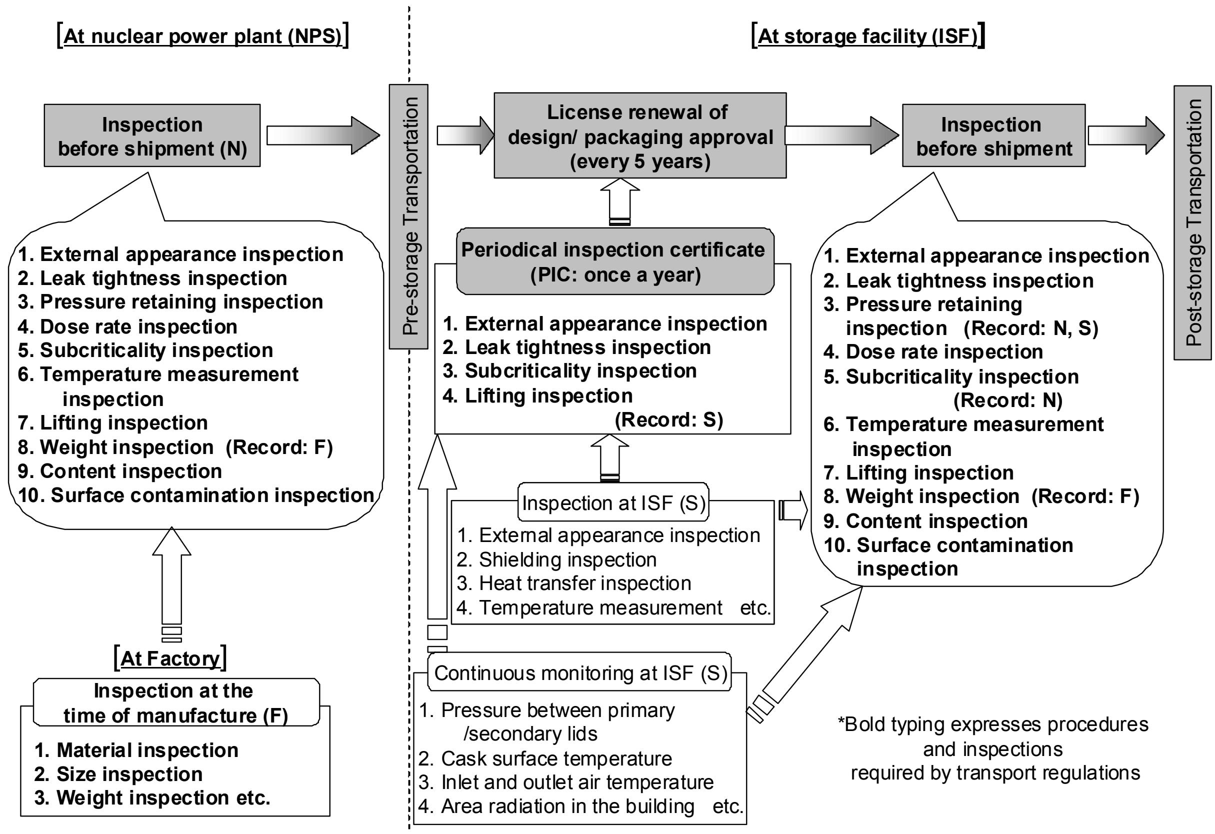

FIG. 2. Schematic of a series of investigations required for dual-purpose casks (from the viewpoint of transport).

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DEFINITIONS

AR	At reactor storage. Storage facility at the location of an operating reactor that is governed by site operating procedures. N.B. this differs from the definition used in other IAEA publications.
AFR	Away from Reactor. Storage facilities that are not at-reactor and for which transport to and from is governed by external transport regulations. N.B. this differs from the definition used in other IAEA publications.
BEFC	Back End of the Fuel Cycle.The portion of the nuclear fuel life cycle that assures safe management of nuclear fuel after it is withdrawn from the reactor. Phases in the BEFC may include storage, packaging, transportation, reprocessing, recycling, and disposal.
Damaged fuel	Spent fuel that requires non-standard handling to comply with applicable safety, regulatory, or operational requirements.
Disposal	Emplacement of spent fuel or waste in an approved location without the intention of retrieval.
FEFC	Front End of the Fuel Cycle. The portion of the nuclear fuel life cycle comprising the facilities that manufacture and irradiate (reactor operations) the fuel thus producing spent fuel.
Interaction	An exchange of information and/or materials between participants and/or phases in the BEFC. The objective is to ensure co-operative and coordinated resource utilization in pursuit of safe and effective spent fuel management. Interactions occur at interfaces.
Interface	A boundary between phases and/or participants in the BEFC where a common need, concern, or interest of one phase or participant influences another in order to increase effectiveness of spent fuel management. Essential interfaces are those which, if improperly managed, could result in a 1) plant or facility shutdown, 2) an unsafe or unsecured situation, 3) insufficient resources, or 4) an unusable product.
Operator	NPP, Storage Facility, Reprocessing Plant, and Disposal Facility. Any organization responsible for the safety of one or more phases of the nuclear fuel cycle. Different entities may be responsible for operations associated with the different phases of the nuclear fuel cycle.
Phase	A step in the nuclear fuel cycle that changes the package configuration or physical location of spent fuel.
Participant	The management entities and stakeholders associated with the phases (i.e. individuals, organizations, institutions, etc.) that must cooperate to achieve effective spent fuel management.
Policy maker	The entity (or entities) that establishes national policy and associated laws governing nuclear materials.

Public/Political organizations	Members of the general public who organize to achieve an objective and/or influence policy relative to management of nuclear materials. Political organizations may include local citizen groups, political parties, multi-national organizations (e.g. Greenpeace).
Reprocessing	The process allowing the recovery of remaining fissile material from spent fuel, and reducing the volume of high-level waste. In addition, it allows the sorting and conditioning of waste in an appropriate way.
Spent fuel storage	Retention of spent fuel in a facility that provides for its containment, with the intention of retrieval. Spent fuel storage maintains spent fuel in a safe condition pending subsequent phases in the BEFC. Dry storage refers to storage in a gaseous environment, such as air or an inert gas. Wet storage refers to storage in water or other liquid environment.
Transfer	Relocation of spent fuel to a different storage facility within the site boundary of a nuclear facility. Transfers of spent fuel to AR storage facilities are normally governed by site operating procedures.
Transport	Relocation of spent fuel from one nuclear facility to another. The transport step includes loading operations at the consignor's facility and unloading operations at the consignee's facility. As opposed to transfers of spent fuel which are governed by site operating procedures, transport of spent fuel is governed by transport regulations. Transport considerations may depend on the age of the spent fuel and mode of transport. These considerations may include: • Pre-Storage Transport – Transport that occurs prior to an extended storage period; • Post-Storage Transport – Transport of spent fuel subsequent to an extended storage period (during which ageing processes may have compromised properties of fuel or fuel packaging); • Intermodal Transport – Transport that requires one or more transitions from one mode of transport (road, rail, marine or air) to another before the ultimate delivery of the cask.

ABBREVIATIONS

AGR	Advanced gas cooled reactor
CASPAR	Cultural, artistic and scientific knowledge for preservation, access and retrieval
CRP	Coordinated research programme
HLW	High level waste
HWR	Heavy water reactor
ISFSI	Independent Spent Fuel Storage Installation
KEEP	Keeping emulation environments portable
LWR	Light water reactor
MOX	Mixed oxide fuel
NIMBY	Not in my backyard
NIMTO	Not in my term of office
NPP	Nuclear power plant
OEF	Operating experience feedback
OPF	Open planets foundation
QA	Quality assurance
SFM	Spent fuel management
SRA	Stress relieved and annealed
TM	Technical meeting
WWER	Russian type of PWR (wodo-wodyanoi energetichecki reactor)

ANNEXES

PAPERS PRESENTED AT THE IAEA TECHNICAL MEETING OF INTERFACE ISSUES FOR THE BACK-END OF THE FUEL CYCLE VIENNA, AUSTRIA 3 – 6 NOVEMBER 2009

Annex I

CHAIRMAN'S SUMMARY OF THE IAEA TECHNICAL MEETING OF INTERFACE ISSUES FOR THE BACK-END OF THE FUEL CYCLE VIENNA, AUSTRIA, 3 – 6 NOVEMBER 2009

Robert Einziger

10 December 2009

A technical meeting chaired by R. Einziger (USNRC) on Potential Interfaces Issues in Spent Fuel Management was held on 3–6 November 2009, in Vienna, Austria. There were representatives from 12 countries that are currently generating nuclear power. Nineteen (19) papers were presented covering various countries spent fuel programmes and interface issues.

Hans Forsström (IAEA) indicated that long term storage was a major issue, particularly: 1) monitoring fuel behaviour; 2) fuel removal from casks/ponds; 3) removal of damaged fuel; 4) repackaging. Mr Forsström also indicated that approximately 60 new nations have indicated an interest in obtaining nuclear power capabilities.

Mr Buday (Hungary) indicated that ageing management, expansion of the interface matrices, and prioritization of issues should be considered in the breakout sessions. A set of four Japanese papers addressed technical issues at the interfaces. An interesting presentation was related to transport vibration degradation of cask seals due to relocation, and degradation testing applicable to long term dry storage. Mr Lee (Korea, Republic of) indicated that Korea will be carrying out a demonstration but the purpose and type of demonstration is still evolving. Korea expects only a 20-30 year storage programme. Other interface issues identified by the speakers were:

- Confidentiality vs. openness;
- Storage at decommissioned plants;
- Interface between the front and back end of the fuel cycle;
- Political parties as stakeholders;
- Interfaces that are not direct relationships.

The majority of the meeting was devoted to breakout groups to discuss the interface matrices generated at the consultancy meeting held in 2008 (participant–participant, phase–phase, and participant–phase). These breakout groups identified the important interfaces, issues with these interfaces, and both successes and failures in dealing with the issues.

All the groups generally used the same process during the evaluations. They were:

- Clarify terms;
- Add, delete, or subdivide phases and participants;
- Identify issues;
- Determine the most important issues;
- Identify and elaborate, if possible, on the ways the interaction was handled in their country.

While all three groups started with the matrices, all three groups eventually developed another spreadsheet format to capture the data.

Mr Demazy's group on participant–participant interfaces introduced a number of new participants such as “political organizations” including lobbies, “bordering and transit countries”, and the back-end receiver” as distinguished from facility operators. They decided not to deal with “financial organizations” because they had a hard time defining the term, but they did recognize it as important.

Mr Saegusa’s group worked on the participant–phase matrix. They added a new phase “programmatic” and subdivided the reactor storage into “wet” and “dry” phases. They also included a phase “at-reactor transfer”. The term “government” was changed to “policy maker”. A difference with the other groups is that this group identified the interactions as directional. The participant could influence the phase but not the other way around.

Mr Standring’s group evaluated phase–phase interfaces. By looking at the individual steps in the nuclear fuel cycle and adding “at-reactor transfer” as a phase, they expanded the matrix into 25 different phases. Some of the phases were very specific to a particular nuclear power plant (NPP). Of particular interest was the addition of the “fuel designer/manufacturer” and the “power generator” since this connects the front and back end and allows a country either designing a new system, or just starting out to consider the whole fuel cycle in their decision process. They drew a number of conclusions related to the effects of up front decisions, effects of the way damaged fuel was managed, records management, and most importantly the uncertainty caused by the unknown path forward.

At the close of the meeting, each group leader gave a summary to the whole TM and comments were solicited. Based on these comments and the reports of the group leaders, the chair suggested that the interface matrices treat three groups of issues: 1) operations; 2) safety; 3) security. The information collected at the TM was used at the follow-up consultancy meeting.

Annex II

THE CURRENT APPROACH OF INTERFACE ISSUES IN SPENT FUEL MANAGEMENT IN BELGIUM

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SYNATOM
Belgium

II–1. INTRODUCTION

Apart from the nuclear reactors operated by the Belgian Nuclear Research Centre of Mol, the first three Belgian NPPs started up in 1975, two of them on the Doel site, in the northern half of the country and one on the Tihange site, in the southern part.

Four other NPPs were started up between 1982–1985, totaling 7 units, four on the Doel site and three on the Tihange site. All these units are operated by ELECTRABEL, the Belgian main electricity producer, owned by the French company GDF–SUEZ.

A brief presentation of the other main stakeholders is necessary for the understanding of the interfaces in the field of SF management in Belgium.

SYNATOM is a specialized subsidiary of ELECTRABEL: its scope of responsibilities covers both the front-end and back-end of the fuel cycle; by the delivery of enriched uranium or Pu recovered from reprocessing to its mother company ELECTRABEL and by the management of burnt fuel.

Other responsibilities have been entrusted to SYNATOM by a law passed in 2003 according to which the financial provisions for the dismantling of the NPPs and for the management of the fuel cycle have to be managed by the company.

TRACTEBEL Engineering (TE, is a subsidiary and the engineering office of ELECTRABEL.

TRANSNUBEL is the Belgian leader in the transport of radioactive material that provides: On-site assistance; expertise in packaging licensing, transport of radioactive goods and related logistics.

ONDRAF/NIRAS is a public body responsible for the management of radioactive waste in Belgium including waste treatment, storage and ultimate disposal under the supervision of the minister in charge of energy. ONDRAF/NIRAS pursues R&D activities in view of final disposal of conditioned waste pertaining to categories A, B and C. All the activities of ONDRAF/NIRAS are fully covered by contributions paid by the waste producers, either based on agreed upon apportionment keys or included in the cost price applied whenever wastes ownership is transferred by the producer to ONDRAF/NIRAS.

FANC is the Federal Agency for Nuclear Control. The main responsibilities of the agency with respect to spent fuel management comprise:

- The deliverance of the transport agreements on the basis of applications duly filed by the casks vendors and the verification that the transport complies with all safety requirements;

- The completion of the regulatory framework in view of wastes final disposal and the general aspects of waste management in collaboration with ONDRAF/NIRAS;
- The follow-up and the development of regulations at the national and international levels.

Other companies interface, the scope of their activities will be explained later on in the paper.

The target of this paper is to show how those companies and agencies interact in the various faces of SF management and how possible issues are treated.

II–2. SPENT NUCLEAR FUEL STORAGE

Organization

For historical reasons, two different policies have been implemented for spent fuel centralized storage on the NPP sites; storage in heavy dry metallic casks was chosen on the Doel site while storage under water in a centralized pond was the preferred option on the Tihange site. Both centralized storage facilities are away-from-reactor, but on the NPP site.

SYNATOM is responsible and takes care for the management of spent fuel once declared definitively unloaded by the reactor operator.

Dry storage on the Doel site

The safety report of the dry storage facility requires that the casks must be licensed for transportation, even though off-site transportation is not anticipated now.

The reactor operator ELECTRABEL defines its needs in terms of transfers of SF from the AR cooling pool for an operational period of 10 years. On the basis of that requirement, the characteristics of the SF assemblies to be transferred (spent fuel length, initial enrichment, decay time, burnup, etc.) and the characteristics of the lifting equipment of the concerned unit SYNATOM with the assistance of the TE, define the specifications to be included in the request for tender.

After consultation of the tenderers, comparison of the offers and conclusion of the orders the follow-up of cask fabrication is entrusted to TE until the final delivery on the site. SYNATOM is the owner of the casks.

The order requires that the cask vendor has the obligation to have its cask model duly approved by the licensing authorities at its delivery on the site. The agreement granted by the French and Belgian authorities has a limited validity period: in the past, two or three years; more often five years now. After that period the agreement has to be extended for an additional period.

The loading patterns are defined by TE which also verifies that all the SF characteristics cope with the limiting parameters indicated in the cask agreement and that the anticipated dose rates comply with the transport regulations.

Cask loading is carried out by the reactor operator with the assistance of handling operations specialists of TE.

Loaded cask transfer to the centralized cask storage is performed by TRANSNUBEL, owner of the transport license on the Doel site. After cask positioning in the storage area, the monitoring equipment is connected for continuous follow-up of the helium pressure in the primary lid inter-gaskets volume.

Tihange site

A transfer package has been purchased by SYNATOM for the SF transfers between each AR pool and the centralized storage pool. Transfers are undertaken in wet conditions, i.e with the package partly filled with water.

The annual transfer programme is specified by the reactor operator TRANSNUBEL, the carrier defines the loading patterns on the basis of lists of SF assemblies eligible for the transfers proposed by SYNATOM. TRANSNUBEL also verifies that the SF assemblies' characteristics comply with the requirements of the transfer authorization granted by the FANC.

The loading patterns are transmitted, for information, to the FANC.

The transfer to the storage pond and the packaging unloading are operated by the reactor operator with the assistance of TRANSNUBEL and TE.

II–3. POTENTIAL INTERFACE ISSUES AND THEIR TREATMENT

Many interfaces can only be effectively managed if the scope of responsibilities for each stakeholder is precisely defined and if each of them makes every possible endeavor in timely and high quality delivery.

Some difficulties have occurred in the past which have led to the contacts between stakeholders reinforced.

Meetings managed by SYNATOM are organized every two to three months between the reactor operator (ELECTRABEL), the cask vendor, the fuel and casks owner (SYNATOM), the engineering office (TE) and the cask carrier (TRANSNUBEL) during which any possible problem is discussed and remedies are decided. Such problems can be:

- A change in the loading program issued by the nuclear operator;
- Unexpected delays in the fabrication of the casks;
- Difficulties in the use of a cask or an equipment that need to give rise to corrective actions and lessons for the future;
- Treatment of the consequences of incidents that occurred in the use of an equipment;
- Possible difficulties in the treatment of new requirements received from the licensing authorities;
- Tests to be organized and their follow-up, distribution of roles and responsibilities;
- Evolving interpretation of articles of the transport regulations.

Important efforts are paid to get the information in real time in order to take any (corrective) action; so the consequences of deviations from the programme or any unexpected event can be minimized.

Other meetings are organized twice a year between the same participants and the licensing authorities (FANC). The aim of those meetings is:

- To identify all the expectations from the authorities' side and from the operators' side during the coming six to twelve months;
- To inform the authorities of any difficulty or of the methodology proposed in the treatment of specific questions asked by the authorities in the framework of the extension of agreements;
- To inform the authorities of the progress on specific actions requested by them;
- To share any information useful in the activities related to fuel and casks transfers.

II–4. SPENT FUEL TRANSPORTATION

Organization

Fuel transfers to the reprocessing site were organized until the late nineties in the framework of reprocessing contracts concluded between SYNATOM and the French company AREVA.

The actors, the regulations, the transport means evolved a lot over time; should transports have to resume now the overall organization in place ten to twenty years ago would be reviewed in coordination with the authorities.

Potential Interface Issues and their Treatment

In order to minimize risks and to avoid interface difficulties the complete transport responsibility from the power station to the reprocessing site was in the nuclear carrier's hands.

II–5. SPENT FUEL REPROCESSING AND/OR DISPOSAL

Organization

Reprocessing contracts were agreed in the seventies by SYNATOM for an overall quantity of 672 tU. The SF assemblies has been transported and reprocessed. The recovered valuable materials have been recycled and the repatriation of conditioned wastes started in 2000 and should be finished in 2015.

The present chapter will be devoted to a short description of the organization for the repatriation of residues as this involves the largest number of stakeholders.

The reprocessor (AREVA) sends the residues specification to SYNATOM for approval two years prior to transport; as soon as it has been approved by the French Safety Authorities. The approval has to be granted by ONDRAF/NIRAS, the agency responsible for waste management in Belgium. During that period frequent interactions between ONDRAF/NIRAS and AREVA are arranged by SYNATOM in view of having all questions of the agency duly answered by AREVA. The study of the specification can lead to additional acceptance criteria if it is anticipated that some characteristics of the residues could fall outside the safety analysis for the storage facility or design criteria of the disposal facility.

Concurrently the organization for the transport back to Belgium is initiated. This implies the choice of the most appropriate transport cask, the definition of the slots compatible with the requirements of each stakeholder (reprocessor, cask carrier, ONDRAF/NIRAS and its industrial operator BELGOPROCESS). Other stakeholders also exist and need to be consulted: the federal police, the ministry of interior affairs, the governors and mayors of crossed departments and communes, the representatives of local partnerships, the administrator of the railway network.

The full responsibility for the transport is placed with TN International, a subsidiary of AREVA. TNI delegates some missions to a nuclear carrier for the road transport from La Hague to the closest railway station (Valognes), to the SNCF for the transport between Valognes and the Mol station (located at some kilometers of the storage site) and to TRANSNUBEL for the transport between the Mol station and the storage site.

TRANSNUBEL, acting on behalf of TN International, is also responsible for the contacts with the local authorities, the FANC, the federal police.

When a transport is ready for completion all stakeholders are informed again and the transport can take place. Press releases and local press conference allow information to the public.

Potential Interface Issues and their Treatment

No technical problem worth to be reported ever prevented a transport to be completed.

Some difficulties arose before the very first transport operations; i.e. when an opposing NGO filed a first legal complaint against ONDRAF/NIRAS and a second one against the Belgian State, SYNATOM, TRANSNUBEL and the National Railway Company in the days prior to the departure of the convoy from La Hague. The transportation fortunately was not delayed.

Before each departure from the Mol station to the storage site the stakeholders give the opportunity to some local opponents to spread their message in front of press media. The help of the local police forces was in some cases necessary to free the way, but it never gave rise to violent demonstrations.

Before a transport can start several authorizations need to be granted by the French and by the Belgian authorities, sometimes these are delivered just in time.

Interfaces with the control authorities at each intermodal point have never given rise to any problem.

On arrival of the convoy at the storage facility the cask is unloaded and emptied by BELGOPROCESS. Each waste drum is checked for identification (the documentation attached to each drum has been analyzed a long time before). Additional measurements are performed on a limited number of drums by the teams of ONDRAF/NIRAS and BELGOPROCESS, before the reception of the whole consignment can be attested.

II–6. CONCLUSIONS

The success of such complicated operations is the result of the combination of several factors:

- The roles and responsibilities of each technical stakeholder should be clearly defined;

- One of them should play the role of the conductor, accepted by the group, sharing the tasks, following the actions and sending in due time the necessary action reminders;
- Each stakeholder should be aware that he is the necessary link in a chain that may not be disrupted;
- Procedures and contracts should be clear and exhaustive in the definition of the responsibilities and actions to be undertaken.

Information in real time to the other stakeholders is necessary whenever unexpected events occur.

Annex III

OVERVIEW OF SPENT FUEL MANAGEMENT FOR THE NUCLEAR FUEL CYCLE IN CHINA

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III–1. THE PRESENT POLICY AND STATUS OF NUCLEAR ENERGY DEVELOPMENT AND SPENT FUEL MANAGEMENT IN CHINA

With the economic development and population growth in China, the energy demands are growing rapidly. The energy structure which mainly relies on fossil fuel resources is not sustainable. In order to enlarge energy supply and reduce emission of pollutants and greenhouse gases, the Chinese government is making a very important decision on the energy strategy option for adjusting energy structure and ensuring the security of energy supply.

Nuclear power is the only economic, safe and clean energy which can extensively replace fossil fuels. The Chinese government have defined a nuclear energy development strategy and established a mid and long term development programme for nuclear power in China (2005–2020); this comprises a three staged approach moving from thermal reactor; to fast reactor and fusion reactor. According to the programme, the installed capacity of operating NPPs will reach 40 GWe, accounting for 4% of the total installed capacity in China, and the installed capacity of NPPs under construction will reach 18Gwe (Table III–1).

The development of an fast breeder reactor (FBR) follows the route of experimental FBR, prototype demonstration FBR, and commercial FBR. According to the country's development programme proposal, experimental FBR will be completed and operated in 2009, prototype demonstration FBR will be completed around 2020; the first commercial FBR will be completed around 2035. The batch construction of NPPs using FBR will be realized around 2040.

The Chinese government has already defined the country's spent fuel management policy which is following closed nuclear fuel cycle route through reprocessing to make full use of uranium resources; so as to protect the environment, minimize nuclear waste and ensure the healthy and sustainable development of nuclear power in the country.

At present, the accumulated quantity of spent fuel discharged from reactors in China is 1300 tHM, most of which is stored in at-reactor storage pools, a small amount has been transported to away-from-reactor (AFR) storage pools. The spent fuels from CANDU reactors are stored in a dry storage facility. It is estimated that the accumulated quantity of spent fuel in 2020 will reach 7500 tHM, and in 2025, 13 000 tHM. Currently, the capacity of the AFR storage pool is 500 tHM, but is being expanded to 1300 tHM.

Therefore, research, engineering and construction works in this field are making headway in China. At present, the power reactor spent fuel reprocessing pilot plant has been completed, uranium cold tests have been completed and the hot tests will be performed at the end of 2009. The preparatory works for the 800 tHM large-scale reprocessing plant are underway,

TABLE III–1. NUCLEAR POWER PLANTS UNDER CONSTRUCTION IN CHINA

Operator	Serial number	Project	Technology	Total units	Commercial operation
China National Nuclear Corporation (CNNC)	1	Nuclear Power Quinshan Joint Venture Company Phase II	CNP600	2	2011, 2012
	2	Quinshan Nuclear Power Company Phase II	G2+	2	2013, 2014
	3	Fuqing NPP	G2+	6	2013
	4	Sanmen Nuclear Power Company	AP1000	2 (6)	2013, 2014
	5	Hainan Nuclear Power Company	CNP600	2	2014
	6	Cashima NPP Phase 2	CNP300	1	2011
China General Nuclear Power Group (CGNPG)	7	Ling Ao NPP Phase II	G2+	2	2010, 2011
	8	Liao Ling Hongyanhe NPP	G2+	4	2012, 2014
	9	Ningde NPP	G2+	4	2012
	10	Yangijang NPP	G2+	6	2013
	11	Taishan NPP	EPR(1700)	2 (4)	2013
China Power Investment Corporation (CPI)	12	Shandong Haiyang NPP	AP1000	4	2014
Huaneng Power International	13	Huaneng Shandong Shidao Wan Nuclear Power Company	HTR	1	2013

the capacity of its storage pools will be 6000 tHM (first phase 3000 tHM, second phase 3000 tHM) and the plant is planned to be completed and operated around 2025.

For high level waste (HLW), China has adopted the route of centralized geological disposal. The Chinese government has prepared an R&D guide for HLW geological disposal, this defines development goals in three steps, i.e. laboratory R&D and repository siting (2006–2020), underground tests (2021–2040), prototype repository demonstration and repository construction (2041-the mid of this century). The aim is to complete the HLW geological repository in the middle of this century. Preliminary studies in the fields of disposal geology, geology chemistry, engineering and safety evaluation have been performed; emphasis has been placed on studies of the lithosphere stability, tectonic and seism geologic characteristics, hydrogeological and engineering geological conditions of the northwest pre-selection area (i.e. the pre-selection area of BeiShan in GanSu Province). Research and test device modelling of the geological disposal chemical environment has been established through studying the chemical behaviour of key radionuclides in samples from BeiShan. A conceptual repository design and underground laboratory have been completed; the buffer/backfill materials have been studied and determined, and preliminary exploration of the heat-water-force coupling phenomena of the repository has been performed.

For intermediate low level waste (ILW), China has adopted the technical route of regional near surface disposal. The planned ILW disposal sites in China are: northwest disposal site, southern China disposal site, southwest disposal site, eastern China disposal site and northeast disposal site. The northwest disposal site and southern China disposal site have been completed and operated, the southwest disposal site has entered into design phase; the sitting of the eastern China disposal site has been completed.

III–2. PRELIMINARY VIEW ON POTENTIAL INTERFACE OF SPENT FUEL MANAGEMENT IN CHINA

III–2.1. Spent fuel storage and transport interface

As the burnup of spent fuel steadily increases, in order to balance transport cask manufacturing costs, transport cask loading capacity, construction costs of the reprocessing facility and ease of reprocessing, it is necessary according to the decay of the short-lived nuclides in spent fuel and the reduction rule of the decay heat (Fig. III–1), taking into account the factors such as recycling nuclear materials from reprocessing as soon as possible, it is proposed that spent fuel should be transported to centralized AFR pool storage at the reprocessing plant after 5–8 years cooling. The spent fuels are transported by rail and by sea as much as possible and only by road for short distances.

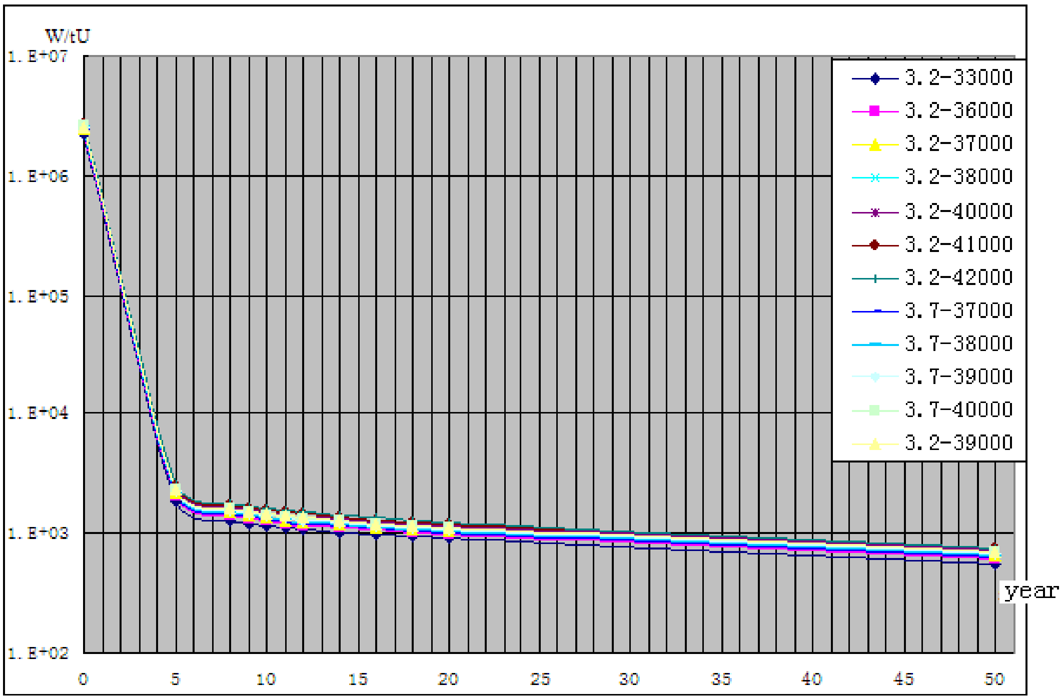

FIG. III–1. Decay heat of spent fuel of different ^{235}U enrichment and burnup versus cooling time (years).

III–2.2. Interface between spent fuel reprocessing and MOX fuel

Spent fuel reprocessing is a key link and key technology of closed nuclear fuel cycle. Advanced reprocessing process should aim at economic, safety and waste minimization, while taking into account U, Pu separation from MA. According to the development of reprocessing technology and Pu recycling mode, the following three options are considered:

- Spent fuels from thermal reactors are reprocessed using proven PUREX process. Separated and purified U and Pu is manufactured into MOX fuel for recycling in thermal reactors, the MA and FP enter into HLW stream and are vitrified. In this way, 30% natural U can be economized, while the volume of HLW for geological disposal can decrease to 1/4 compared to once-through, but the U resources saving is limited （about 30%）through U and Pu recycling in thermal reactors, and since the long-lived nuclides such as MA are not separated from the HLW the radiotoxicity is only reduced by one order of magnitude; the reduction of the wastes which require long time isolation is also limited. This is the universal technical route for reprocessing plant and recovered U and Pu recycling;
- Spent fuels from thermal reactors are reprocessed using an advanced integrated process of MA full separation; separated Pu enters into FBR cycle, and separated MA is burnt in a FBR or ADS. The advantage of Pu recycling in an FBR is that it greatly (about by 60 times） increases U resources utilization and the wastes are minimized; radiotoxicity is also reduced by two orders of magnitude. But this needs to be prepared over a long period of long time, firstly, commercialization of FBR technology in China will take a long time, secondly, a decision needs to be made whether to adopt breeding recycling or burning recycling (require multiple recycling), both FBR spent fuels and ADS irradiated targets need to be reprocessed, reprocessing technology for FBR spent fuel is not yet mature, and reprocessing of the ADS irradiated targets is a new subject;
- Spent fuels from thermal reactors are reprocessed using PUREX as the main process and an auxiliary process is used to separate the HLW stream. Separated Pu is first recycled in thermal reactors, then used in the FBR cycle when available, separated MAs are first temporarily stored in the form of oxide product, the plan is to eventually burn them in a FBR or ADS. This option is realistic, the Pu can be used in the midterm, leads to better U utilization and wastes are minimized in the long term.

Based on the above analysis, option 1 is planned for the technical route for the reprocessing plant to be completed around 2025 in China; the U-Pu co-precipitation management will be adopted. The MOX fabrication plant will be constructed at the same time, this way the formation of ^{241}Am due to decay of ^{241}Pu can be avoided (the time interval between the two plants should less than 3 years). This also avoids costs associated with removing ^{241}Am before Pu recycling. The separated Pu can timely be manufactured into MOX fuel for use in thermal reactors or FBR. The option 2 or option 3 can be adopted in subsequently constructed reprocessing plant according to R&D progress.

III–2.3. Interface between spent fuel reprocessing and radioactive waste disposal

At present spent fuel from CANDU reactors is first stored and their further processing and disposal route are yet to be defined; the spent fuels from other types of reactors has to be reprocessed. As the above analysis has illustrated, advanced reprocessing technology can greatly reduce the volume and radiotoxicity of the waste for geological disposal while the disposal density can be increased, thus the capacity and costs of repository can be saved, the containment and surveillance time after operation can be reduced, and the burden for future human resources can be reduced. Therefore, the interface with the wastes from reprocessing, including waste acceptance, conditioning and disposal methods before disposal should be considered in the construction of the repository.

III–2.4. Interface between advanced nuclear energy technology and nuclear fuel cycle technology

Advanced nuclear energy technology and nuclear fuel cycle technology is a complete engineered system. It requires uniform planning and implementation, top design and systematic planning by the government; the parts of the whole system should be developed in a coordinated and matched way. The sustainable development of the nuclear fission energy involves key technology at three levels:

- Improve and increase the level of thermal reactor nuclear energy to developed from “GEN I” to “GEN III” technology;
- Develop FBR nuclear energy system to optimize the use of U resources;
- Develop nuclear fuel cycle (including wet and dry reprocessing), waste treatment and disposal technology (including nuclear waste transmutation) to minimize nuclear wastes and safely dispose.

Annex IV

INTERFACE ISSUES IN RADIOACTIVE WASTE AND SPENT FUEL MANAGEMENT IN HUNGARY

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Hungary

IV–1. INTRODUCTION

There are usually three main areas distinguished amongst the key issues in nuclear power generation:

- Assurance of nuclear safety;
- Handling of wastes and of spent fuel (SF) related to nuclear power generation;
- Assurance of public acceptance of nuclear power, including the above issues.

In present paper will focus on Hungarian related interfaces in SF management related to the second bullet point.

IV–2. WASTE MANAGEMENT CONCEPTS

Worldwide there are generally two concepts for the handling of hazardous wastes. Of these the dilution and dispersion is nowadays used in relation to radioactive wastes (RW) only in the case of very low activity levels and quantities. The procedure followed worldwide in relation to radioactive wastes is the segregation (isolation) from the environment. In practical terms this concept in relation to different radioactive wastes is performed by overland (on surface, near surface and/or deep geological) disposal of the wastes.

IV–3. THE ELEMENTS OF THE SOLUTIONS FOR MANAGEMENT OF SF AND HIGH LEVEL AND LONG-LIVING WASTES (HL AND LLW)

The solutions for management of SF and HL and LLW are composed from the elements given below (the list does not provide detail the exact definition of the different elements):

- Interim storage (the duration is several decades; the purpose is waiting for more favourable conditions of the material to be handled). The interim storage – as an independent element – might be grouped in different manners depending on what material, what location and what method is involved in the storage (e.g. AR storage; AFR Dry Storage; AFR Wet Storage and others for HL and LLW);
- Reprocessing, advanced reprocessing amended with separation and transmutation (the purpose of this is to increase fuel utilization efficiency, decrease the radioactivity and radio-toxicity of the wastes to be disposed; which were generated as a result of the process);
- Disposal (in the long term is the only presently known solution for isolation, its duration is definite). It includes, or might include the disposal of both SF and HL and LLW;

- Since the above mentioned elements are linked to each other by appropriately arranged and secured transportations, it is inevitable to deem the transportation operation(s) (one or more) as an independent element.

IV–4. BACK-END STRATEGIES OF THE FUEL CYCLE (BE OF FC)

Basically it is possible to create four fuel cycle back end strategies from the above elements. The list given below does not include precise definition of the four alternatives, neither does it evaluate the different alternatives:

- Opened fuel cycle (direct disposal or once through) within which the SF stored temporarily – mainly in two phases (AR storage and AFR storage) is disposed in appropriate capsules in deep geological repositories;
- Conventional closed fuel cycle (reprocessing) within this alternative following the interim storage the SF is reprocessed. Those main components arising from this, which might be further utilized (U and Pu, or a part thereof) is recycled to fuel manufacturing (MOX fuel fabrication), while the surplus materials are in appropriately conditioned form – e.g. vitrified – and capsulated disposed in deep geological repositories;
- Advanced closed fuel cycle (Reprocessing + P and T) is similar to the conventional closed cycle, but within this strategy following reprocessing the minor actinides are also removed. The separated isotopes are transmuted in combination with power generation and only the purely reprocessing wastes, and those conditioned wastes generated during transmutation will be, following appropriate capsulation, disposed in deep geological repositories;
- Postponed solution (Wait and See). Practically one of the above mentioned three solutions (or potentially introduction of a not known today and not described above method) after interim storage with longer than usual – i.e. several decades – period of interim storage.

It shall be noted, that the range of the fuel cycle back-end strategies presented here might be further extended if taking into account the locations of the individual elements (e.g. taking into consideration regional SF disposal).

IV–5. DEFINITION OF INTERFACES OF SPENT FUEL MANAGEMENT (SFM) OPTIONS

In our judgment the definition of interfaces in spent fuel management (SFM) options is possible on at least two levels, in accordance with:

- Technical character interfaces between the elements (forms of interim storage, reprocessing methods, disposal, and transportation) considered in the development of the different strategies. These interfaces arise in the form of easily conceivable technical and/or licensing problems, when the individual elements cannot be realized or adjusted to each other as originally planned. Three examples from Hungary are provided on such cooperation, interface problems and resolution thereof;
- Introduction of the strategic level interface: In this case we bear in mind review of occurrence of such change(s) and consequences of them on level of certain countries, which leads, or might lead to reconsideration of the total waste

management and disposal programme. In this relation later on the Hungarian practice will be presented.

IV–6. EXPERIENCES GAINED WITH TECHNICAL CHARACTER INTERFACE IN HUNGARY

IV–6.1. Take-over of slightly burned out fuel assemblies from Germany

In 1996, following two years of preparatory works Paks Nuclear Power Plant took over from the former East German nuclear power plant (Greifswald; Bruno Leuschner NPP) 235 WWER type, Russian manufactured, slightly burnt fuel assemblies. The fuel assemblies were originally used in Unit 5 of the Greifswald NPP, they reached approx. burn-up level of one effective day at 55% power level of the reactor during the course of start-up, when, due to an incident the reactor was shut down and the commissioning was suspended and never completed.

This interface involved the transfer between two WWER-type power plants, however not by usual means of using the WWER power plants designed C-30 or TK-6 containers, but by using a Castor-84 transport container. The task to be solved was to create a reception platform and auxiliary mechanical elements for the Castor container in Paks Nuclear Power Plant in order to be able to transfer the fuel assemblies intended to be taken over by Paks NPP. Obviously the solution of the technical issues was accompanied with execution of the relevant licensing procedures, as well as obtaining the support of the local public and defeating the resistance of certain NGO organizations (Energy Club and the Greenpeace).

The activity described above was successfully completed in February 2006 with the receipt of 235 fuel assemblies. Altogether three Castor-84 containers were required for loading into them the amount of fuel assemblies in question. The transportation and handling of the three containers there and back did not cause any halts either from technical, legal or social political aspects.

IV–6.2. Storage of leaking fuel assemblies in ISFSF

The spent fuel assemblies generated in Paks Nuclear Power Plant are stored in an MVDS-system (license bearer ALSTEC Ltd.) interim storage (ISFSF). The selection of the storage facility took place in 1992-1993, the licensing in 1994-1996, carried out in parallel with construction of the facility.

The original idea was that the facility would be suitable for interim storage for all fuel assemblies which did not need to be loaded into hermetic capsules in the cooling pools of Paks NPP; due to their condition. At the time of selection of the facility there was not a single fuel assembly kept on record at Paks NPP, which should have been declared as leaking, though there were such operational experiences, which indicated, that very likely there were a few fuel assemblies in the reactors of Paks NPP with minimum leakage (pinhole leakage).

Contrary to the original intention, it has been incorporated into the licensing documentation of the facility as well as into its final safety assessment report that the facility is suitable for the storage of fuel assemblies which might not be deemed hermetic or leak tight. The reasoning behind this position was a definitive definition for leakers, and even more difficult it would have been to define such source terms for those leaking assemblies, could not be derived for the safety report.

Recently the Hungarian nuclear authority has continuously urged a resolution to the issue of leaking (and within that the issue of assemblies with pinhole leaks, and those with greater leaks) fuel assemblies. The efforts of the authority are justified on the basis that operational personnel of the power plant have several times experienced slight activity increases in the primary reactor coolant circuit, and further, there is now a fuel assembly with greater leak, which could be precisely determined. In order to meet the authority anticipation, as a first step the power plant has obtained and introduced a telescopic system leakage inspection equipment (HŐFER-BECHTEL) operating on the basis of state-of-the-art principles. (This new tool represents the third generation of leakage inspection equipment at the Paks NPP. The first leakage inspection equipment was delivered by the Russian party as an accessory part of the power plant, this was followed by the second generation developed and installed by KWU-Siemens).

In principle, there are three options for complying with the authorities expectations:

- The first solution might be creation of possibility for receipt of fuel assemblies enveloped in hermetic sleeve in the MVDS-like interim storage. There are no obstacles for such a transfer on behalf of the power plant, since the transport container (C-30) used in between the power plant is suitable for transportation of fuel assemblies enveloped in hermetic sleeve. However, the receipt of fuel assemblies enveloped in hermetic sleeve raises a number of technical difficulties on the MVDS side;

PURAM has investigated the feasibility of receipt of fuel assemblies enveloped in hermetic sleeve and found, that this issue might be solved, but would cost a lot of money, and a number of principal equipment of MVDS should be modified along the transportation route. The solution for this possibility is not excluded, the question shall be solved on the basis of economic assumptions (how many fuel assemblies are affected all together, and thus which solution might be deemed to be an optimal?).

- The second alternative might be the modification of the final safety assessment report of the MVDS. A precondition for this is the definition of the appropriate source terms and execution of corresponding recalculation of the safety assessment report;

The general designer of the facility and its English partner are dealing with resolution of this issue, however they have not yet prepared a new safety assessment report.

- The third alternative could be if the fuel assemblies, which were found to be leaking would be shipped back to Russia by the power plant. This is also a possible solution both from legal and economic aspects.

There is no decision taken yet in the issue of the above outlined interface problem, the only intention was to provide a feeling on the complexity of this question. It shall be noted, that the handling of the damaged fuel assemblies resulting from the well-known severe accident, that took place at Paks NPP in 2003 and caused damage to a number of fuel assemblies, is out-side the scope of this interface; they will be handled totally independently from this.

IV–6.3. Extension of the operational life-time of the ISFSF

The planned operational life-time of the MVDS type ISFSF operated near to the Paks NPP is 50 years. The operational license was issued in 1997. The notary of Paks city, as special authority for the first instance construction licensing, also participated in the course of the licensing procedure. The license issued by him stated a condition, that the MVDS shall be emptied by 2047. Such a wording actually provides possibility for a 50 year operational life for the fuel assemblies loaded in 1997, however, in relation to the fuel assemblies loaded in the later years, this section of the text, stated as a condition provides a shorter operational life-time; that cannot be justified from technical point of view.

The specialists of the power plant have recognized this problem at the time of issue of the license, however, since they did not wish to jeopardize the commissioning date in 1997, being extremely important at that time, they did not initiate the re-formulation of the above condition, and this did not take place even after PURAM took over the facility as licensee and operator in 1998.

Independently from the above an extension of the operational life of the power plant from 30 years to 50 years was in progress. This circumstance and the above inconsistency jointly and/or separately necessitate an extension to the operational life of the ISFSF. The related preparatory activities (investigation of the technical feasibility; analyses of the ageing management; analyses of the list of the main equipment to be replaced; the financial planning of the process; investigation of obtaining of the public promotion related to this issue, etc.) are in progress at PURAM, however, the compilation of the licensing documentation has not start yet. The investigation of this issue already partly leads us to the scope of questions of the strategic level interfaces.

IV–7. HANDLING OF THE STRATEGIC LEVEL INTERFACES IN HUNGARY

As it was mentioned earlier it is also possible to introduce a higher level of interfaces, the management of which cannot be avoided either. The so called strategic level interfaces might also start from technical licensing or social political basis, however, in these cases the interface issues will arise not between the different elements of the fuel cycle, but even the necessity of reconsideration of certain parts of the fuel cycle or even of the whole backend might arise.

The issue of such higher level interface has merged in Hungary as well. We are aware for years of such changes which have impact on the overall fuel cycle, especially on solution of financing related to waste management.

According to the Hungarian practice the waste management tasks are financed from the Central Nuclear Financial Fund (CNFF), to which payments are executed by the power plant, while the state grant the stable value of the free finances in the fund by paying interest determined by the law.

Due to the above written, any change of major effect, which influences the waste management programme from economical and/or technical and/or social-political aspect might lead to changes of the whole programme, since the programmes optimization might have such consequences as well. The following are factors which can or have given rise to such significant changes:

- Extension of operational life-time of Paks NPP;
- Review of the decommissioning related strategy of Paks NPP;
- Updating of the potential back-end options;
- Taking into consideration the HLLLW and SF related country-specific cost factors;
- Reconsideration of the public promotion system;
- Taking into account the net present value (NPV) in the calculations, instead of updating of the considered discount rate.

These changes take (took) place at different times, thus their impact would or would have change the level of the payments made by Paks NPP to the CNFF in several steps and in abrupt manner, as well the national spent fuel and waste management programme determining the future of Paks NPP. Thus, a decision was made that the financial and strategic impacts of the changes will be taken into account by means of changing the national waste management programme in one step (in one stage), and the activities related to justification of the new waste management programme were started at PURAM in 2007. Since the new plan related to decommissioning of the nuclear power plant was completed only in December 2008, the publication for justification of the new version of the national programme was completed by PURAM on 30 June 2008.

The evaluation of the programme is in progress and, the mid- and long term plans were prepared on the basis of the actual programme in force (which is revised annually, recently the ninth such plan came into force, see www.rhk.hu) will be prepared, beginning from next year, on the basis of the new programme.

Actually the waste management programme is a kind of logics, composed from technical steps, which incorporates the total operational life-time of the nuclear power plant, the interim storage and final disposal of the different activity level wastes and spent fuel generated during operation of the power plant, the decommissioning and demolishing of the power plant, as well as erection, operation and final closure of the waste storage facilities, in other words all the waste management activities related to nuclear power and nuclear technologies of the whole country.

A schematic presentation of the first waste management programme used in Hungary is given in Fig. IV–1. Version 2 (Fig. IV–2) is a draft of the new programme which was considered by PURAM to be the most favourable alternative for the decision makers.

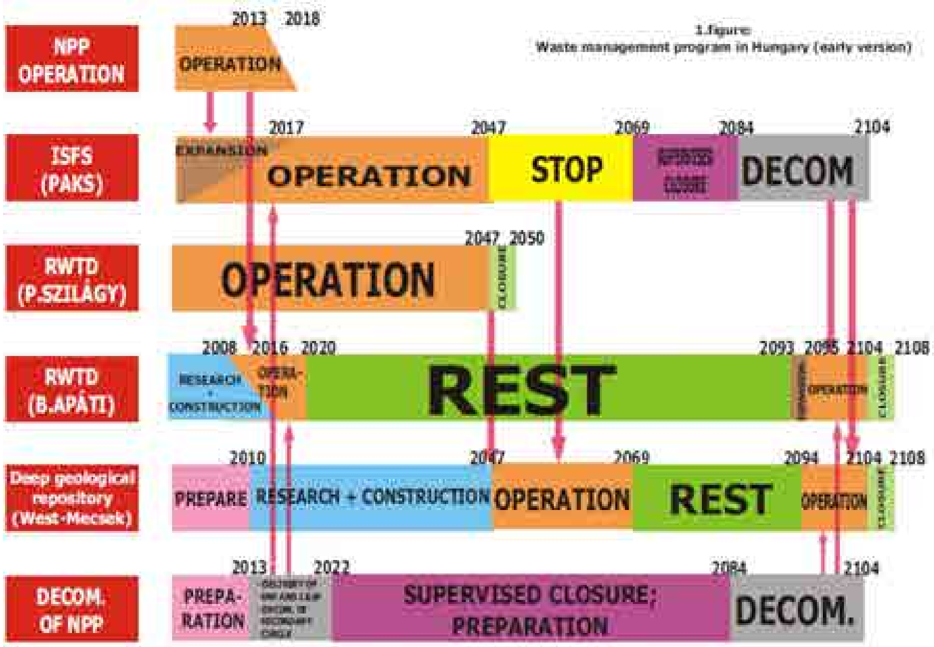

FIG. IV–1. Waste management programme in Hungary (first edition).

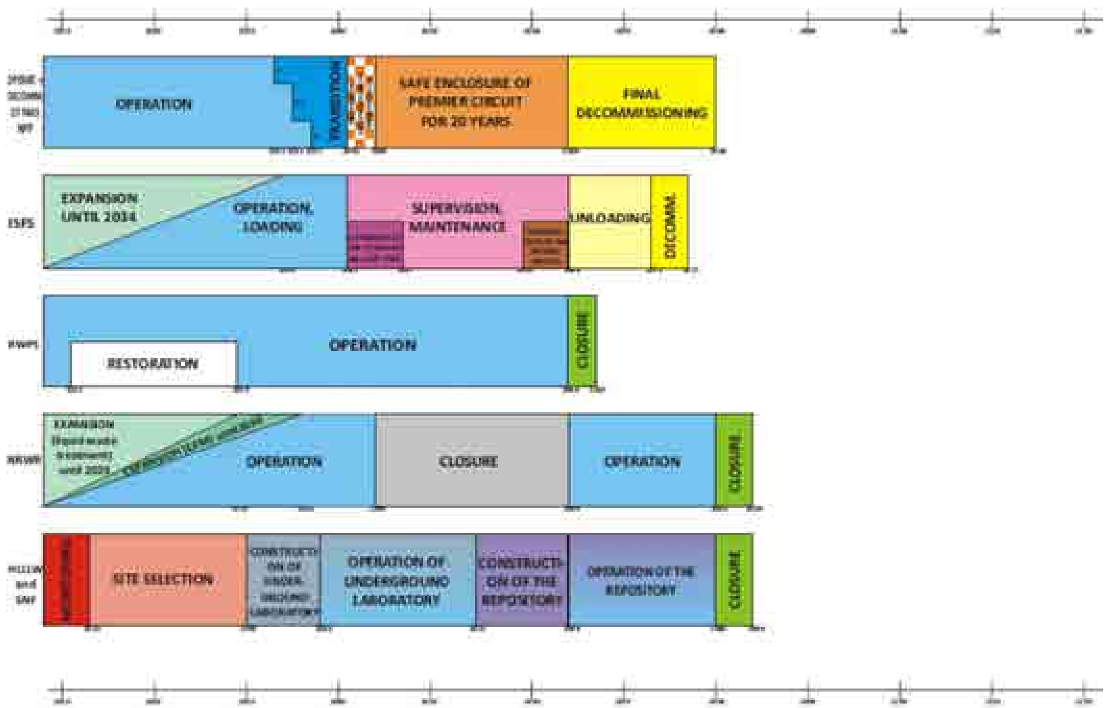

FIG. IV–2. Draft Hungarian waste management programme (third edition).

Simply due to volume limitations there is no possibility to present in this paper the exact definition and detailed impact analysis of the major changes (see in the previous listing) taken into consideration in the field of the strategic interface, however, it should be sensible that every piece of the presented changes have fundamental impact on the overall system, as well as on the details of the system related financing.

Within this paper we have provided a perception on, those items of the listed major changes, which can be generally taken into consideration (see the first three bullets in the above list). How they influence the overall programme:

- The extension of the operational life-time of the nuclear power plant will obviously result in increase of quantities of the generated SF. This change shall be taken into consideration on one hand from quantity aspect; however the simple quantity changes in the given case might induce quality changes as well. For example the suitability of the interim storage facility shall be taken into consideration taking into account the extended operational life-time. In addition to that, taking into account the increased spent fuel quantities it might be necessary to reconsider the cycle back-end strategy, including the investigation of possibility of regional disposal of the SF. As a consequence of longer operation of the nuclear power plant the quantity of the SF and amount of HLLLW will increase both from point of view of operational and also from point of view of decommissioning wastes. In this aspect the increase of the capacity of the relevant repositories shall be assured, moreover the operational period of these facilities shall be adjusted to the new situation;
- The changes of the decommissioning strategy are forced out by the changing world; however such a need might emerge also from taking into consideration the changes of the economic circumstances as well. In the present world a decommissioning strategy linked to seven decades supervised guarding cannot be accepted as a basis. A radical decrease of the duration for supervised guarding is today a world tendency, and taking this into account requires reconsideration of every element of the program, including even investigation of such issues, as what possibilities for early realization of certain program elements, for example exactly that of the deep geological repository can be realistically taken into account, in other words what level time saving is feasible in the construction of the deep geological repository;
- The review of the potential back-end strategies is and again and again emerging issue. The necessity of this is usually justified by the technical and social-political changes having place worldwide. Within the framework of this it shall be investigated, whether one could realistically take into consideration – instead or along with the earlier considered option – the introduction of a new back-end strategy.

It should be noted that in Hungary, based on the principal approval of the Hungarian Parliament from 30 March 2009, the expansion of the nuclear power generation (i.e. construction of new reactor-units) can be deemed realistic. The quantitative and qualitative changes related to construction of the new units will also have deterministic influence on the national programme being recently in preparatory phase; however, it will be possible to take into consideration only later, when the expansion related detailed technical information will be known.

IV–8. SUMMARY

A review of the principal fundamentals of the nuclear waste management was given, the complex entirety of the nuclear fuel cycle and of the elements providing basis for it, as well as the interfaces which can be defined in between them.

A strategic level was introduced into scope of interface concept, where the impact thereof appears on a higher level (by changing in overall or of part of the fuel cycle).

Certain ways of Hungarian approaches to the interfaces in between the elements and to the strategic interfaces were presented by adding, that the listed examples obviously do not cover the whole scope, and from the point of view of strategically context they do not include the changes related to future commissioning of the new reactors in Hungary.

Based on the above stated the management of the interrelations, which could be revealed in the fuel cycle, both on particular and strategic levels, is of deterministic character and thus the international cooperation is justified and this could lead to synergic results.

Annex V

INTERFACE ISSUES BETWEEN STORAGE SAFETY AND POST-STORAGE TRANSPORT SAFETY IN JAPAN

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V−1. INTRODUCTION

In Japan, interim storage facility using dual–purpose casks (metallic dry casks for storage and transport) will be in operation within a couple of years.

These dual-purpose casks are currently planned to be transported to the reprocessing facility after the long term storage (assumed to be the period of 40-60 years) without any direct inspection of their contents, which needs to open the cask lid. Therefore, interface issues between the long term storage and the post-storage transport safety are becoming especially important.

Major part of these interface issues is the integrity of such safety functions as containment, prevention of criticality, radiation shielding and decay heat removal during storage period. To ensure the integrity of safety functions for the long term storage, it is important to understand degradation phenomena by aging effects which will be expected to metallic casks and spent fuel and to consider adequately them in the safety evaluation of the interim storage facility for the long term storage and post-storage transport.

Japan Nuclear Energy Safety Organization(JNES), which is the Technical Support Organization(TSO) for Japanese government (Nuclear and Industrial Safety Agency), has extracted ‘expected degradation phenomena by aging effects’ and has carried out various tests and evaluation, including corrosion test of metallic cask materials [V−1], material test for basket and neutron shielding material [V−1], metallic cask lid leak test (including full scale mock-up cask drop test), simulated thermal test for metallic cask [V−1], spent fuel cladding integrity test. In addition, PWR spent fuel experimental storage test is currently planned. However, in this paper, summary of following 3 items are reported:

- Metallic cask lid leak test (including full scale mock-up cask drop test);
- Spent fuel cladding integrity test;
- PWR spent fuel experimental storage test (long term test plan).

V−2. METALLIC CASK LID LEAK TEST (INCLUDING FULL SCALE MOCK-UP CASK DROP TEST)

As previously mentioned, metallic casks are assumed to be transported after the long term storage without any direct inspection of their contents.

For above usage, it is important to assure the integrity of spent fuel and cask itself during/after the long term storage. To assure the integrity, cask cavity should be kept inert atmosphere and the gasket used for the cask lid shall maintain enough sealing capability during storage. The gasket (which may have been degraded during storage by relaxation effects) shall still keep enough sealing capability even under the accident condition of transport (for example, nine-meter drop). Therefore, durability evaluation for degraded gasket under the accident condition is required.

In this study, elementary tests (Gasket leak rate tests against dynamic displacement with simulated degraded gasket) and leak rate tests after full scale mock-up cask drop tests were carried out. In the elementary tests, small-scale degraded gaskets were slid using hummer impact and leak rates were measured as a function of displacement. On the other hand, in the leak rate tests after full scale mock-up cask drop tests(nine meter drop), a full scale mock-up cask with a degraded gasket was let fall, and displacement and leak rate were measured after the test . The data from these two tests were compared. And in the leak rate tests after full scale mock-up cask drop tests, leak rates after drop tests with degraded/new gasket were measured for comparison.

V–2.1. Elementary tests (gasket leak rate tests against dynamic displacement with simulated degraded gasket)

Gasket would be exposed to high temperature and high radiation. However, in case of metallic gasket, degradation by radiation would not be critical, and relaxation due to high temperature and stress is expected to mainly affect. Base on this consideration, JNES conducted gasket leak tests against dynamic displacement with simulated degraded gasket; the gasket setup is shown in Fig. V–1 and the test results are given in Fig. V–2.

Following metallic gaskets were used in the tests:

- Type – double type (Fig. V–1);
- Materials – coil spring (inner shell high Ni alloy and outer shell Al);
- Size – hoop diameter about 1/10 of the actual gasket, section diameter same as the actual gasket;
- Simulated degradation – Larson-Miller Parameter (LMP) =7353 equivalent to 60 years storage. It has been confirmed that evaluation method with LMP is adequate as evaluation method for metallic gasket degradation by preparatory test;
- Displacement method – slide to the radial direction gaskets using hammer impaction (Impact speed: Max 700mm·sec^{-1}).

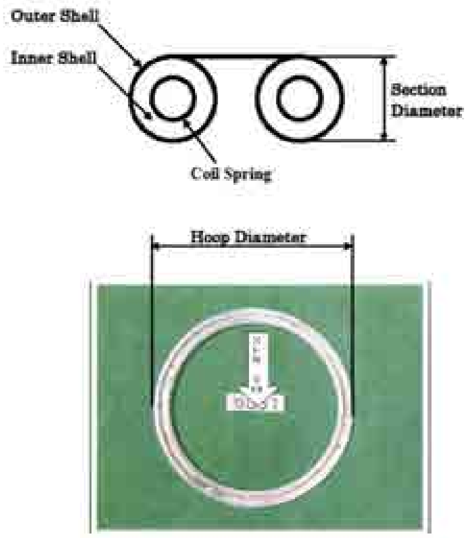

FIG. V–1. Metallic Gasket (Double Type).

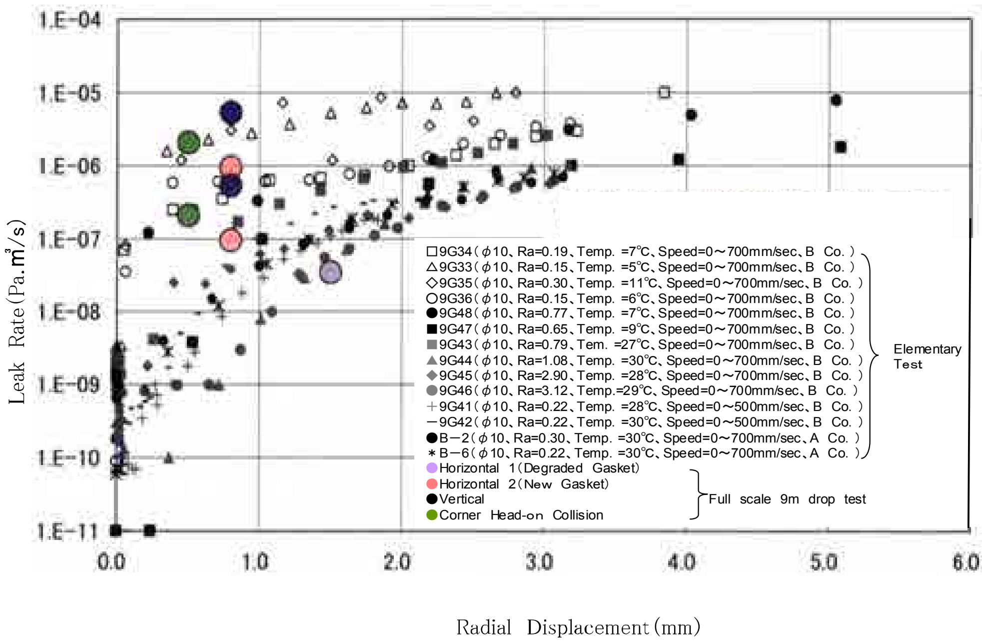

FIG. V–2. Leak rate as function radial displacement (elementary test and full scale mock-up cask drop tests).

V–2.2. **Leak rate tests after full scale mock-up cask drop tests**

Nine meter drop test (horizontal) using a full scale mock-up cask with simulated degraded metallic gasket was carried out to compare with the results from elementary test. The cask had two lids (primary and secondary lids) which had a sealing gasket for each. The space between primary and secondary lids was filled with inert gas of higher pressure than atmosphere. On the other hand, the cask cavity was filled with inert gas of the same pressure to atmosphere.

Similar drop tests (vertical, horizontal and head-on collision) of the same cask with new gasket were also conducted for comparison.

Specification and figure of the cask used in the tests are shown in Table V−1 and Fig. V−3, respectively.

Specification of metallic gasket in full scale mock-up cask drop test was the same as those used in elementary test except hoop diameter and this gasket was heated and relaxed to LMP = 7400.

Acceleration rate of main components, strains around body flange and of lid bolts, and relative radial displacements of the lid were recorded in the tests. Leak rates of primary and secondary lids and pressure of the space between primary and secondary lids were inspected before and after each test.

Results from full scale mock-up cask drop test are shown in Fig. V−2, where the leak rates from full scale mock-up cask drop test were multiplied by one-tenth considering hoop size difference between elementary tests and full scale mock-up cask drop test.

Leak rate of lids with relaxation (degraded) gasket from full scale mock-up cask drop test is found to be in good agreement with results from elementary test. A comparison of the leak rate of new gasket (colour of pink circles in Fig. V−2) with that of degraded (relaxation) gasket (colour of purple circle), there found no distinctive difference between new and relaxation gasket.

As a conclusion, it seems to be an adequate evaluation method to estimate leak rate of the lid with relaxation (degraded)/new gasket as a function of radial displacement, because the results from full scale mock-up cask drop test is in good agreement with elementary test results. The degraded gasket, after the long term storage, still has the same sealing capability as that of new gasket.

TABLE V−1. SPECIFICATION OF THE MOCK-UP CASK

Length	6.7 m
Maximum diameter	3.5 m
Weight	135 t
Cavity	Filled with inert gas (pressure 1×10^5 Pa)
Lids	Primary and secondary lid with metallic gasket Space between primary and secondary lid is filled with inert gas (pressure drop 4×10^5 Pa)
Metallic gasket	Double type Material: Coil spring and inner shell (High Ni alloy), outer shell (Al) Sectional diameter 10 mm (gasket for inner lid), 5.6 mm (gasket for primary lid)

FIG. V–3. Full scale mock-up cask.

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V–3. SPENT FUEL CLADDING INTEGRITY TEST

Japanese regulation requires that the integrity of spent fuel during the storage shall be maintained. From the viewpoint of prevention for the failure of spent fuel due to cladding tube thermal creep and for the degradation of cladding tube mechanical properties, the temperature and cladding stress of spent fuel shall be limited. JNES conducted the following three tests; thermal creep test [V–2, V–3], irradiation hardening recovery test [V–4] and hydride reorientation test from 2000 to 2008 [V–5].

V–3.1. Thermal creep

In creep tests [V–2] and [V–3], creep deformation properties of unirradiated and irradiated BWR
50 GW·d·(tU)$^{-1}$ type Zry-2 and PWR 48 GW·d·(tU)$^{-1}$ type Zry-4 claddings have been investigated under various temperatures (603–693 K) and hoop stresses (50–300 MPa for Zry-2, 30–250 MPa for Zry-4) conditions of the two different deformation mechanism regions, those are speculated as grain boundary sliding and dislocation creep, based on the zircaloy creep deformation map by Chin and others [V–6].

Creep strain was expressed by (1) in consideration of conventional primary creep and secondary creep concept.

$$\varepsilon = \varepsilon_p^s + \dot{\varepsilon}\, t \qquad (1)$$

Where, ε : Creep strain

ε_p^s : Saturated primary creep strain

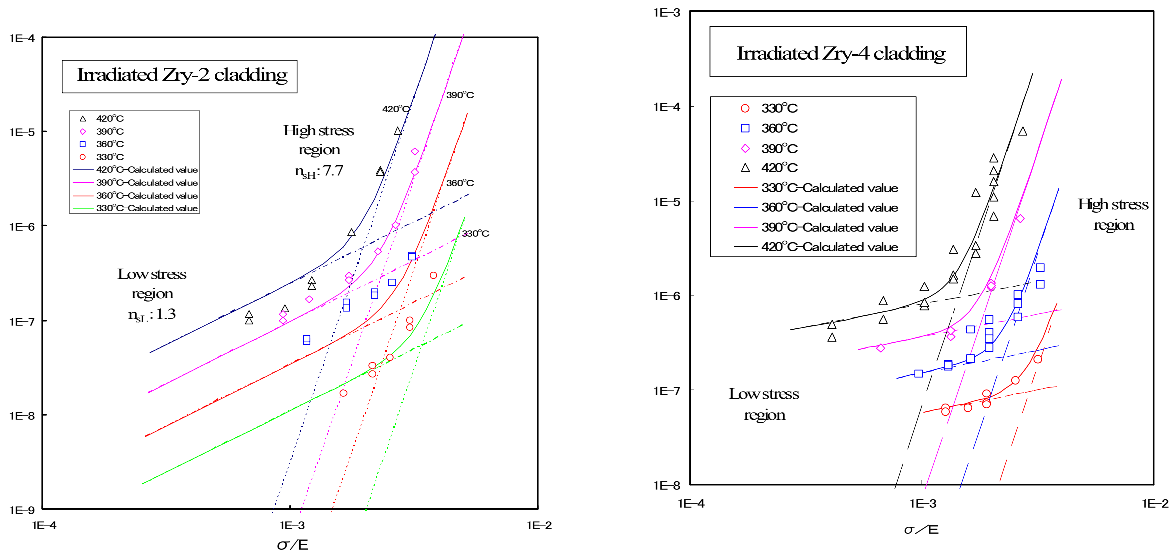
$\dot{\varepsilon}$: Secondary creep rate

t: Time

Secondary creep rate was determined separately for two deformation mechanism regions. As a result, secondary creep rate in low stress region and secondary creep rate in high stress region was different as shown in Fig. 4. From the analysis of the test results, creep equations were derived for PWR and BWR irradiated claddings. Details of the creep equation are described in [V–3].

The threshold creep strain of transition to tertiary creep from secondary creep was studied. The relation between the threshold strain and secondary creep rate of Zry-4 cladding is shown in Fig. V–5. The threshold strain of transition to tertiary creep is likely to be more than 10 % for unirradiated cladding and 1% for irradiated cladding.

The thermal creep behaviour of irradiated 55 GW·d·(tU)$^{-1}$ type fuel cladding was also investigated. As for BWR cladding, the creep test results for 55 GW·d·(tU)$^{-1}$ type Zry-2 cladding showed the good agreement with those for 50 GW·d·(tU)$^{-1}$ type Zry-2 cladding. As for PWR claddings, test results indicated that there was not large difference in thermal creep behaviour between 48 GW·d·(tU)$^{-1}$ type Zry-4 and 55 GW·d·(tU)$^{-1}$ type cladding material (MDA and ZIRLO).



(a) BWR 50 GW·d·(tU)$^{-1}$

(b) PWR 48 GW·d·(tU)$^{-1}$

FIG. V–4. Stress dependency of secondary creep rate (σ : Hoop stress, E: Young's modulus).

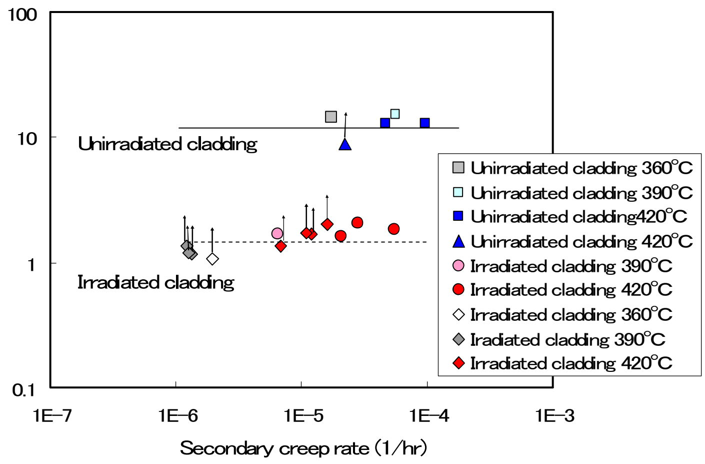


FIG. V–5. Relation between threshold strain to tertiary creep and secondary creep rate.

V–3.2. Prevention for the degradation of cladding mechanical property

V–3.2.1. Hydride reorientation and its effect on cladding mechanical properties

It is well known that the dissolved hydrogen precipitate as hydride along radial direction under the condition in which somewhat large circumferential stress is applied during the precipitation stage and the more radial hydride lead to less circumferential ductility of the cladding. Therefore the radial hydride reorientation during the spent fuel cooling stage should be considered as one of the cladding degradation mechanism in dry storage.

Zry-2 cladding from the BWR spent fuel (40, 50 and 55 $GW{\cdot}d{\cdot}(tU)^{-1}$) and Zry-4 cladding from the PWR spent fuel (39 and 48 $GW{\cdot}d{\cdot}(tU)^{-1}$) are the main test materials. In addition, MDA and ZIRLO claddings (55 $GW{\cdot}d{\cdot}(tU)^{-1}$) prepared from PWR spent fuel also were tested. Temperature (473–673 K for Zry-2, 523–613 K for Zry-4), hoop stress (16–100 MPa for Zry-2, 85–130 MPa for Zry-4) and cooling rate (0.6–30 $K{\cdot}h^{-1}$) of cladding specimens are the test parameters. The temperature conditions were set considering the cladding temperature in the dry storage. The hoop stress conditions were set to find the threshold stress for reorientation at each temperature.

Hydride reorientation test

In hydride reorientation test, hydride reorientation treatment (HRT) was carried out. The biaxial stress in cladding tube specimens was applied by internal pressure of Ar gas. The specimen temperature was held at the HRT solution temperature in the furnace to dissolve the hydrogen for 30 (for PWR cladding) or 60 (for BWR cladding) minutes, then decreased to around room temperature to precipitate the hydride. The morphology of hydrides, including the orientation, was evaluated from metallography after HRT.

In order to clarify the HRT temperature, hoop stress and cooling rate dependence of hydride reorientation quantitatively, the degree of radial hydride orientation was evaluated using Fn(40) and Fl(45) [V–7], which are defined in (2) and (3). Details of the evaluation method are described [V–5].

$$\mathrm{Fn}(40) = \frac{\text{Sum of the number of hydrides in radial direction } \pm 40^\circ}{\text{Sum of the number of all hydrides}} \tag{2}$$

$$\mathrm{Fl}(45) = \frac{\text{Sum of the length of hydrides in radial direction } \pm 45^\circ}{\text{Sum of the length of all hydrides}} \tag{3}$$

In Fig. V–9, Fn(40) and Fl(45) are plotted versus HRT hoop stress for BWR Zry-2 claddings with Zr liner. For 300°C HRT specimens, a slight increase in Fn(40) and Fl(45) was observed at HRT hoop stresses of 40 MPa and 70 MPa, and significant hydride reorientation occurred at HRT hoop stress of 100 MPa. In 400°C HRT, both Fn(40) and Fl(45) were increased with the hoop stress within the tested hoop stress range. The results of hydride reorientation tests for irradiated PWR 48 $GW\cdot d\cdot(tU)^{-1}$ type Zry-4 cladding were evaluated in the same way as for BWR cladding, and the results are shown in Fig. V–10. The Fl(45) increased for the specimens with HRT conditions of 120 MPa/300°C/30ºC$\cdot h^{-1}$ compared to as-irradiated specimens as shown in Fig. V–10.

Ring compression test

The mechanical-property change due to radial hydride reorientation was evaluated by the ring compression test at room temperature. In the ring compression test, ring specimens with 8mm in length were prepared from the cladding tube after HRT. Ring specimens were compressed in the radial direction on the flat plane with a crosshead speed of about 2 $mm\cdot min^{-1}$ at room temperature. The test for as-irradiated specimen was also carried out as the reference.

The results of ring compression tests were evaluated in terms of the crosshead displacement ratio as defined in Fig. V–11. Fig. V–12 summarizes the results for BWR Zry-2 cladding with a Zr liner. The crosshead displacement ratio did not show the particular hoop stress dependence for the HRT 300°C and 250°C specimens, although some degree of reorientation was observed for HRT 300ºC, 70–100 MPa specimens. On the other hand, the crosshead displacement ratio increased for HRT 400°C, 0 MPa (only heat treatment with no stress) specimens compared to as-irradiated specimens, and it decreased with the increase of hoop stress in HRT, or the amount of radial hydride. The results of the ring compression test for 48 $GW\cdot d\cdot(tU)^{-1}$ type Zry-4 cladding are summarized in Fig. V–13. The crosshead displacement ratio was almost the same level for the specimens with HRT conditions of 250ºC/100 MPa and 275ºC/100 MPa compared to as-irradiated specimens. The specimens with HRT conditions of 340ºC/100 MPa and 300ºC/100 MPa showed lower crosshead displacement ratio compared to as-irradiated specimens, although Fl(45) did not show the increase after these HRT conditions.

Based on these test results, limit values of temperature and stress to prevent the degradation of mechanical properties in the dry storage were determined for each type of BWR and PWR cladding.

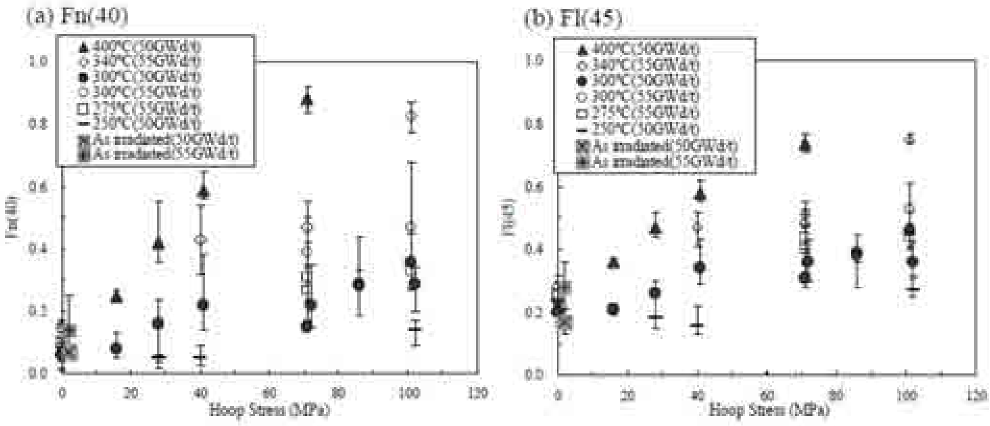


FIG. V‒6. Correlation between the degree of reorientation and the HRT conditions for irradiated BWR Zry-2 cladding with Zr liner (cooling rate: 30ºC·h^{-1}).

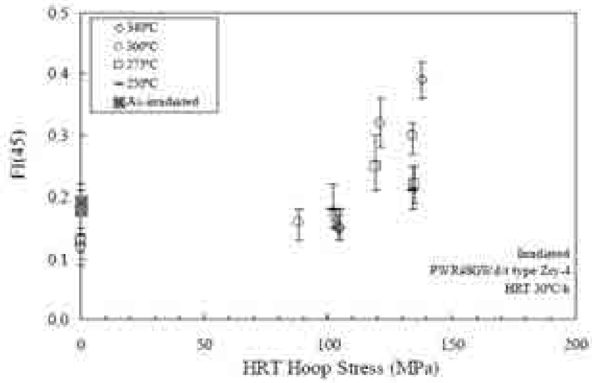


FIG. V‒7. Correlation between the degree of reorientation and the HRT conditions for irradiated PWR Zry-4 cladding.

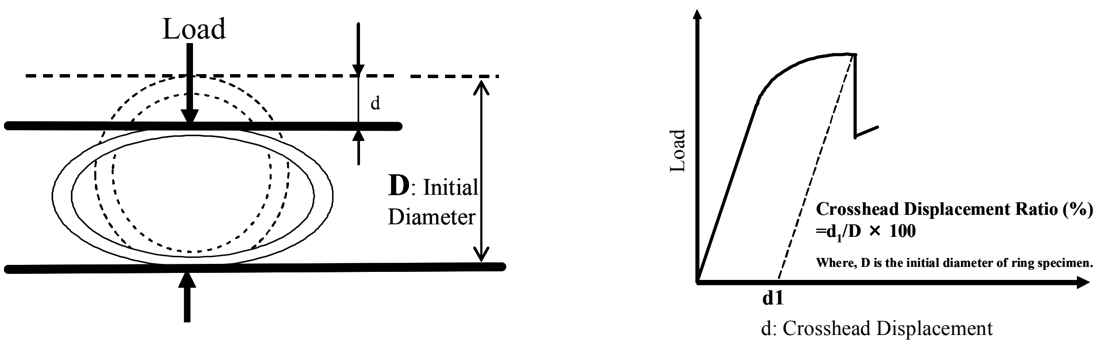


FIG. V‒8. Ring compression test method.

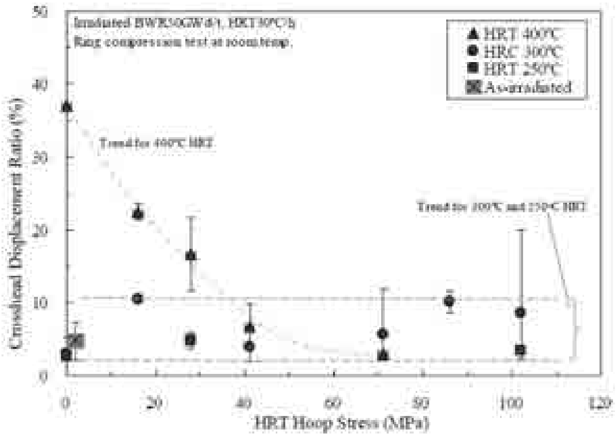


FIG. V–9. Correlation between the ductility of specimens and HRT conditions for irradiated BWR Zry-2 cladding with Zr liner.

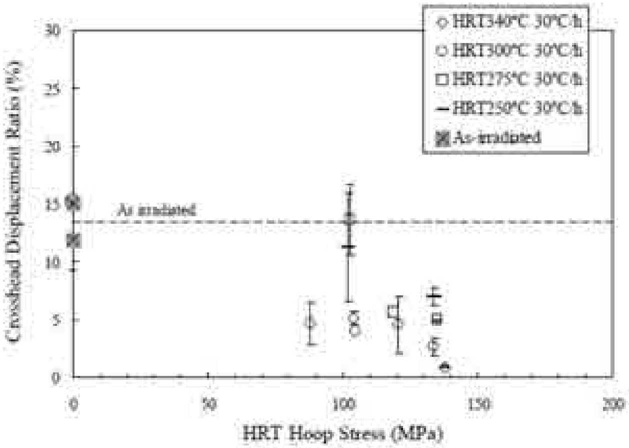


FIG. V–10. Correlation between the ductility of specimens and the HRT conditions for irradiated PWR 48GWd/t type Zry-4 cladding.

V–3.3. Recovery of irradiation hardening

In recovery test, isothermal annealing of irradiated cladding tube was performed, and then hardness of annealed material was measured [V–3]. Annealing temperature conditions were from 270ºC to 420ºC. Annealing duration is maximum about 10 000 h.

The correlation of annealing conditions (time, temperature) and micro Vickers hardness of irradiated Zry-2 and Zry-4 cladding is shown in Fig. V–11. No recovery in hardness was observed for PWR 48 GW·d·$(tU)^{-1}$ type Zry-4 cladding below 300 ºC. Although it is reported that below 305ºC the irradiation-hardening is not annealed [V–7], BWR 50 GW·d·$(tU)^{-1}$ type Zry-2 shows a slight decrease in hardness after 5000 h annealing at 300ºC. This implies that, it is necessary to consider the recovery of irradiation-hardening to evaluate the cladding strength under dry storage, if applicable.

V–4. PWR SPENT FUEL EXPERIMENTAL STORAGE TEST (UNDER PLANNING)

V–4.1. Background and outline

When the Nuclear Safety Commission (NSC) established the "Safety Review Guidelines for Spent Fuel Interim Storage Facilities using Dry Metallic Casks", NSC presented the following message in the decision statement.

'The licensee should continue to inspect the conditions of the dry storage in the nuclear power plants, and try to accumulate expertise related to long term integrity in terms of confirming integrity of the metallic casks and their contents during transport after storage'.

To meet above decision statement, JNES (as TSO for NISA) and Japanese PWR utilities have a plan of "PWR spent fuel experimental storage test."

The purpose of experimental storage test is to accumulate expertise related to long term integrity of fuel as well as basket and metallic gasket.

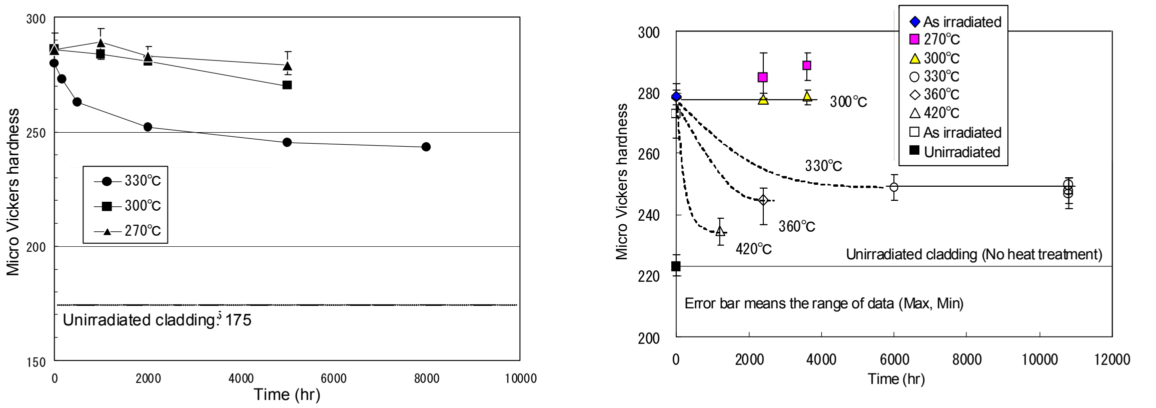
The details of the experimental storage test plan have not yet been fixed; however, the concept is as follows.

Actual PWR spent fuel will be stored in the small size model cask .Cask cavity will be filled with inert (helium) gas whose pressure will be lower than atmospheric pressure. Sealing system has primary and secondary lids with gaskets, and the space between these lids will be kept higher than atmospheric pressure. The sealing system can simulate actual storage conditions in the metallic cask. This model cask will be installed in a hot cell. After the long term storage, assumed to be 40–60 years, the components of the model cask and spent fuels will be examined.

V–4.2. Tasks of the JNES and Japanese utilities

JNES intends to accumulate expertise required for stipulating the regulations and safety standards for transport of spent fuel after long term storage through the experimental storage, under the collaboration with, but, independently from utilities.

Tasks under planning are shown in Table V–2. Major components which JNES is going to examine are basket metallic gasket (as components of the metallic dry cask) and spent fuel (as contents of the cask).



(a) BWR 50 GW·d·(tU)$^{-1}$ type

b) PWR 48 GW·d·(tU)$^{-1}$ type

FIG. V‒11. Irradiation-hardening recovery property.

TABLE V‒2. PROPOSED TASK SHARING OF PWR SPENT FUEL EXPERIMENTAL STORAGE TEST [(*1)]

Item	Fuel	Basket		Metallic Gasket
		Model Cask	Surveillance Specimen	
Design and fabrication of model cask	U	U	J/U [(*2)]	U
Tests before storage	U	U	J	J/U[(*3)]
Tests during storage	J/U [(*5)]	-	J	J/U [(*4)]
Tests after storage	J/U [(*5)]	J/U	J	J/U[(*4)]

(*1) JNES (Government), U: Utilities

(*2) Surveillance specimen will be provided by JNES to the utilities and the utilities will set it into the model cask. The utilities will design the model cask in such a manner that the surveillance specimen provided by the government can be set.

(*3) JNES will test the metallic gasket provided by the utilities, which is originated from the same lot as the metallic gasket used in the model cask.

(*4) JNES will test the metallic gasket provided by the utilities, which will be used in the model cask.

(*5) JNES will study the necessity for the post irradiation examination of the fuel from the viewpoint of advancement of regulation hereafter.

V–4.3. Tests carried out by JNES

JNES will carry out the following tests:

V–4.3.1. Basket

Material test (chemical composition, metallographic structure, and mechanical strength, etc.) of surveillance specimen to confirm that properties are within the predicted range by acceleration test, etc. in order to verify safety in post-storage transport of spent fuel

V–4.3.2. Metallic gasket

Mechanical property test (amount of deformation, physical property values, etc.) of the metallic gasket of the primary lid to confirm that properties are within the predicted range by acceleration test, etc. in order to verify safety in post-storage transport of spent fuel

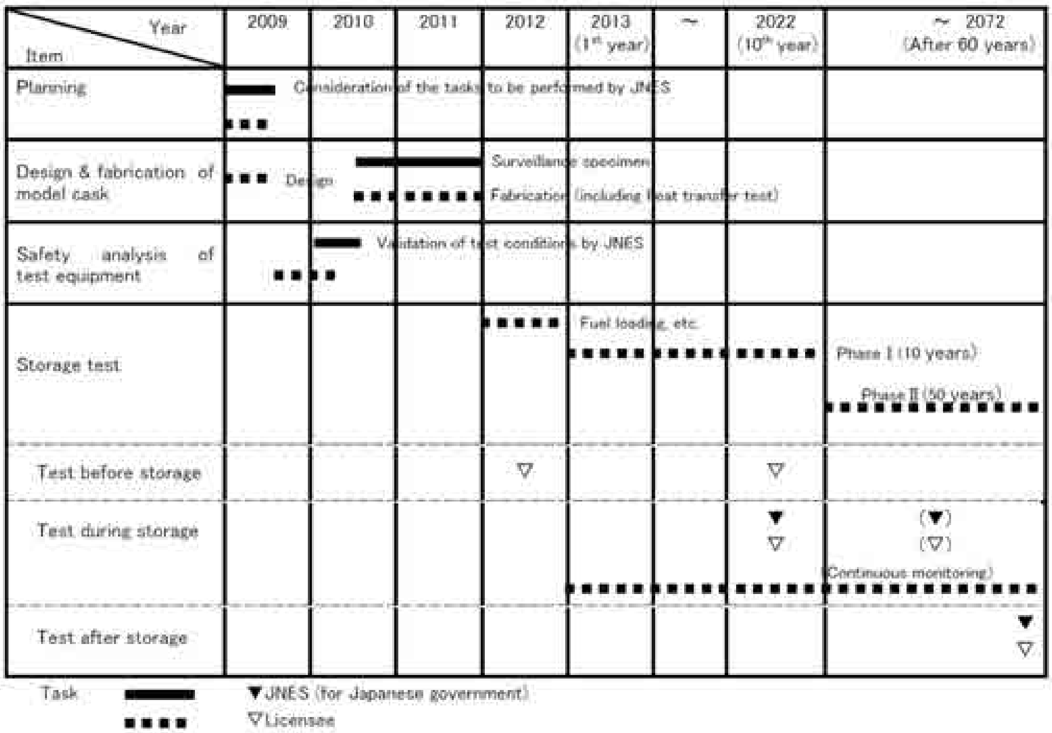
V–4.3.3. Spent fuel

If necessary, the post irradiation examination of the fuel will be carried out to examine the integrity of spent fuel in order to verify safety in post-storage transport of spent fuel (necessity of this test will be studied in future).

V–4.3.4. Schedule

Table V–3 shows the planed schedule of the experimental storage test.

TABLE V–3. PLANED SCHEDULE OF PWR SPENT FUEL EXPERIMENTAL STORAGE TEST



V–5. CONCLUSION

Metallic cask lid leak test: The degraded gasket after the long term storage still have the same sealing capability as that of new gasket, and leak rate of degraded gasket after the drop accidents can be estimated by the results from elementary test with small size gasket.

Spent fuel cladding integrity test: Degradation effects of creep, hydride reorientation and recovery of irradiation hardening were studied.

PWR spent fuel experimental storage test: Integrity of basket and metallic gasket will be tested after the long term storage under the actual storage conditions and integrity of spent fuel will be also tested if necessary.

The test data and evaluations carried by JNES will be a part of supporting data to proceed with Japanese current plan.

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Annex VI

EXPERIMENTAL STUDY ON INFLUENCE OF MECHANICAL VIBRATION DURING TRANSPORT OF TRANSPORT/STORAGE CASK FOR SPENT NUCLEAR FUEL ON CONTAINMENT PERFORMANCE OF METAL GASKET DURING STORAGE IN JAPAN

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VI–1. BACKGROUND

Transport casks of spent nuclear fuel will receive mechanical vibration during transport. It is known that the containment performance of metal gaskets is influenced by large external load or displacement [VI–1]. Quantitative influence of such vibration during transport on the containment performance of the metal gasket has not been known, but is crucial information particularly if the cask is stored as it is after the transport.

VI–2. PURPOSE

The purpose is to quantify influence of mechanical vibration during transport of transport/storage cask with metal gasket on the performance during storage.

VI–3. EXPERIMENTAL APPARATUS AND SPECIMEN

In order to obtain a relationship between the amount of lateral sliding (displacement) of the lid and the leak rate, a 1/10-scale model of a lid structure of metal cask with a metal gasket of double O-ring type was fabricated and assembled as shown in Fig. VI–1. The gasket had a diameter of 10 mm and was coated with aluminium sheet. The scale model consists of three flanges bolted together and helium gas was installed in a groove of one of the outer flanges. In this test, 2 atm (gauge pressure) of helium gas was filled in the space between the flanges. Eddy current displacement sensors (accuracy of ±0.01mm) were used to measure displacement of the flanges. Sliding load and relative displacements were applied to the middle flange using loading test equipment. To simulate the thermal ageing effect of the metal gasket due to the heat from spent fuel loaded inside the cask, the flanges with the metal gasket were heated for 20 hours at 180°C inside an oven prior to the tests. The temperature and the time conditions were assumed to simulate the heat history of the gasket after spent fuel loading before transport, with the aid of Larson Miller Parameter equation. Instantaneous leak rates were quantitatively measured at the lid by the helium leak detector. The leak rate and change of axial force of the bolts in the lid flange were measured with elapsed time.

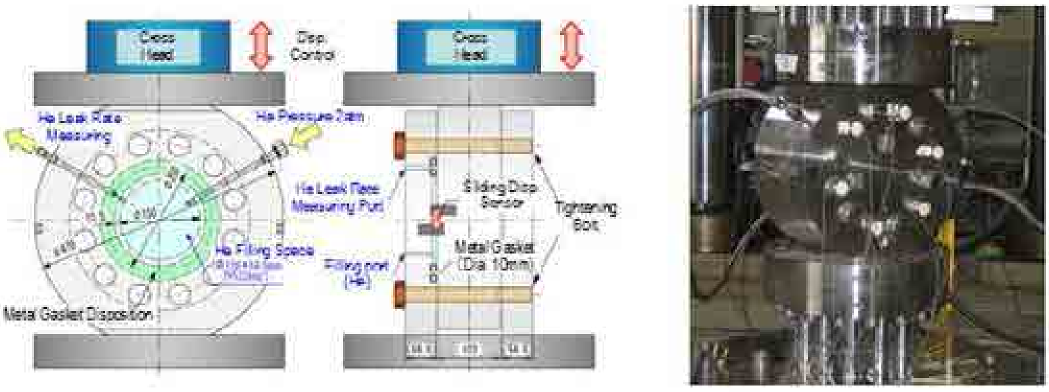


FIG. VI–1. Scale model of a lid structure of metal cask with a metal gasket.

VI–4. EXPERIMENTAL CONDITIONS

Mechanical vibration during sea transport of spent fuel shipping cask has been measured and reported by Shirai, et al [VI–2]. They obtained a time history of acceleration measured at a trunnion supports of the spent fuel shipping cask transport frame as shown in Fig. VI–2. Using the history of the acceleration, they also calculated a time history of a lateral sliding of the secondary lid of a transport/storage metal cask for spent fuel as shown in Fig. VI–3.

Based on their results, cyclic displacement of ± 0.02 mm was assumed for vibration during sea transport and given to a cask flange model as described in 3.1 above. Due to the constraint of the experimental apparatus's ability, the loading speed was 0.01 mm·s^{-1} and the frequency of the vibration was 0.125 Hz. The experiments were carried out three times for each condition.

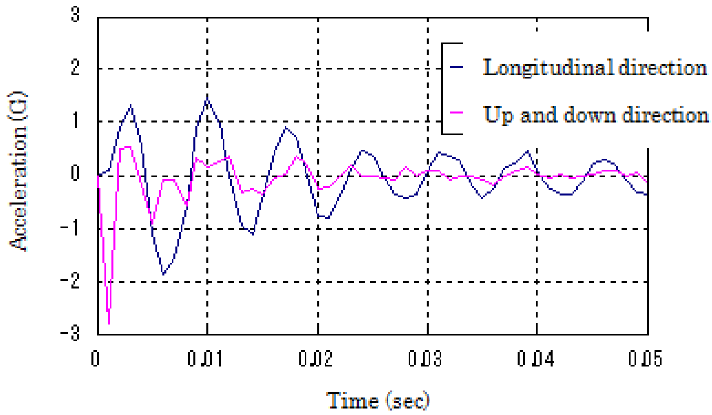


FIG. VI–2. Time history of acceleration measured at a trunnion supports of the spent fuel shipping cask transport frame.

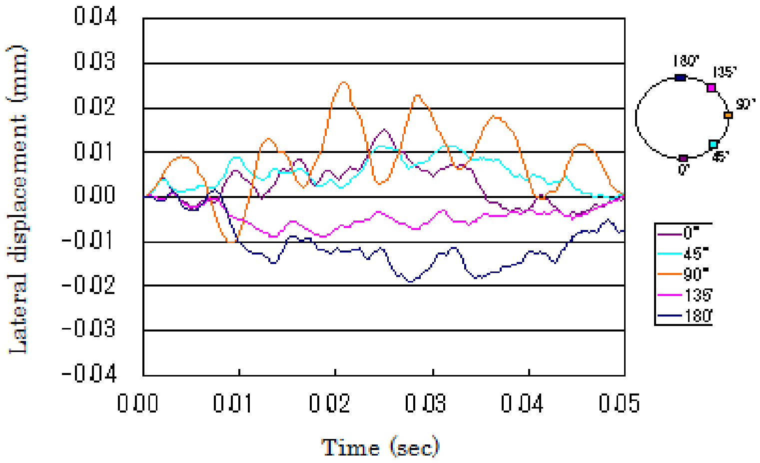


FIG. VI–3. Time history of a lateral sliding of the secondary lid of a transport/storage metal cask for spent fuel.

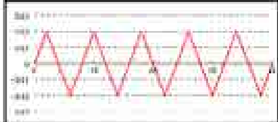
Based on their results, cyclic displacement of ± 0.02 mm was assumed for vibration during sea transport and given to a cask flange model as described in 3.1 above. Due to the constraint of the experimental apparatus's ability, the loading speed was 0.01 $mm \cdot s^{-1}$ and the frequency of the vibration was 0.125 Hz. The experiments were carried out three times for each condition.

Furthermore, in order to confirm the effect of loading speed and the effect of cyclic loading, three kinds of conditions were developed and the experiments were carried out as shown in Table VI–1.

VI–5. SET-UP OF INITIAL CONDITION

In this experiment, it is important to investigate the influence of the lateral sliding displacement of the cask lid flange with the metal gasket on the leak rate of the gasket. In order to prevent from difficulty of giving continuous and various sliding motions between the metal gasket and the lid flange in the experimental apparatus, the initial condition of the experimental apparatus and specimen were set-up as shown in Fig.VI–4. Namely, to avoid flange contact the bolt force was relaxed for the amount of the decrease of the tensile force due to the creep deformation of the gasket by the heat from the spent fuel installed to the cask before transport as shown in Fig. VI–5.

TABLE VI–1. EXPERIMENTAL CONDITIONS

Conditions	Displace ment	Speed	Heat load	Pattern of given displacement	Test number
(1) Static, one-direction-al loading	3mm	0.01mm/s	Equivalent heat before storage LMP=6930 (180℃×20hr)		No.1 :SSS70011 No.2 :SSS70012 No.3 :SSS70013
(2) Cyclic loading	±0.02 mm	0.01mm/s			No.1 :SRL3010011 No.2 :SRL3010012 No.3 :SRL3010013
(3) Dynamic, one-direction-al loading	3mm	85mm/s			No.1 :SSS80011 No.2 :SSS80012 No.3 :SSS80013

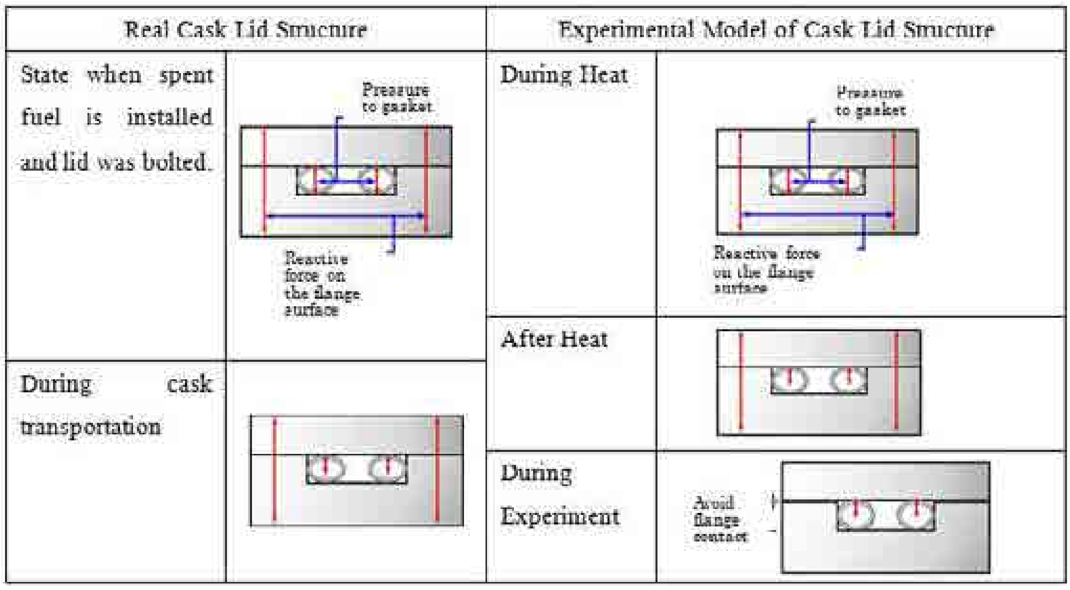


FIG. VI–4. Status of containment boundary of real cask and experimental model.

Fig. VI–5 shows change of tensile force in bolts with elapsed time. The line from (a) to (b) shows the increase of tensile force in the bolts by tightening the bolts in the lid flange with the metal gasket. The line from (c) to (d) shows decrease (relaxation) of the tensile force due to the creep deformation of the gasket by the heat from the spent fuel installed to the cask before transport. The horizontal broken line shows the tensile force in bolts recommended by gasket supplier. The point (e) shows tensile force at this experiment.

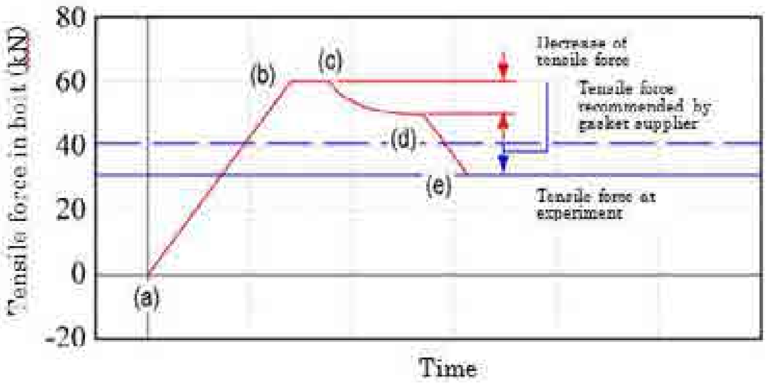


FIG. VI–5. Change of tensile force in bolts with elapsed time.

VI–6. RESULTS

VI–6.1. Static and one-directional loading results

Fig. VI–6 shows leak rate measurements as a function of radial displacement of the flange with the gasket. If the displacement is less than 0.1 mm, no leakage was found. When the displacement increased up to 3mm, the leak rate reached to about 1×10^{-6} Pa·m^3·s^{-1}. During the displacement increase, the tensile force in bolts decreased a little, but not significantly.

Fig. VI–7 shows leak rate change with time after the static and one-directional displacement experiments. The leak rate was 1×10^{-6} Pa·m^3·s^{-1} after the experiments, but recovered up to 1×10^{-8} Pa·m^3·s^{-1}.

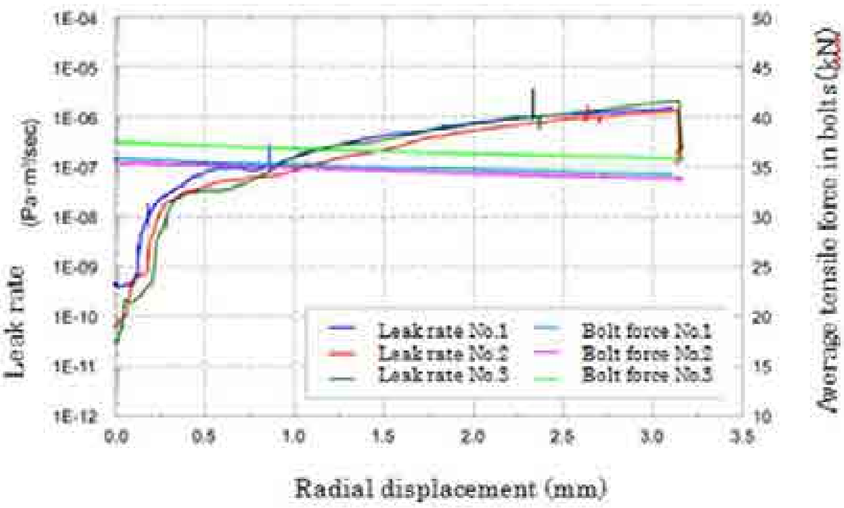


FIG. VI–6. Leak rate as a function of radial displacement.

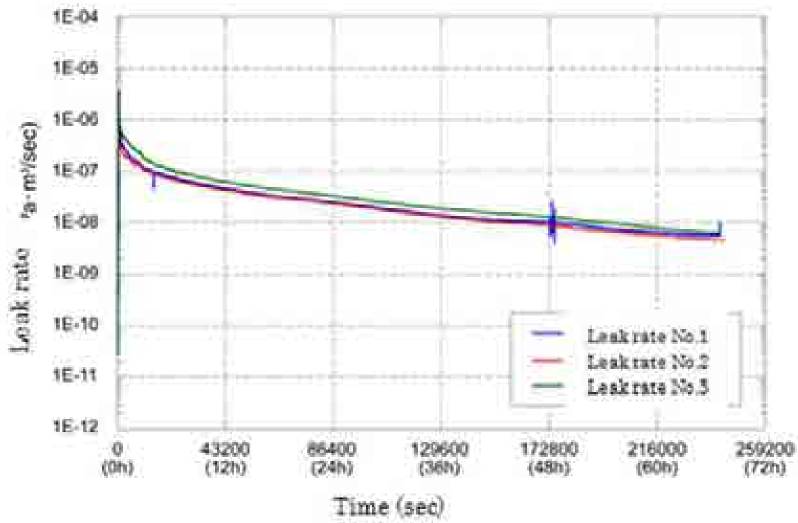


FIG. VI‒7. Leak rate with time under static and one-directional displacement loading.

VI‒6.2. Cyclic loading

The frequency of the cyclic displacement experiment was 0.125 Hz and the nominal displacement was ± 0.02 mm. Fig. VI‒8 shows a representative result showing leak rate and displacement as a function of time. The initial leak rate was 1×10^{-10} Pa·m^3·s^{-1}. The leakage started when the radial displacement exceeded ± 0.022 mm and increased as the displacement increased. When the cyclic displacement stopped, the leak rate recovered until the initial leak rate if the cyclic displacement was less than ± 0.025 mm. The leak rate did not recover to the initial leak rate if the cyclic displacement was more than ± 0.035 mm. Nevertheless, the leak rate is still less than 1×10^{-8} Pa·m^3·s^{-1}. Corresponding leak rate of a full-scale cask lid model would be less than 1×10^{-7} Pa·m^3·s^{-1} taking account of the scale factor.

On the other hand, a decrease of the axial force of the bolts was also observed during the vibration, which might have contributed to the increase of the leak rate as shown in Fig. VI‒9

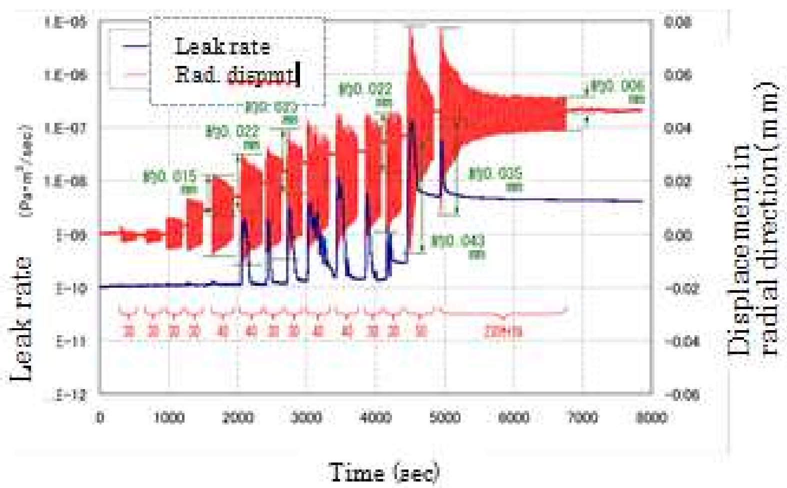


FIG. VI‒8. The third measurements of leak rate and radial displacement with elapsed time under cyclic loading to the cask lid flange..

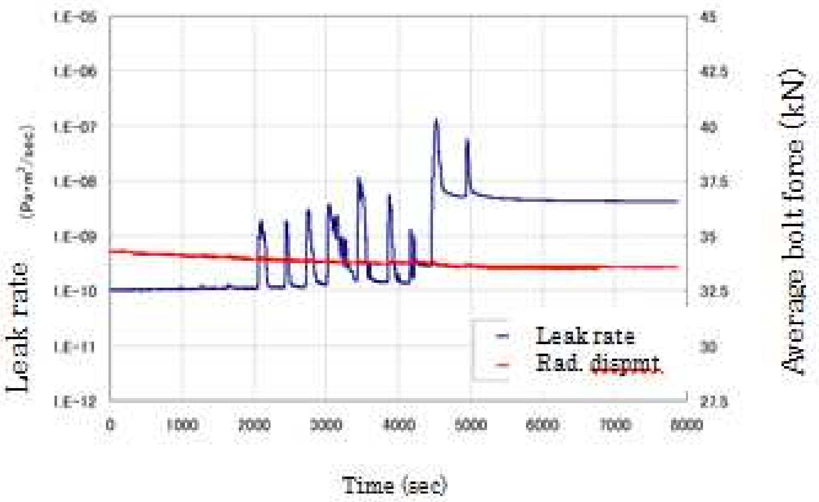


FIG. VI–9. The second measurements of leak rate and average axial bolts force with elapsed time under cyclic loading to the cask lid flange.

VI–6.3. *Dynamic and one-directional loading*

Based on a result of a free drop test of a full-scale cask and the experimental apparatus ability in this study, the loading speed was set as 85 mm·s^{-1}.

Figs VI–10, VI–11 show leak rate, radial displacement, and average bolt force as a function of time. The radial displacement increased instantaneously up to 3 mm as the leak rate increased up to 1×10^{-6} Pa·m^3·s^{-1}. Fig. VI–11 shows a decrease of the average bolt force at the time of the leakage increase.

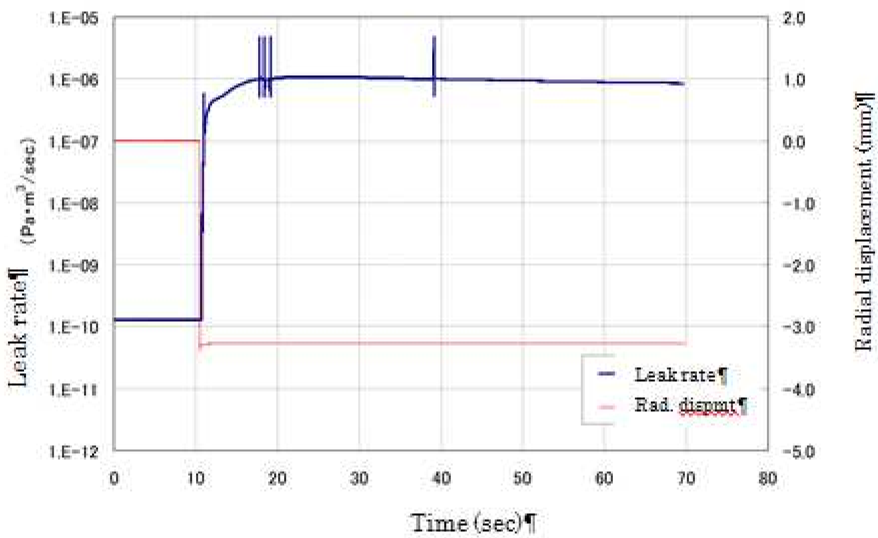


FIG. VI–10. Leak rate and radial displacement as a function of time

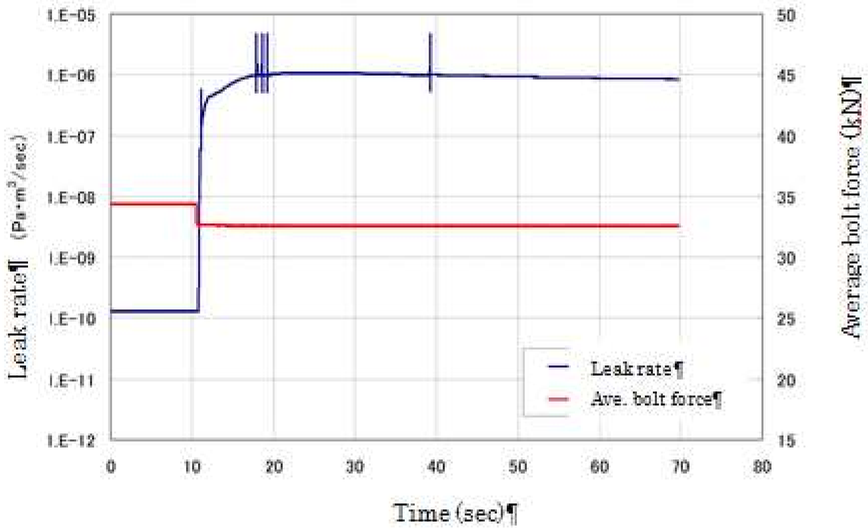


FIG. VI–11. Leak rate and average bolt force as a function of time.

The leak rate measurements were continued and recovered up to 1×10^{-8} Pa·m^3·s^{-1} after 72 hours.

VI–7. SUMMARY AND DISCUSSION

The static one-directional static loading experiments showed that no leakage was observed if the displacement was as small as 0.1 mm. If the displacement increased up to 3 mm, the leak rate increased up to 1×10^{-6} Pa·m^3·s^{-1}, but recovered up to 1×10^{-8} Pa·m^3·s^{-1} after 72 hours.

The cyclic loading experiments showed the leak rate did not increase permanently if the displacement was within ± 0.02 mm.

The dynamic loading experiments showed that the leak rate after the maximum displacement coincided with the leak rate at the same displacement by the static and one-directional loading experiments. The leak rate did not depend on the loading rate or displacement rate.

These results indicate that the mechanical vibration during transport would influence the containment performance of the metal gasket for storage if the amount of the vibration exceeded a threshold value. A further research will be required on the mechanism of the leakage.

ACKNOWLEDGEMENT

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Annex VII

INTERFACE BETWEEN THE STORAGE OF SPENT FUELS AT JAPAN'S INTERIM STORAGE FACILITY AND THE TRANSPORT THERE OF AFTER STORAGE IN JAPAN

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The Recyclable-Fuel Storage Company (RFS) plans to construct Japan's first interim storage facility for spent fuels of light water reactors. This facility is designed to have no equipment to refill spent fuels. It stores transported casks as they are. After storage, casks are carried out from the facility without opening their lids and transported to a reprocessing facility and others. Spent fuels are planned to be stored for 50 years. In other words, as spent fuels are transported after storage in the same state as they are carried into the interim storage facility, it is necessary to consider the transportation 50 years later in designing the cask. Here, we examined the points to be taken into account for storage in consideration of transportation 50 years later.

VII–1. OUTLINE OF RFS AND STORAGE FACILITY

RFS was jointly invested and established by electric power companies to serve as a company specializing solely in spent fuel storage. RFS plans to eventually store about 5,000 ton-U of spent fuels in Aomori Prefecture, Japan (Fig. VII–1). First, RFS plans to construct a building with storage capacity of up to about 3,000 ton-U that is large enough to store up to 288 dry storage metal casks (Fig. VII–2). RFS takes charge of storage and electric power companies take charge of transportation between facilities.

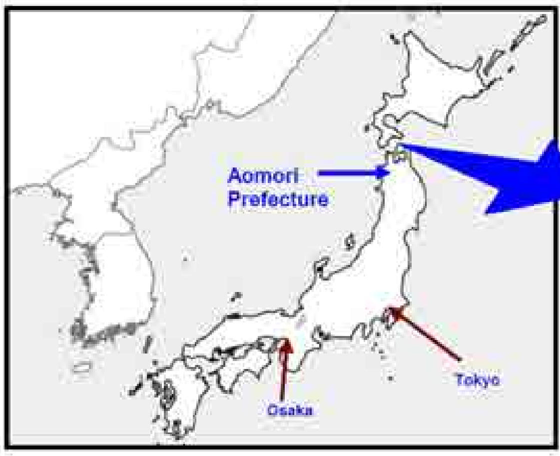

FIG. VII–1. Location of Aomori Prefecture in Japan.

FIG. VII–2. Schematic diagram of storage facility.

VII–2. CONSIDERATION OF DESIGN OF METAL CASK

To store spent fuels and transport them after storage in the same casks on a continuous basis without changing their containers, the safety functions of casks must be maintained over a long period of time. As for the component materials of metal cask, which are important to maintain the safety functions, those materials that are fully reliable against aged deterioration in the environment during the design storage period are selected. Here, we discuss the following two points, i.e., "consideration of lids" and "consideration of fuel failure," concerning the containment function as one of the important functions of cask.

VII–3. CONSIDERATION OF LIDS

The metal cask has a multiple containment structure with a primary lid and a secondary lid (Figs. VII–3, VII–4). Radioactive substances are contained in the metal cask by making the pressure in the space between the two lids positive in advance to create a pressure barrier and making the pressure inside the cask negative. To seal the lids, metal gaskets are used from the viewpoint of maintaining the containment function over a long period of time. The pressure between the lids is monitored on a continuous basis during storage. If the primary lid or the secondary lid develops a leak, it can be detected as the pressure between the lids decreases. Moreover, the cask has a structure that can mount a tertiary lid, which uses a elastomer gasket, at the time of transportation or at the time of an anomaly of containment function.

VII–3.1. Metal gaskets used for primary and secondary lids?

The metal gasket to be used in this storage facility is the same type of gasket as the one used for the dry storage casks stored in storage facilities of power stations in Japan.

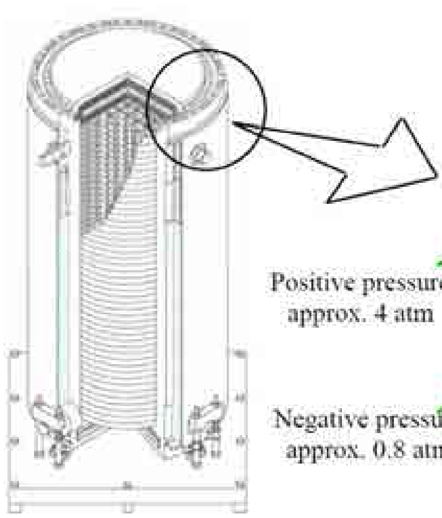

FIG. VII–3. Schematic diagram of metal cask.

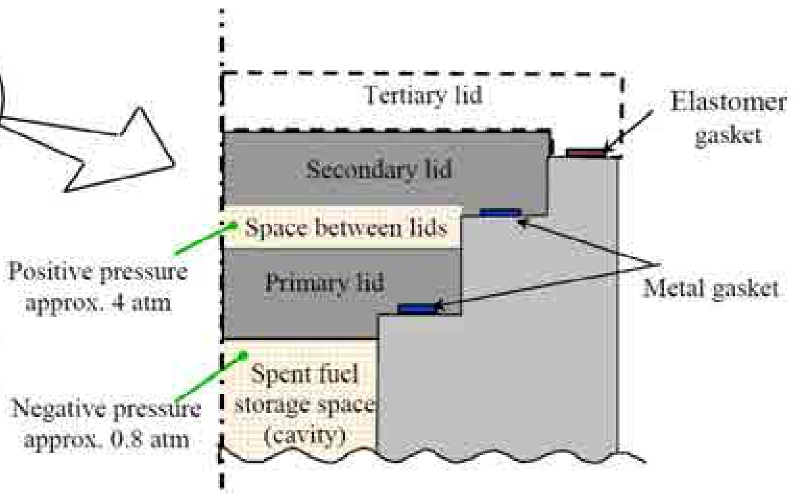

FIG. VII–4. Detail of lids of metal cask.

A possible cause of deterioration of containment function after storage over a long period of time is a decrease in reaction force due to aging of metal gaskets. In the case of comparison on nearly the same temperature and time conditions, it is reported that the larger the cross-sectional diameter is, the larger the residual reaction force becomes. For the casks to be used in the storage facility, a metal gasket with a cross-sectional diameter of about 10 mm is planned to be used. The dry storage casks in power stations use a metal gasket with a cross-sectional diameter of about 6 mm, but no problem has occurred during more than ten years of use.

To the best of our knowledge, the relations between the leakage rate and plastic deformation rate of metal gaskets and the environment temperature and time course are known. Based on this knowledge, even after a lapse of 50 years as the design storage period of our company's cask, the pressure inside the cask can be kept negative. However, these results represent the impact assessment in the case of static storage. As for an impact of transportation on metal gaskets after a long period of storage, relevant knowledge is now being accumulated.

VII–3.2. Use of tertiary lid at the time of transportation

As for the continuous use of metal gasket from storage to transportation after storage as a sealing boundary, the cask has the structure that can mount an additional tertiary lid that uses a elastomer gasket over the secondary lid for transportation because relevant knowledge is now being accumulated as mentioned earlier. The elastomer gasket has successfully been used for many years in the present cask for wet transportation. Its heat resistance, cold resistance and others can keep the hermeticity of the lid under test conditions such as falling as required by regulations. The tertiary lid is additionally mounted not only for transportation after storage but also for transportation before storage from power stations to the storage facility. Depending on the knowledge to be accumulated from now on, it is also conceivable to use a metal gasket as a sealing boundary for transportation.

The tertiary lid is also used for the purpose of mounting an additional lid in the event of an anomaly of containment function during storage. Specifically, if a leak is found in the secondary lid, the soundness of the primary lid and the maintenance of negative pressure inside the cask should be checked. Then, the metal gasket of the secondary lid should be replaced, and the containment function should be restored and the storage should be continued. In the case where no leak is found in the secondary lid and the containment function of the primary lid is considered unusual, an additional lid should be mounted in a metal cask and the cask should be delivered to electric power companies. It is required by domestic regulations in Japan to give consideration to the repairability of containment function mentioned above.

As described above, the function as a sealing boundary during transportation can be satisfied by using a tertiary lid that uses a elastomer gasket in a metal cask, in addition to primary and secondary lids that use metal gaskets, and it is also effective as a measure to be taken in the event of an anomaly of containment function.

VII–4. CONSIDERATION OF FUEL FAILURE

If fuel failure happens during fuel storage, it can be considered that fission product gas is released from failed fuels, the temperature rises as the pressure inside casks increases and the thermal conductivity decreases. It is necessary to review the impact of fuel failure for transportation after storage on a continuous basis.

VII–4.1. Fuel failure rate

As for spent fuels to be housed in metal casks in Japan, only those spent fuels whose soundness is checked based on various data in operation, sipping inspections and others should be housed, and those spent fuels that have pinholes or hair cracks should not be housed. However, considering a trace quantity of leak due to fuel damage, which may happen accidentally during storage, a fuel failure rate of 0.1% is set for the purpose of calculating pressure in a cavity space. This failure rate is conservatively set in consideration of the probability of fuel leakage during dry storage of spent fuel assemblies in the U.S.A. (about 0.01%) and the probability of fuel leakage in light water reactors in Japan (about 0.01% or lower).

VII–4.2. Pressure and temperature in cavities

To keep the negative pressure in cavities that contain spent fuels, the design should be made to achieve the leakage rate (standard leakage rate) that can keep negative pressure even if helium gas in the space between lids continues to leak only into the space where spent fuels are contained during the storage period.

As for the initial pressure in cavities, in other words, the pressure in cavities is designed to remain negative even though a leak at the standard leakage rate is assumed to happen in a state where the pressure in the early stage of storage is maintained, and fission product gas is assumed to be released from failed fuels, showing that the negative pressure in cavities is maintained by monitoring the pressure in the space between lids.

Failed fuels cause an increase in the temperature inside metal casks as they release fission product gas whose thermal conductivity is lower than that of helium. As for such failed fuels, a failure rate of 0.1% is set as is the case mentioned above. Even if it is evaluated, the rise in internal temperature is small, and the effects on the soundness of baskets and fuels in cavities can be ignored.

As mentioned above, it is evaluated in advance that no problem happens concerning the containment function, heat removal function and other functions of casks, and the design is made not to allow the changes in the internal environment during storage to have an impact on the transportation after storage.

VII–5. ALLOTMENT OF WORKS

Electric power companies house spent fuels into casks at their power stations and transport them to the storage facility. RFS conducts management and storage operations of spent fuels, deliver them after completion of storage to electric power companies at the storage facility, and electric power companies transport them to a reprocessing facility and others. Therefore, the work records at power stations where electric power companies have responsibility are transferred to RFS together with the records produced at the time of transportation after completion of transportation before storage. RFS take primary responsibility for keeping these records and other records such as measurement data during storage throughout the storage period. When metal casks are carried out, necessary records are transferred to electric power companies.

When spent fuels are delivered between RFS and electric power companies, the lids of casks are never opened. Therefore, it is important to manage the publications to check the soundness of casks and spent fuel assemblies, and each of the companies concerned should keep the publications on its own responsibility.

VII–.6. CONCLUSION

As mentioned above, as for the use of RFS's interim storage facility, spent fuels are transported in the same casks without opening their lids and delivered by allowing each of the companies concerned to check the soundness of fuels and casks through transfer of the publications at interfaces between power stations, interim storage facility, reprocessing facility and others.

Our company is now undergoing the safety review of the basic design by the government. After approval of the basic design, the approval for design and construction will be received, and then the construction of storage building and production of casks will be commenced. We plan to start receiving casks and commence the storage business in 2012.

Annex VIII

INTERFACE ISSUES ARISING IN INTERIM STORAGE FACILITIES USING STORAGE/TRANSPORT DUAL PURPOSE DRY METAL CASKS IN JAPAN

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VIII–1. BACKGROUND

The annual amount of spent fuels (SFs) discharged by the operation of commercial reactors nowadays is estimated to be around 10 000 tU level worldwide. While the amount of SFs already reprocessed account about one-third, the rest are currently stored in storage facilities, typically, in wet pools attached to nuclear power plants (NPPs). Cumulative amount of SFs stored is estimated to be about 250 000 tU by 2010 (I. Hanaki, Japan).

While wet pool system is dominant in storage facility designs, new design concepts for storage facilities have been continuously developed. One of these new designs is that using dual purpose dry metal casks. "Dual" here means that the casks are not only designed as storage containers, but also designed as transport containers that will satisfy relevant regulatory requirements for transport of radioactive materials such as TS-R-1. Advantage of adopting such "dual" design in storage facilities lies in that this could contribute to reduce the burden associated with handling operations, because, under such designs, SFs once loaded into casks can easily be "transported" to storage facilities, and after storage of several decades, they can again be "transported" to their destinations, regardless they are reprocessing facilities or final disposal sites. Other than these, adopting this kind of design can reduce the amount of radioactive wastes discharged through storage operation, thus can reduce operation costs while maintaining safety level.

In Japan, where 53 commercial NPPs are now in operation and with the annual amount of SFs produced sums up to about 1000 tU, keen needs are perceived among SFs producers (namely, utilities) to secure adequate SFs storage capacity. Therefore, a new application for constructing storage facility of 3000 tU scale in Mutsu city, located in northern part of Aomori prefecture, has been submitted in March 2007 by a subsidiary company of utilities named RFS (Recyclable Fuel Storage Company), using 288 large scale dual purpose dry metal casks. The basic concept of this facility is similar to the one already in operation in Germany (BLG at Gorleben), with one important difference that no storage of HLWs containers are planned in Mutsu. Other aspects of the facility designs are similar, including that the both facilities do not equip with hot cells that will enable the casks to be opened for inspections during storage period.

One example of dual purpose dry metal cask designs applied by RFS is shown in Fig. VIII–1.

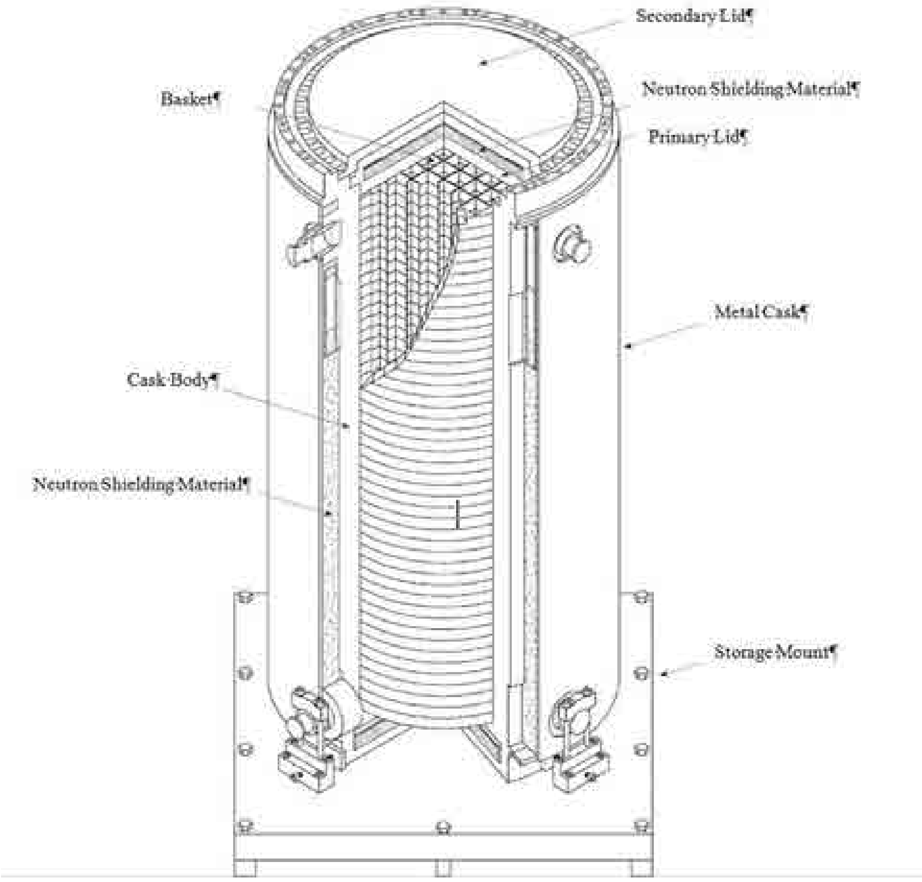

FIG. VIII‒1. BWR large scale cask: Max. capacity 69 BWR fuel assemblies.

VIII‒2. INTERFACE ISSUES CONCERNING DUAL PURPOSE DRY MATERIAL CASK STORAGE

In Japan, safety standards for storage facilities are laid out by NSC (Nuclear Safety Commission of Japan) and safety analysis of the applications of these facilities are conducted by NISA (Nuclear and Industrial Safety Agency, Ministry of Economy, Trade and Industry) based on this standard. In 2002, NSC laid out specified safety standard for storage facilities using dual purpose dry metal casks. In this NSC standard, it is assumed that storage casks have valid transport license during storage period.

In Japan, like in many other countries, licensing process of storage facilities and those of transportation casks are supposed to be conducted independently. One example of such independent licensing process is observed in duration of license. For storage facilities, it is assumed that "design lifetime" to be between 40 to 60 years, but there is no specific explicit duration of license[2]. On the other hand, for the license of transport casks, it is generally granted with 5 years duration of license, with possible renewal of unlimited times as long as casks are maintained in good conditions. As licensing process of casks for storage safety and

[2] Instead, it is required that operators maintain their facilities in good conditions, and conduct aging management program in 10 year intervals. Also, there is explicit design life time for each cask.

for transport safety should be conducted in a harmonized manner, this indicates one good example of interface issues between storage and transport[3].

Other than the above, there are several interface issues arising for SFs storage facilities using dual purpose dry metal casks, which are not equipped with hot cells. These are:

- Storage safety significantly dependent on items before storage:

As at SFs storage facilities which are not equipped with hot cells, it is important to confirm that casks loading operations are conducted properly, especially those operations such as selection of fuels to be stored, vacuum dehydration, sealing with metal gaskets, etc. Although storage safety cannot be secured without confirming that these loading operations be conducted securely, these operations are generally conducted by those who are not storage facility operators and by owners of SFs who are utilities. It is therefore important that there should be clear allocation of responsibilities between them.

- Storage safety significantly dependent on the transportability of casks during storage:

In storage facilities that are not equipped with hot cells, it is generally assumed that the safety during storage can be maintained by monitoring that inert atmosphere in cask cavities, and inspection of fuel rods inside the casks are not expected.

In such facilities, those casks whose confinement mechanisms failed in a manner that they cannot be fixed on site (i.e., deterioration of metal gaskets in the primary lid) are not allowed to be stored, and have to be transported to sites where they can be fixed. Therefore, it is important that these casks are designed in such a manner that they can safely be transported to these sites even in cases when their containment mechanisms failed. This is important because storage safety of these casks in storage facilities, in case of failures with containment mechanisms, can only be secured that they can smoothly be transported[4].

- Post storage transport safety significantly dependent on documentation during storage:

When storage period expired, casks that have been stored will be transported to their destinations, whether they are reprocessing facilities or final disposal sites. Although SFs stored in these casks, even after storage of decades, are not expected to experience any significant degradation that would affect post storage transport or their acceptability in their destinations. The basic idea is to conduct inspections based on documentations during storage, but it is still important to consider well ahead on which kind of data can be used to indicate which requirements to be satisfied, because unlike their initial transport from NPPs to storage facilities, in the post storage transport, it is not expected to open the lids of the casks and conduct inspections that would confirm the integrity of fuel rods as well as fuel baskets before

[3] In the US, in order to harmonize this interface issue, it is required that applicants of storage facilities to submit a copy of the Certificate of Compliance issued for the cask under part 71 of 10 CFR (transport section), as well as a safety analysis report showing that the cask is suitable for storage of spent fuel for a period of at least 20 years (72.230 10 CFR).

[4] In Japan, casks that will be stored in such storage facilities are required to equip with independent sealing mechanisms that will work when the primary lids failed. One example of such designs is to equip an independent confinement function (attachable third lid) in these casks.

each shipment[5]. This will indicate interface issue that post storage transport safety depends on documentation during storage.

It is estimated that, after several decades, when the storage period expired, the amount of post storage transport of SFs will consist substantially major portion in the total SFs transported within Japan. This will inevitably increase the importance of considering this issue well in advance from the perspectives of stakeholders.

Interfaces are shown in Fig. VIII–2.

VIII–3. HOLISTIC APPROACH THAT JAPANESE REGULATORY BODY HAS ADOPTED IN ORDER TO SECURE SAFE OPERATION OF STORAGE FACILITY

Recognizing the importance of interface issues listed above, Japanese regulatory body NISA, in 2009, based on the recommendations by the Transport Safety Working Group and Storage Safety Working Group of the Nuclear Fuel Cycle Safety Committee, has announced to adopt the following principles for conducting safety analysis of storage facilities using dual purpose dry metal casks. The main points of the above announcements are as follows:

- To require that transport licensing procedures of the casks to be conducted in line with safety analysis of storage facilities;

In case where applications for spent fuel storage facilities with dual purpose dry metal casks are submitted to the regulatory body, NISA will require that the casks used as storage containers in such facilities are transportable, in the sense that they meet national requirements for transport of nuclear materials based on TS-R-1;

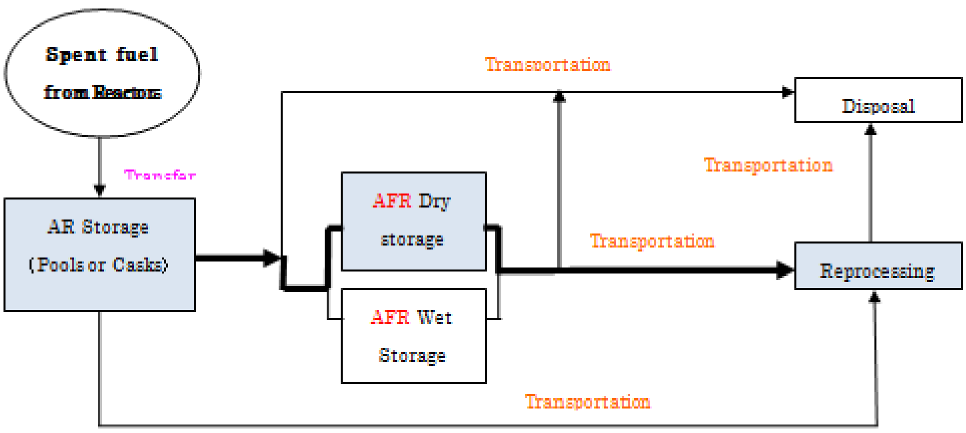

FIG. VIII–2. Interface issues chart.

[5] In Japan, it is common to include visual inspections of contents of casks (fuel rods and fuel baskets) as one of inspection methods that should be conducted before each shipment. These inspection methods are explicitly described in the SARs of SFs transportation casks, and consists part of package design license.

- Also, applicants are expected to demonstrate that the transport license issued to these casks, usually in Japan with 5 years durations, are supposed to be renewed and maintained as effective during the lifetime of each cask.

- To clearly identify the roles of SF producers (i.e., NPP operators) in relation to storage facility operators;

  - In Japan, spent fuel storage operators are supposed to be organizations not that are not SF owners, thus owners of SFs will "entrust" their SFs to the operator of storage facilities;
  - As the safety of the storage facilities cannot be secured unless ensuring that the casks are properly loaded and that the casks will be taken away from the storage facility after the storage duration has been elapsed, it is necessary to clearly identify the role of the owner of SFs, which applicants of the spent fuel storage facilities are expected to demonstrate.

- To identify with foundation that post-storage transport can be conducted safely;

  - In the storage facilities that use dual purpose dry metal casks, storage safety is generally monitored through containment functions, because as long as containment functions are sound, there can be no changes in the atmospheres of cask cavities. Under such situations, fuel rods and cask atmospheres inside these casks are not expected to be directly inspected during storage period;
  - However, even under such circumstances, it cannot be completely denied that there can be some hypothetical occurrences of pinholes and/or hair cracks during storage, although the possibility of such phenomenon is, as long as safety conditions for storage are strictly kept, very low and will not significantly affect the storage safety in any manner;
  - Thus, it is also required to confirm that such phenomenon during storage, if occurred, will not affect post-storage transport safety in any conditions including accident conditions of transport (i.e., 9m drop tests). Therefore, the possible changes (deteriorations) of fuel rods during storage as well as their integrity at the end of designated design lifetime, which are to be examined at the safety analysis of storage facility, should be referred in conducting safety analysis of transport package[6].

VIII–4. DETAILED CONSIDERATION ON THE SAFETY OF POST STORAGE TRANSPORTATION OF SPENT FUELS – INSPECTION METHOD OF TRANSPORT SAFETY BY DOCUMENTATION

The third point in the previous section has already drawn attention of NSC years ago. NSC, in drafting its safety standard on SFs storage facilities using dual purpose dry metal casks in 2002, discussed issues on the safety of post storage transport, and urged relevant regulators as well as utilities the need to consider how to demonstrate it with adequate technical foundations.

[6] Under Japanese regulation, it is generally required to conduct inspections for SFs packages including several visual inspections (visual inspections to confirm that the contents of the package be in sound situation). However, SFs packaged designed considering all these characteristics of dual purpose dry metal casks, these inspections can be substituted by report inspections.

On this issue, the basic technical ideas of NISA at present are as follows:

- In Japan, only intact SFs are allowed to be stored in storage facilities using dual purpose dry metal casks;
  - This requirement is explicitly announced in NSC safety standard in Japan[7];
- Cask loading operations will be conducted based on detailed descriptions by storage operators, which will be incorporated into safety ordinances established with legal enforcements;
- Detailed descriptions will refer the maximum temperature allowed during vacuum dehydration, pressure in cask cavity filled with inert gases, maximum moisture density allowed in cask cavities, as well as maximum allowed leak rate of metal gaskets used in primary and secondary lids;
- Based on current scientific knowledge, SFs deterioration during storage periods, which can be divided into the following four factors, can safely be said to be restricted within safety allowance level throughout storage period, if the points in section 7.3 are provided. Deterioration beyond safety margin can be avoided by monitoring the pressures of the space between primary and secondary lids as well as surface temperature of casks during storage.

VIII–4.1. Deterioration caused by chemical factors

Corrosion of fuel cladding due to atmospheres in cask cavities can be prevented if inert atmosphere can be maintained during storage period.

Thickness of oxide layer on the surface of cladding caused during irradiation in 55 $GW \cdot d \cdot t(U)^{-1}$ has been measured as approximately 20 µms at most for uniform corrosions and 100µms at most for knob-like corrosions (nodular corrosions) for BWRs[8]. Even if all the amount of oxygen contained in moistures (10 wt% maximum) in cask cavity are to react with zirconium during storage, and all of generated hydrogen to be absorbed in fuel cladding, thickness of the oxide film formed is supposed to be below 1 µm level, and increase in hydrogen below 6 ppm level, which are negligibly small, thus will not affect the integrity of claddings[9].

VIII–4.2. Deterioration caused by thermal factors

Breakage of fuel cladding due to progress of creep deformation, reduction of fuel cladding strength due to recovery of irradiation hardening, and embrittlement of fuel cladding due to hydride reorientation can be prevented if the fuel cladding temperature is kept below certain condition, which is to be explicitly defined at the time of designing of casks, during storage period.

The temperature of fuel claddings can reach several hundred degrees in the beginning of storage period due to decay heat caused by fission products (FPs) in fuel rods. As cladding temperature rises, tensile stress act toward fuel claddings will also increase. However, once storage begins, temperatures will drop as decay heat is attenuated with time, and inner pressure drops accordingly.

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[7] This will enable to control the amount of hydrogen generated by radiation under safety level.

[8] JNES, 07-Standard-0002 Report on the demonstration of reliability of high burnup 9 ×9 type fuel in 2006.

[9] JNES, AESJ-SC-F0002-2008, Safety design and inspection standards of the metallic casks for the spent fuel interim storage facilities.

As far as creep is concerned, it has been demonstrated that fuel claddings have the creep deformation margin of more than 1 %, based on tests conducted using zircaloy-2 and zircaloy-4 fuel cladding of BWR 50 $GW\cdot d\cdot t(U)^{-1}$ fuel and PWR 48 $GW\cdot d\cdot t(U)^{-1}$ fuel irradiated in reactors[10].

Also, it has been demonstrated in irradiation hardening tests using zircaroy-2 fuel cladding of BWR fuel and zircaroy-4 fuel cladding of PWR fuel, possibility of recovery of irradiation hardening is small in both types provided that temperature is below certain level[11].

In addition, the fuel cladding temperature and the condition of circumferential stress which will not degrade circumferential mechanical properties of the fuel cladding is already calculated for each fuel type[12].

VIII−4.3. Deterioration caused by mechanical factors

Damage of fuel cladding due to external force during storage is considerably small as compared to Gs during normal conditions of transport. Also, casks will be designed that external forces during normal condition of transport will not affect the safety of casks during transport;

VIII−4.4. Deterioration caused by radioactivity factors

Amount of neutron irradiation during storage is minimal as compared to those in reactors.

- As a result of all the above, post-storage transport safety can be satisfied if casks are designed that would safely accommodate SFs with deteriorations within safety margin (including hypothetical pinholes and/or hair cracks that would occur in non-systematic manner during storage). It is also important to maintain necessary documentations in a retrievable manner for decades.

VIII−5. CONCLUSION

The main points of the above NISA announcement lie in that, in order to assure the safety for storage facilities using dual purpose dry metal casks, it is important to see issues not included in storage period only, but to see interface issues such as interface between storage and transport. Also, this will require that regulatory activities such as safety analysis be conducted in holistic manner, not in independent manner.

In addition, there has been an experience of dry metal cask storage in Japan by two utility companies up to 14 years. One is in Fukushima-Daiichi NPP of Tokyo Electric Power Company since September 1995, and the other is in Tokai-Daini NPP of Japan Atomic Power Company since December 2001. So far, inspection of cask cavities as well as SFs stored has been conducted in each facility, the newest one in Tokai-Daini NPP in January 2009. Based on these inspections (Figs. VIII−3 and VIII−4), it has been confirmed through Kr-85 gas monitoring test that there has been no breach in fuel claddings up to 10 years of storage.

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[10] JNES, 04-Kiroho-0001, (2004).
[11] JNES, 06-Kiroho-0006, (2006).
[12] JNES, 06-Kiroho-0006, (2006) and 07-Kiroho-0004, (2007).

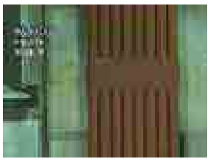
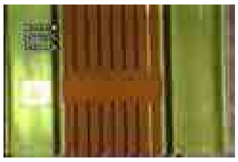
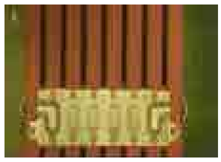
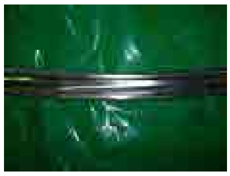
	Fukushima-Daiichi		Tokai-Daini
Fuel Type	8×8	8×8 Retrofit Japan[13]	8×8 Barrier Japan[14]
Storage period at the time of inspection conducted	Approx. 10 years	Approx. 5 years	Approx. 7 years
Burnup	Up to approx. 28,000MWd/t	Up to approx. 32,000MWd/t	Up to approx. 33,500MWd/t
Cooling period in wet pool before storage	Approx. 7 years	Approx. 5 years	Approx. 9 years/ Approx. 8 years
At the time of cask storage began			
At the time of inspection conducted			
Metal gasket of primary lid[15]			

FIG. VIII–3. Results of inspection of fuel claddings conducted in storage facilities attached to NPP sites.

[13] Similar to Pressurized 8X8 Retrofit fuel in the U.S.A. domestic BWR.

[14] Barrier fuel applied to 8X8 Retrofit Japan.

[15] In Fukushima-Daiichi NPP, it was observed that the surface of metal gasket turned white. This is caused because insufficient drying of the flange of primary lid at the time of cask loading for the first one, and because the cask was left in the wet pool for a few days for the second one. In both cases, the results of leak tightness tests conducted indicated the sealing functions have been maintained above required level.

	Fukushima-Daiichi		Tokai-Daini
	Medium Scale	Large Scale	
Leak tightness of primary lid at the time storage began	9.0×10^{-10} (year 1995)	4.9×10^{-10} (year 2001)	1.6×10^{-10} (year 2002)
Leak tightness of primary lid at the time of inspection	5.3×10^{-8} (year 2000)	1.6×10^{-7} (year 2005)	9.0×10^{-11} (year 2009)
Designed leak rate	below 1×10^{-6}		below 1.6×10^{-7}

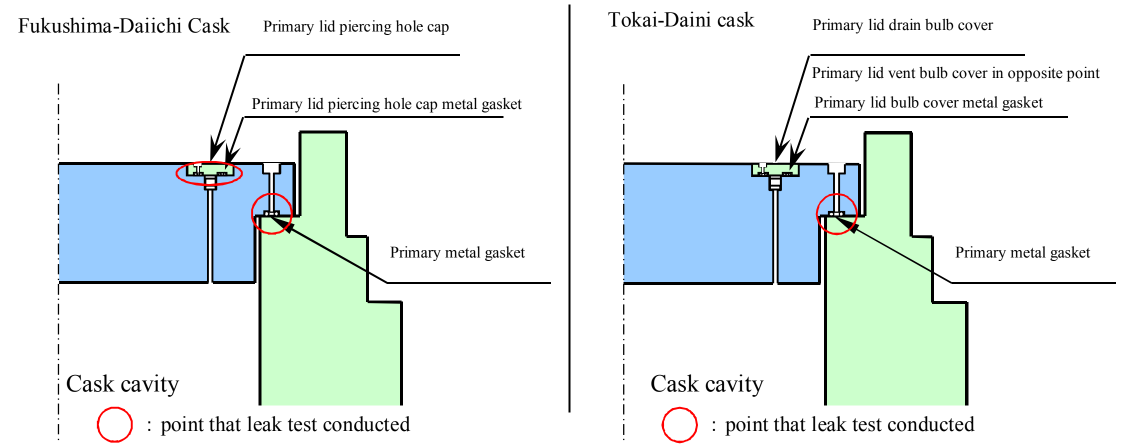


FIG. VIII–4. Sealing mechanisms of the primary lid for the casks inspected (image).

Based on the above announcement, current applicant of the Mutsu storage facility, RFS, has fixed their application accordingly. In addition, Japanese utilities have unveiled a plan to conduct a demonstration test of higher burn up fuels up to 48 $GW\cdot d\cdot t(U)^{-1}$ so far using model scale dry casks, in order obtain data that would facilitate future renewal procedures of transport license for these dual purpose casks. This demonstration is scheduled to begin as early as 2012, and NISA, together with its technical support organization (TSO), JNES (Japan Nuclear Energy Safety Organization), is considering utilizing this demonstration program in order to verify safety margin of dual purpose dry metal cask storage designs.

Annex IX

STUDY OF ALTERNATIVES FOR NUCLEAR FUEL DISPOSAL IN MEXICO

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IX–1. INTRODUCTION

The management of spent fuel from nuclear power plants has become a major policy issue for virtually every nuclear power programme in the world. For the nuclear industry, to find sufficient capacity for storage and processing or disposal of spent fuel is essential if nuclear power plants are to be allowed to continue to operate. At the same time the options chosen for spent fuel management can have a substantial impact on the political controversies, as: proliferation; radiological risks; environmental hazards; economic costs of the nuclear fuel cycle.

Today, in some countries, the spent nuclear fuel is seen as a valuable energy resource, since most of its mass is uranium and plutonium that could be recovered and re-used for additional energy production, these countries have decided to reprocess the spent fuel and recycle the plutonium recovered while vitrifying the residual waste to be buried in deep repositories. Other countries tend to view the spent fuel as a waste, arguing that the cost of recovering its energy content is more than that energy is worth, and that reprocessing and recycling weapons-usable plutonium creates unnecessary proliferation hazards. Today the spent fuel is a potential energy resource whose exploitation could not be economically justified at present, but may become important at some unknown point in the future. Countries that think on that way have plans to direct disposal of the nuclear spent fuel in deep repositories.

In such context, many countries have not defined yet a policy regards the option to manage the nuclear spent fuel, instead have in some way to wait and see what happens in the near future as technology continue been developed.

IX–2. THE CASE OF MEXICO

Mexico has two BWRs reactor operating at Veracruz in the Gulf of Mexico; Laguna Verde (LVNPP). These reactors have an electrical output of 654 MWe each, these two units had been operating since 1990 and 1995 respectively, so the unit 1 has operated 14 fuel cycles and the unit 2 has been operating for 9 cycles.

Currently the cycle length selected for LVNPP is 18 months, to reach this cycle lengths, an average enrichment of 3.7% is required and the average burn-up is between 9500 to 11 000 MW·d·t(U). So the nuclear fuel when is discharged has an average burn-up of around 40 000 MW·d·(Tu)$^{-1}$. The BWRs for LVNPP have a core containing 444 fuel assemblies, so a fuel reload is required each cycle, typically a quarter of core is replaced during reload which means 120 fuel assemblies on the average.

After the operation during 14 cycles the unit 1 has discharged on the order of 1560 fuel assemblies and unit 2 about 1000 FA. That makes a total of 2560 Fas. Each reactor has his own pool for spent fuel, both pools are identical with a capacity of 3177 Fas each. The total capacity of the whole plant is then 6354 fuel assemblies, which is enough to operate the reactors for 32 more cycles before the spent fuel pools will be totally fulfilled. However if the facility wants to extend the life of the plant, some kind of solution will be necessary. By now the capacity for spent fuel storage in our plants is enough for at least 15 more years. However, each interface to implement a solution when time arrives has many problems to solve, and is very convenient to start planning now for the future.

Some studies had been performed to determine which will be the most promising solution in the long run and which the best technology to be implemented. The studies performed include the option for fuel reprocessing and later recycling of recovered plutonium, investigations about direct disposal and interim storage.

IX–3. REPROCESSING AND PLUTONIUM RECYCLING

IX–3.1. Feasibility of plutonium use in BWRS

Exploring the reprocessing option, several studies has been performed at ININ which is the institute for nuclear research in Mexico, these studies include:

- Core performance using plutonium;
- Possible amounts of plutonium existing in the spent fuel;
- Design of MOX fuel;
- Optimum plutonium concentrations;
- Maximum MOX fuel load in core;
- Amount of Spent fuel needed to implement MOX.

IX–3.2. Rights over the fuel

Up to now, the spent fuel stored in reactor pools at the site belongs to the utility, however, as the utility belongs to the government the actual proprietary of the spent fuel is the government, who can decide what to do trough the Energy Secretariat, but the policy has not been defined yet. Another point to take into account, concerned to the international treaties in nuclear and proliferation matters.

IX–3.3. Reprocessing

Mexico do not have reprocessing facilities, if the policy about spent fuel is defined, and the recycling option is selected, the spent fuel would have to be sent to some foreign facility to be reprocessed to obtain the plutonium and later the fabrication of MOX fuel must be contracted. At this point still is not clear, what in that kind of contracts is included and what is not for instance: what happens with radioactive waste and recovered uranium, in addition the transport of fuel and waste should be solved.

IX–3.4. Fuel fabrication

The institute for nuclear research in Mexico, constructed and operated a small nuclear fuel fabrication plant to develop the fabrication technology for UOX fuel, that plant was close in 1996, however the plant still exist and could be adapted to work with mixed oxides to start tests.

IX–3.5. Licensing of MOX fuel

Typically each reload of the nuclear reactor must be licensed by the regulatory body before the new fuel be introduced to the core, for MOX fuel the process should be the same as for UOX fuel.

IX–3.6. Economics

Several scenarios for fuel recycling had been calculated, and finally we arrive that recycling is more expensive that direct disposal of spent fuel, however, as the calculation are based on international assumptions and prices it is possible that real costs differed from calculated.

A summary of the results are found in the Table IX–1. There it may be noted that the "Once Through" scenario is less expensive than the "Recycling" scenario, with a non-trivial difference of 263 million dollars. Note that a constant money calculation has been used to obtain these results.

The costs for final conditioning and disposition have uncertainties attached. These costs were obtained from the literature and are considered generic. Thus the costs for the recycling option can be even higher.

TABLE IX–1. ECONOMIC EVALUATION OF FUEL CYCLE POLICY

	Unit cost		Once through		Recycling	
	USD (Kg HM)	Kg (F.A.)	# F.A.	Cost Billions USD	# F.A.	Cost Billions USD
UOX Fuel Assembly	2123	180	6488	2.479	5868	2.242
MOX Fuel Assembly	1500	180			620	0.1674
Reprocessing	1140	180	6488	0.981	4340	0.8906
Final disposition	840	180			2180	0.3296
Uranium recovered	140	169.2			4340	0.1028
High level waste	-300	7.258			4340	-0.009431
Total				3.460		3.723

IX–4. REPOSITORY FINAL DIPOSAL

The concept of a deep repository has been investigated as a technological option for spent fuel disposal, however, the number of reactors operating in Mexico up to now, probably do not justify the amount of investments needed to consider a deep repository viable since the economical point of view. Even that several proposals for a possible site has been taken into account for a possible site studies and some additional studies on engineering barriers and packing of containers are being planned for the future.

IX–5. DRY STORAGE

Recent Practices in several countries, is the interim storage of spent fuel into special containers designed for that purpose, in such a way that the fuel can remain from 40-50 years before to take a decision what option will be best for final disposition. This technology can be implemented in reactor site or away from site. In Mexico this kind of technology has a high probability to be implemented in a short term waiting for a policy definition; however that is not a cheap solution. Since in one or other way the spent fuel will not be storage in casks forever and something has to be done to solve the problem.

For LVNPP we have calculated taken into account the current operational conditions, will be necessary two BWR casks for each reload of the reactor, so, for 40 years we have 27 fuel cycles 18 months each and 54 spent fuel casks to storage the total spent fuel discharged from one reactor, plus the fuel remaining in the core which amounts for another 3 reloads, that makes a total of: 60 spent fuel casks.

This option will cost for each reactor, taken into account that each cask has a cost of 800 000 USD and the services of filling it which costs from 250 000 to 300 000 USD, that makes a total of 1.1 million dollars each, and for one reactor spent fuel during his useful life the cost will be 66 million dollars plus maintenance and surveillance. In addition an interim storage facility must have to be constructed near the reactor site or away from it with the corresponding costs.

IX–6. RECOMMENDATIONS

The main recommendations for spent fuel management in México are: first to prevent spent fuel pools at the LVNPP to be fulfilled and select one of the available technologies for final disposal, otherwise the best option will be to defer the decision and to use dry storage, then to take a decision 40–50 years from now. Thinking that at the time will be a better or more developed technology.

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Annex X

MONITORING OF NUCLEAR FUEL LEAK TIGHTNESS IN CONDITION OF SLOVAK WET INTERIM SPENT FUEL STORAGE FACILITY AND CONSIDERATIONS ABOUT FINAL DISPOSAL

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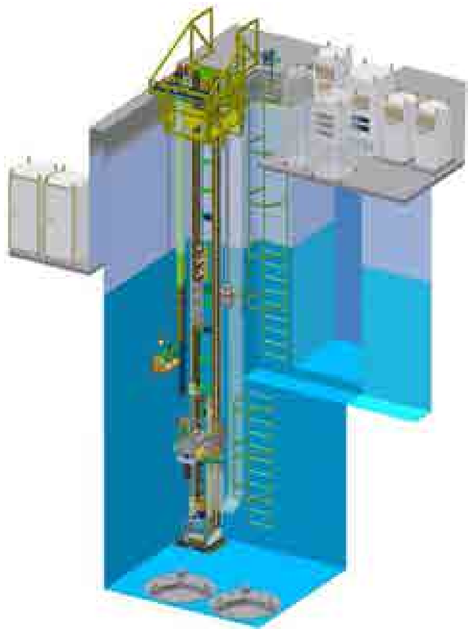
X–1. INTRODUCTION

The Interim Spent Fuel Storage Facility (ISFSF) in Jaslovské Bohunice [X–1], [X–2] is an important component of the spent nuclear fuel management system. The facility has been used for storage purposes since 1987. ISFSF is a nuclear facility providing for a safe storage of the spent nuclear fuel from WWER-440 reactors for the time period of 50 years before the fuel is further processed in a reprocessing plant or appropriately disposed of.

It is necessary to keep the concentration of fission products in storage pools on the low level for assurance of acceptable activity of the coolant. This can be done with periodical monitoring of the fuel elements condition, defects identification and closing of leaking assemblies or fuel elements respectively, in special hermetic caskets. This was the main reason for including not only "Sipping in pool" system, but also inspection stand "SVYP-440" (Fig. X–1), into the ISFSF operation [X–3, X–4].

a) Sipping in pool.



b) SVYP-440.

FIG. X–1. WWER-440 spent fuel assemblies control equipment in Slovakia.

System “Sipping in pool” was built and implemented in the storage facility operation in 1999 and since then, the important results have been measured. The system increases the temperature of the fuel assembly (by external heaters), which cause the increasing of the pressure inside fuel elements. If there is any leakage, increased pressure will cause higher fission product release. By measurement of released activity, the assembly tightness is determined.

Since December 2006, the new stand for WWER-440 fuel assemblies’ inspection “SVYP-440” is in operation. By using several modules, it has ability to open and take the fuel assembly apart, so it can examine all fuel elements. If the defect is found, fuel element with defect is closed into the special hermetic case.

X–2. EXPERIMENTAL

Slovakia has more than 20 years of experience with spent fuel storage. Since beginning, there were no leakages detected during storage conditions. Even though the negative effects of fuel cladding are very low, however, due to degradation of Zirconium alloys after long periods of under water storage, there is a finite possibility of defect formation. It is also difficult to estimate the long term degradation process. With use of systems like “Sipping in Pool” or “SVYP-440”, the leakages of the fuel assembly can be detected.

In that case, it is necessary to have as conservative approach to the extent possible. If we compare the volume activity of released fission products around fuel assembly A_O (measured by “Sipping in Pool”), and the volume activity of fission products inside whole fuel assembly A_I (calculated by SCALE code [5]), we could estimate the fuel cladding condition. Therefore, we are introducing the fuel cladding leak tightness coefficient k_{FCT} (1).

$$k_{FCT} = \frac{A_O}{A_I} \tag{1}$$

For the calculations of volume activity of fission products (in particular, ^{137}Cs has been considered) inside whole fuel assembly, sequence ORIGEN-ARP have been used (for version SCALE 4.4a) [X–6]. As a simplification of the calculations, one model of fuel assembly has been used for all measured assemblies. This model, created by VUJE, a.s. company (Ing. Vladimír Chrapčiak, Ing. Radoslav Zajac, December 2002), is a standard model of WWER-440 fuel assembly, with the UOX fuel and 4.2% enrichment of ^{235}U. Only burn-up, power, effective days during operation and days during shut downs were unique for every assembly.

X–3. RESULTS

After SCALE calculations of volume activity of ^{137}Cs inside the fuel assemblies, and measurements of volume activity of ^{137}Cs outside the fuel assemblies by “Sipping in Pool”, 36 values of the fuel cladding leak tightness coefficients k_{FCT} have been obtained (Table 1). All values were described by distribution function (Fig. X–2) to determine the mean value μ and standard deviation σ. It means, that all fuel assemblies with $k_{FCT} = 1.1 \times 10^{-10} \pm 6.5 \times 10^{-11}$, or ($\mu + 1 \times \sigma$), are without any leakages. Fuel assemblies’ producer criteria for the released γ activity of fission products are 10^{-4} Ci·l^{-1} (3.7×10^{6} Bq·l^{-1}). Because there were only small differences (less the 10^{1}) between the measured summary gamma activity and the separated cesium activity, we compared these two activities.

TABLE X–1. MEASURED AND CALCULATED VALUES A_O AND A_I AND FINAL CALCULATION OF K_{FCT}

Number of FA	A_O [Bq/l]	A_O [Bq]	A_I to 1kgU [Ci]	A_I to 1 FA [Bq]	k_{FCT}
1	639	63900	82,7	3,678E+14	1,73736E-10
2	609	60900	82,7	3,678E+14	1,65579E-10
3	527	52700	115	5,11451E+14	1,0304E-10
4	563	56300	113	5,02556E+14	1,12027E-10
5	535	53500	113	5,02556E+14	1,06456E-10
6	609	60900	118	5,24793E+14	1,16046E-10
7	421	42100	86,4	3,84255E+14	1,09563E-10
8	584	58400	81,3	3,61574E+14	1,61516E-10
9	483	48300	102	4,53635E+14	1,06473E-10
10	539	53900	86,4	3,84255E+14	1,40271E-10
11	484	48400	80,8	3,5935E+14	1,34688E-10
12	536	53600	80,8	3,5935E+14	1,49158E-10
13	627	62700	100	4,4474E+14	1,40981E-10
14	135000	13500000	103	4,58082E+14	2,94707E-08
15	6740	674000	109	4,84767E+14	1,39036E-09
16	74000	7400000	109	4,84767E+14	1,52651E-08
17	372	37200	79	3,51345E+14	1,05879E-10
18	366	36600	79	3,51345E+14	1,04171E-10
19	277	27700	122	5,42583E+14	5,10521E-11
20	441	44100	122	5,42583E+14	8,12779E-11
21	461	46100	125	5,55925E+14	8,29249E-11
22	288	28800	132	5,87057E+14	4,90583E-11
23	437	43700	77,7	3,45563E+14	1,2646E-10
24	310	31000	77,7	3,45563E+14	8,97087E-11
25	505	50500	111	4,93661E+14	1,02297E-10
26	346	34600	111	4,93661E+14	7,00885E-11
27	854	85400	114	5,07004E+14	1,68441E-10
28	238	23800	114	5,07004E+14	4,69425E-11
29	320	32000	75,6	3,36223E+14	9,51748E-11
30	182	18200	75,6	3,36223E+14	5,41307E-11
31	278	27800	126	5,60372E+14	4,96099E-11
32	150	15000	118	5,24793E+14	2,85827E-11
33	366	36600	119	5,29241E+14	6,91557E-11
34	105	10500	124	5,51478E+14	1,90398E-11
35	2030	203000	74,5	3,31331E+14	6,1268E-10
36	3880	388000	74,5	3,31331E+14	1,17103E-09

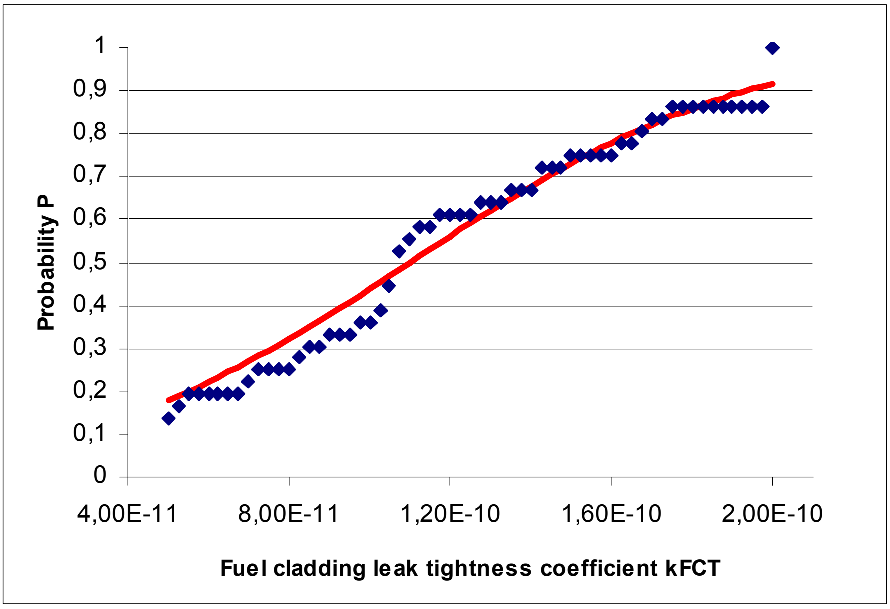


FIG. X–2. Distribution function of continuous variable k_{FCT}.

There were two limit values calculated:

a) $k_{FCT} = 3 \times 10^{-10}$; and b) $k_{FCT} = 8 \times 10^{-7}$.

The first value, $k_{FCT} = 1.1 \times 10^{-10} \pm 1.95 \times 10^{-10} = 3 \times 10^{-10}$, or $(\mu + 3 \times \sigma)$ is from the statistical dispersion of distribution function of continuous k_{FCT}, and means, that 99.73% of the values are within 3 standard deviation. In other words, $k_{FCT}=3 \times 10^{-10}$, there is a 99.73% probability that fuel assemblies will be leak tight.

The second value is calculated using the fuel assemblies' producer criteria $A_{OP} = 3.7 \times 10^{6}$ $Bq \cdot l^{-1}$.

$$k_{FCT} = \frac{A_O}{A_I} = \frac{3.7 \times 10^{8}\,Bq}{4.4655 \times 10^{14}\,Bq} = 8.06 \times 10^{-7}$$

Where

$A_O = A_{OP} \times V_S$;

V_S is volume of the Sipping casket [X–3, X–4];

A_I is average value of the calculated A_I (Table X–1).

With considering of this value, we set the final limit intervals:

1. $(0 - 3 \times 10^{-10})$ - values of the fuel cladding leak tightness coefficient for tight fuel assembly – $k_{FCT}(T)$);
2. $(3 \times 10^{-10} - 8 \times 10^{-7})$ - values of the fuel cladding leak tightness coefficient for fuel assembly suspicious from leakage – $k_{FCT}(SL)$;
3. $(8 \times 10^{-7} - 1)$ - values of the fuel cladding leak tightness coefficient for fuel assembly with leakage – kFCT(L).

As shown in Fig. X–3.

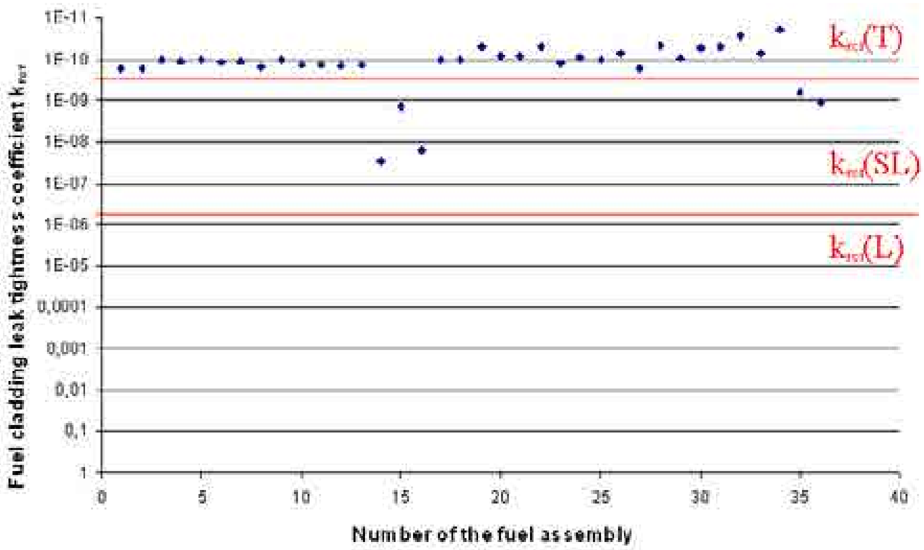


FIG. X–3. Limit values for the fuel cladding leak tightness coefficient k_{FCT}.

The optimal use of the cladding leak tightness coefficient k_{FCT} is its application during periodical monitoring of fuel assemblies. Because of the low level of summary γ activity in the coolant, monitoring of 6 fuel assemblies at ISFSF in Jaslovské Bohunice is provided once per year. Two assemblies are marked as reference, other 4 assemblies are chosen randomly. For those two reference assemblies, values of the cladding leak tightness coefficient k_{FCT} were calculated. For both assemblies, the values of k_{FCT} are about 1.2×10^{-10}. Only during last measurement, few deviations have been observed. The deviation for first assembly is $+7.73 \times \sigma$, and for second assembly is $+16.3 \times \sigma$. This means, that both fuel assemblies are susceptible to the leak tightness. However, during measurements in 2007, values of k_{FCT} increased for every measured assembly. So, the “jumps” can be caused by incorrect measurements by “Sipping in Pool”.

X–4. CONCLUSIONS

From the nuclear safety point of view, it is necessary to keep the fission products inside the fuel elements and to prevent their escape into environment not only during reactor operation or fuel transport, but also during the long term storage of spent nuclear fuel. Therefore, the effective leak tightness monitoring system at all fuel interim storages is necessary. The designed system from 80‘s at the Slovak wet interim storage facility did not assure this task at the desired level, so the system “Sipping in Pool” was implemented in 1999. After several years of its operation, performed measurements showed, that this system is highly effective equipment for fuel cladding defects detection. Since 2006 a new inspection stand “SVYP-440” for monitoring of spent nuclear fuel condition is used as well.

New factor for specifying of spent fuel leak tightness has been introduced. Using the computer simulations (based on SCALE 4.4a code) for fission products creation and measurements by the system “Sipping in Pool”, the limit values of the cladding leak tightness coefficient k_{FCT} has been defined. It is a tool, which is used as additional information, describing fuel cladding leak tightness criteria. Forward-looking, the authors expect that the mean value of k_{FCT} will oscillate about 10^{-10}. Contingent deviations could be caused by incorrect measurements by “Sipping in Pool”, or using of incorrect fuel assembly model for SCALE calculations. Also, the leak tightness coefficient k_{FCT} will depend on the pool water cleaning system. Depending on the residual activity in the pool, the values of k_{FCT} will change. Therefore, further research is needed.

X–5. CONSIDERATIONS ABOUT SPENT FUEL FINAL DISPOSAL

The base for all consideration about this topic is the 'Strategy of a Back-end of Nuclear Energy' in the Slovak Republic, which was issued by Slovak Government at 21.5.2008 and prepared by national nuclear fund of SR. The content of the strategy was stated according to provision of the Slovak Act N° 238/2006 (as amended):

- Objective and time schedule of all activities (in scope of conceptual decommissioning plan, at minimum);
- Technical and technological procedures;
- Proposal for the financial assurance of the Strategy;
- Expected impact to electricity price, price of other goods, economic and social development;
- Expected impact to competitiveness of electricity producers;
- Influence to the balance, assurance and operational reliability of electrical power system of Slovakia and EU;
- Financial assurance of the fund administration;
- Standpoints of the involved state bodies: ministries of health, environment, Nuclear Regulatory Authority.

In consideration about the most proper spent fuel management, the strategy highlights can be summarized in following statements:

- Open nuclear circle (WWER–440 are not licensed for MOX fuel);
- After short-term storage of spent fuel at reactor, interim storage 5–40 years in interim storage facilities at NPP locality;
- As final disposal – deep repository in Slovak republic or at international repository (if such possibility will be available).

ACKNOWLEDGEMENT

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Annex XI

POTENTIAL INTERFACES IN SPENT FUEL MANAGEMENT IN THE SLOVAK REPUBLIC

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XI–1. INTRODUCTION

In Slovakia there are 4 WWER-440 units in operation, 2 WWER-440 units under construction, 2 shut downed WWER-440 units and 1 HWGCR unit under decommissioning. Until 1987 all spent WWER-440 fuel has been transported to the former USSR. Since 1987 the WWER-440 spent fuel is being stored in Interim Spent Fuel Storage Facility Jaslovské Bohunice (ISFSF). Nowadays the amount of spent fuel stored in the ISFSF is about 9500 assemblies.

Fig. XI–1 provides an overview of fuel management in Slovakia.

ISFSF is a temporary solution of the spent fuel “problem”. The fuel cycle back-end strategy adopted by the Government of the Slovak Republic in 2007 considers both final deposition and reprocessing. The Slovak Electric, and JAVYS, joint-stock companies are the only

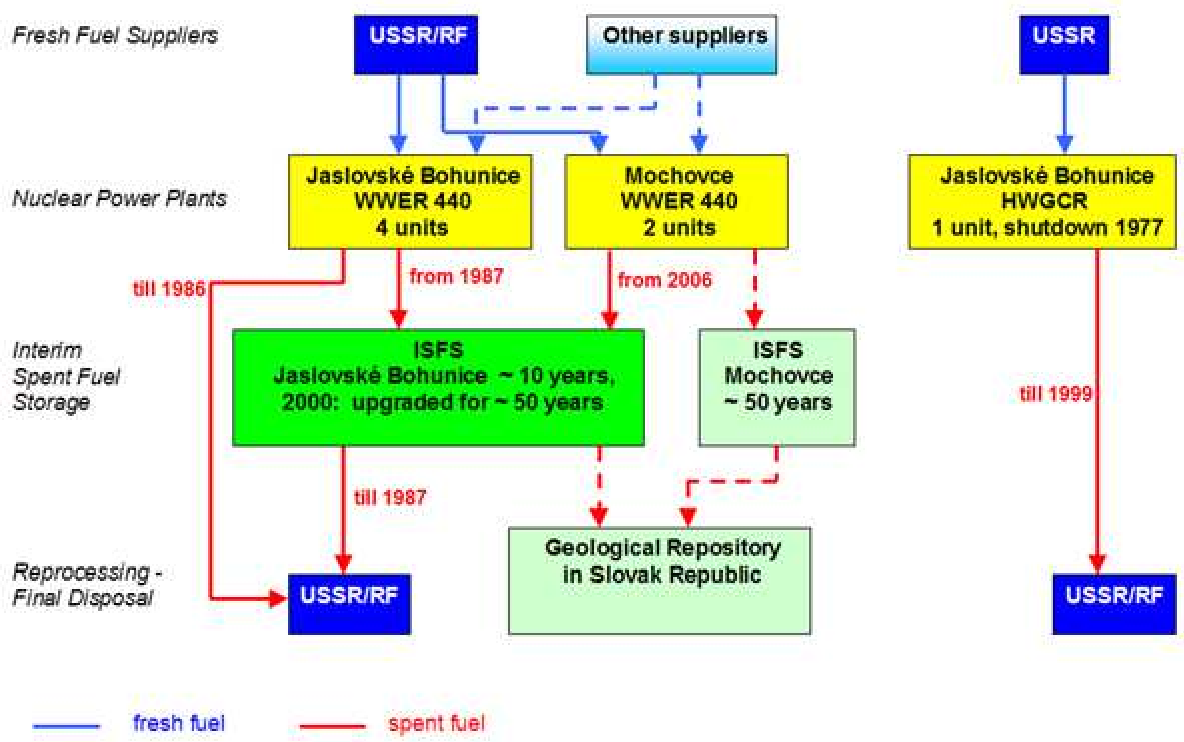

FIG. XI–1. Fuel management in Slovakia.

companies, which have the license issued by the ÚJD SR for spent nuclear fuel handling and transportation. Spent fuel is transferred from reactor core to at-reactor pool for cooling. After 3 to 7 years of cooling time the fuel is transported in transport container C-30 to the ISFSF. Transportation from at-reactor pool to ISFSF is a quite complex problem in solution of which are involved various state and private organization. It is necessary to ensure nuclear safety and nuclear security, confidentiality of information flow and various interactions between consigner, carrier, receiver, state authorities (ÚJD SR, Radiation Protection Authority, Police Corps), and others. Majority of these tasks is regulated by the ÚJD, which also sets the requirements for the transportation of spent fuel in national legislation.

XI–2. SPENT FUEL MANAGEMENT

XI–2.1. AT-reactor pools

The spent fuel is stored inside a base grid located in the bottom area of the spent fuel pool. The base grid of all units at SE-EBO was designed to store 319 spent fuel assemblies and 60 hermetic casings for defective fuel assemblies.

Change No 406/P was approved at both SE-EMO units in 1990 in order to increase the storage capacity of spent fuel pool by installing a more compact base grid.

Based on this change, all the parameters related to cooling, environment protection and seismicity were re-calculated. It was confirmed that the existing systems are sufficient even for the increased amount of stored spent fuel. According to these calculations a new compact grid was designed and manufactured with capacity of 603 fuel assemblies and 54 hermetic casings. A new individual quality assurance program No A19 was approved and implemented for the enhanced compact grid in compliance with the Decision No 105/97 of ÚJD SR.

XI–2.2. ISFSF

ISFSF is a nuclear installation, which serves for temporary and safe storage of spent nuclear fuel from the WWER reactors before its further processing in a reprocessing plant or before its definite disposal. It is designed as a wet storage. It was commissioned in 1986. Active operation began in 1987.

The original design of WWER-440 units presumed, that after three years cooling time in the spent fuel storage at-reactor pools, the spent fuel will be transported into the former Soviet Union. Later, the Soviet side has started to require the storage of spent fuel for at least 10 years in the localities of nuclear power plants. Due to this, a ISFSF was built in Jaslovské Bohunice for the needs of WWER-440 units.

Since 1989 also spent fuel from NPP Dukovany in the Czech Republic was stored in ISFSF. After the construction of storage in the Czech Republic, this fuel has been during 1995–1997 gradually transported back to NPP Dukovany.

ISFSF has been during 1997–1999 reconstructed in order to increase the storage capacity and seismic upgrade. Total storage capacity of ISFSF after reconstruction and seismic improvement is almost three times as big as the designed one - increase from the original 5040 to the current 14 112 spent fuel assemblies (1.694 tHM).

The capacity has been continuously increased through replacement of the original T-12 containers with KZ–48 new containers (completed in 2007) and will be sufficient to store all spent nuclear fuel produced during operation of units Jaslovské Bohunice NPP 1–4. Contrary to Mochovce NPP, in units Jaslovské Bohunice NPP 1–4 it has not come to compacting of spent fuel storage at-reactor pools and the spent fuel is transported from the units Jaslovské Bohunice NPP 1–4 to ISFSF after 2.5–3 years cooling time in spent fuel storage at-reactor pools. Since 2006 the spent fuel from Mochovce NPP is being transported to the ISFSF after 7 years cooling time in at-reactor pools.

XI–2.3. Transport container C–30

For the transport of spent fuel transport container C–30 is used. The original enrichment of nuclear fuel used in WWER–440 units was 1.6, 2.4 and 3.6% of ^{235}U. Since end of nineties a new nuclear fuel with average enrichment 3.82% of ^{235}U (profiled fuel) started to be used for refueling. Profiled fuel enables longer utilization it has higher burn-up, however it also has higher residual heat generation, gamma and neutron emission. C–30 transport container was licensed only for transport of 30 irradiated spent nuclear fuels with original enrichment up to 3.6% of ^{235}U; as given in Table XI–1.

In 2001, Slovak Electric joint-stock company applied for a new license. Application has been thoroughly reviewed by the Nuclear Regulatory Authority of the Slovak Republic. As a result of this review additional requirements have been submitted. In 2004, all requirements were met. A new license was issued. The license enables transport of forty-eight irradiated fuel assemblies with initial enrichment up to 4.4% of ^{235}U; summarized in Table XI–2. New license enables to reduce amount of transports by a third.

After shut down of first unit of V–1 NPP by the end of 2006 the operator, in order to shorten the transition period from operation to decommissioning, applied for a new license for transport of spent fuel form V–1 NPP to ISFSF. The residual heat production for one assembly was increased to 800 W, which is also limit for ISFSF. Also required cooling time was shortened for some assemblies. The ÚJD approved the application and issued a new license (summarized in Table XI–3).

However, after more detailed calculation of spent fuel burnup it came out, that 13 fuel assemblies had slightly higher burnup as 44 $MW \cdot d \cdot kgU^{-1}$. Therefore a new application for transport of spent fuel hit the ÚJD (summarized in Table XI–4).

TABLE XI–1. THE ORIGINAL CONDITIONS FOR C–30 TRANSPORT CONTAINER

	Wet transport	Dry transport
Max. amount of assemblies	30	30
Basket type	T-12	T-12
Max. residual heat	15 kW	8 kW
Avg. burnup	40 $MWd.kgU^{-1}$	14 $MWd.kgU^{-1}$
Max. burnup	44 $MWd.kgU^{-1}$	15 $MWd.kgU^{-1}$
Max. residual heat of one assembly	630 W	-
Max. enrichment	3.6 % U235	2.5 % U235
Min. cooling time	2.5 years	2.5 years

TABLE XI–2. THE CONDITIONS APPROVED IN THE 2004 NEW LICENSE

Basket type	KZ-48	T-12	T-13
Max. amount of assemblies	48	30	18
Max. residual heat	24 kW	24 kW	24 kW
Avg. burnup	50 $MWd.kgU^{-1}$	46 $MWd.kgU^{-1}$	50 $MWd.kgU^{-1}$
Max. Burnup	55 $MWd.kgU^{-1}$	50 $MWd.kgU^{-1}$	55 $MWd.kgU^{-1}$
Max. residual heat of one assembly	605 W	605 W	605 W
Max. enrichment	4.4 % U^{235}	3.82 % U^{235}	4.4 % U^{235}

Minimum cooling time depends on initial enrichment and basket type and varies from 2.8 to 3.6 years.

TABLE XI–3. THE CONDITIONS APPROVED IN THE NEW LICENSE FOR THE MOVEMENT OF V1-NPP FUEL

Basket type	KZ-48	T-12	T-13
Max. amount of assemblies	30	30	5
Max. residual heat	22,4 kW	22,4 kW	4 kW
Avg. burnup	44 $MWd.kgU^{-1}$	44 $MWd.kgU^{-1}$	44 $MWd.kgU^{-1}$
Max. Burnup	44 $MWd.kgU^{-1}$	44 $MWd.kgU^{-1}$	44 $MWd.kgU^{-1}$
Max. residual heat of one assembly	800 W	800 W	800 W
Max. enrichment	3.82 %	3.82 %	3.82 %

Minimum cooling time has been determined to 1.8 year.

TABLE XI–4. THE MODIFIED CONDITIONS APPROVED IN THE NEW LICENSE: FOR THE MOVEMENT OF V1-NPP FUEL

Basket type	KZ-48	T-12	T-13
Max. amount of assemblies	13	13	5
Max. residual heat	10,4 kW	10,4 kW	4 kW
Avg. burnup	44.75 $MWd.kgU^{-1}$	44.75 $MWd.kgU^{-1}$	44.75 $MWd.kgU^{-1}$
Max. Burnup	44.75 $MWd.kgU^{-1}$	44.75 $MWd.kgU^{-1}$	44.75 $MWd.kgU^{-1}$
Max. residual heat of one assembly	800 W	800 W	800 W
Max. enrichment	3.6 %	3.6 %	3.6 %

Minimum cooling time has been determined to 1.95 year.

The license has been prolonged in 2009.

XI–3. CONCLUSION

For the future (2011–2012), the enrichment up to 4.87 % of U235 is planned for use in WWER-440 power reactors. Thus, it will be necessary to re-license all facilities which are used in spent fuel management in Slovakia. However there is a limit for enrichment when calculating the criticality for existing facilities without taking into consideration burnup credit. This limit is 4.4 % of U235 enrichment. Therefore the UJD took an initiative to prepare the methodology for burnup credit implementation. The methodology will contain calculation of actinides and selected fissile products and the UJD will issue it as a guide for operators in

2011. In addition, most probably in 2011, the UJD will issue an amended regulation on spent fuel handling, which will also contain de jure recognition of the methodology.

Annex XII

INTERFACES OF SPENT FUEL MANAGEMENT IN SPAIN: A REGULATORY PERSPECTIVE

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XII–1. INTRODUCTION

The interim storage of spent nuclear fuel (SNF) is currently the keynote of the national waste strategy in Spain. Since pool capacity is being reached in operating plants and to proceed dismantling in those that have been definitely shutdown, there is a growing need for dry storage facilities. The national radioactive waste policy endorsed by the Government, through the 'General Radioactive Waste Plan' (GRWP), establishes the construction of a centralized storage facility to host the spent fuel from all the power plants and some vitrified high level waste reprocessed in France that will be returned to Spain. This storage installation is expected to be operating for a period of 50 years when the geological repository should be available. A consultation and information process remains open for potentially interested communities to host the facility; subsequently the government will decide the final site of the project.

As shown in the Fig. XII–1, the main steps of the spent fuel management comprises transfer to dry storage facilities, storage, transportation of SNF to the CSF and eventually transportation to the disposal repository. Largely all the nuclear material, except a minor amount of vitrified wastes, will be spent fuel assemblies and some non-fuel hardware (NFH); including rod cluster control assemblies, thimble plug devices and neutron sources hardware. From a regulatory standpoint it is essential to have in mind the whole picture since every step has a major influence on the next. For that reason requirements that must be met in the future must be analyzed as early as possible to impose if needed further actions and to acknowledge the burden or limitations that may have an impact on safety. In every step there are also several actors playing a role in the scene and their responsibilities and interactions must be clearly set. Thus in Spain the national radioactive waste management Agency (ENRESA) is the license holder of shipping and storage cask which in turn are used by nuclear facility operator at a site. ENRESA is also responsible for the transportation of SNF and HLW.

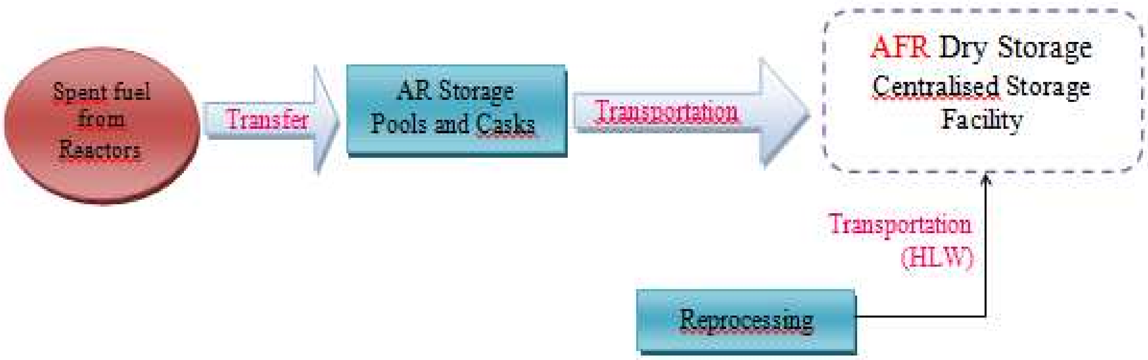

FIG. XII–1. Interfaces of spent fuel management in Spain.

In Table XII−1, the different casks licensed or submitted for approval in Spain are shown.

There are many factors influencing the whole scene, some of them of contradicting or opposing nature, driven by Economics and Regulatory framework changes, the following are just a few:

- Higher burnup;
- Lower decaying periods;
- Uncertainties (Life time of facilities,…);
- Evolving regulations;
- Different actors (shipping storage license holder and nuclear facilities operators, manufacturing companies, dry storage systems designer).

XII−2. SNF STORAGE

Shipping and storage casks need two different licenses according to Spanish regulations. These approvals are granted by the Ministry of Industry with the biding technical report of the CSN. The cask certificate expires 20 years from the date of issuance and normally contains license conditions. CSN periodically inspects the design, fabrication, and use of cask storage systems to examine whether licensees and vendors are performing activities in accordance with safety requirements, certificate conditions and quality assurance program. This quality assurance program undergoes regulatory approval as well.

TABLE XII−1. LIST OF LICENSED CASKS OR CASK LICENSES SUBMITTED FOR APPROVAL IN SPAIN

Cask	HI-STORM 100 storage	HI-STORM 100Z storage	HI-STAR transportation	ENSA-DPT storage and transportation
Fuel type	32 Fuel assemblies (max. 8 damaged SFA) WE PWR 17*17 55GWd/MTU variable decay time	32 Fuel assemblies (max. 8 damaged SFA) WE PWR 14*14 45GWd/MTU 2,5 y.	32 Fuel assemblies (max. 8 damaged SFA) WE PWR 14*14 45GWd/MTU 2,5 y.	21 Fuel assemblies PWR (KWU) 40 GWd/MTU Mín. 5 y. 45 GWd/MTU Min. 6 y. 49 GWd/MTU Min. 9 y.
Max heat load	30kW	30kW	16 kW	25 kW
Designer	Holtec International	Holtec International	Holtec International	Nuclear Assurance Co.
Fabrication	ENSA/HOLTEC	ENSA/HOLTEC	HOLTEC	ENSA
Certificate	Currently under review	2006	Technical position 2009. Not approved yet	2001 (amendments: 2004 and 2009)

The CSN verifies and assesses spent fuel dry storage systems by evaluating each design for safety functions (criticality, containment, shielding, thermal rejection and irretrievability) in normal conditions and in accident conditions such as floods, earthquakes, tornado missiles, and temperature extremes. The requirements for transportation are also accounted for in this stage specially those limiting factors such as fuel reactivity requirements and thermal limitations. In practice the analysis and assessment of storage is done in parallel with transport license.

As a consequence of these limitations the storage certificate has included license conditions to verify transportation requirements. For instance, as prerequisite prior to the actual fuel loading the certificate holder must submit a "Fuel Loading Plan" to ensure spent fuel assemblies, along with the non-fuel hardware (NFH) (including rod cluster control assemblies, thimble plug devices and neutron sources) meet the appropriate requirements for storage and transport in the welded multi-purpose canister. In this particular case cask loading patterns are optimized to maximize heat rejection capability and minimize cladding temperatures, since heat transferred is mainly due to gas convection, the central location in the casks is preferred to higher decay heat assemblies. The extra complexity of these casks loading patterns recommended some kind of regulatory review as well. In such Fuel Loading Plan the following data was required:

- Minimum burn up requirements;
- Maximum burn up and minimum cooling time combinations;
- Decay heat limits per cell and cask, with the assigned decay heats for NFH and neutron sources;
- Allowable locations based on type of NFH or neutron source, maximum burn up and minimum cooling time combinations for transport and storage;
- Date when transportation is feasible.

The recently published CSN regulation IS-20 on safety requirements on the design approval of spent fuel storage casks (Instrucción de 28 de Enero de 2009, del Consejo de Seguridad Nuclear Número IS-20, por la que se establecen Requisitos de Seguridad relativos a Contenedores de Combustible Gastado, BOE nº 42 18/02/09) addresses the importance of the different sets of regulations. Thus, in the overall design criteria in bulletin 3.1.13. it says "Design of storage casks and dual purpose casks will take into account compatibility and dependencies with design criteria for transportation".

This ruling also addresses other issues arising from the interfaces between the license holder and the power station operator licensee, such as:

- Responsibilities for providing with operating procedures, inspection and maintenance;
- Manuals and documentation record keeping;
- Requirements for pre-operational tests;
- Proper interchange of information during storage and operational activities.

XII–2.1. SNF and NFH characterization

The nuclear and physical characterization of spent fuel assemblies and the associated non-fuel hardware is a key element in the subsequent interim storage management programs. Operating power plants must record all relevant data and transmit it to the national radwaste management Agency. In the past years these interfaces arising between operational phases and the subsequent radwaste management have been particularly addressed throughout the CSN oversight process. Regulatory requirements have been augmented as a consequence of this oversight and Licensees of nuclear plants have also raised their awareness in this regard.

Electric utilities have actively collaborated by developing a pilot project to enhance their nuclear database to keep and retrieve the entire relevant database for the next step in the spent fuel management in a coherent and homogenous fashion among the different utilities.

CSN has also enforced some actions in relation to the performance of physical inventory and accounting records of spent fuel pools, especially relevant for NFH.

Recently some issues have been brought up regarding the characterization of spent fuel during visual tests prior to cask loading:

- Significant cladding failures were identified in spent fuel assemblies previously classified as undamaged after UT examinations. As a result of this singular isolated case it has been understood that final fuel classification can only be granted when all the different examination and test methods are completed;
- Presence of small metallic objects was also identified among fuel rods. This issue was not particularly addressed in the cask Storage or Transportation Safety Analysis. Licensee was requested to justify that this unanalyzed condition did not pose an undue burden on safety before resuming cask loading activities. The greater concerns were aimed at potential debris fretting during handling and transportation operations.

Complete characterization of fuel assemblies is a thorough process that comprises several techniques from coolant water chemistry operational data to non-destructive test methods. The classification of fuel damaged or undamaged may have an impact in the transfer activities from wet to dry storage interfacing different actors and for that reason, ideally the complete characterization of spent fuel must be well established as soon as possible to allow prompt and adequate actions to be taken when appropriate.

XII–3. TRANSPORTATION

The Spanish experience with spent fuel transportation casks is limited considering that only Vandellos I Nuclear Power Plant, currently in latency period of decommissioning, regularly sent its spent fuel for reprocessing to France from the beginning of its operation. All the vitrified high level waste from the operation of this plant awaits' to be shipped back to Spain in a short term, when the centralized storage facility is in operation.

During the past years only irradiated fuel rods have been transported as a part of Research and Development programs (2 transports per year, as average). These transports use approved or validated packages, so they do not need any special approval.

The upcoming construction of a Centralized Temporary Storage will require an increase in the number of transports of spent fuel. Some shipments will be performed with casks that were loaded for more than 15 years. At this point, we have two concerns regarding both technical and administrative aspects:

1) From a technical point of view, and in relation to transport, after a long storage period the behavior of that kind of cask as well as the spent fuel loaded is not well known.

In this point, Spain does not have any kind of surveillance program to identify the cask behaviour or the fuel characterisation during storage conditions. The un-analysing safety questions identified during the evaluation process, as well as the problems with the fuel characterisation, described in the previous section, were solved by specific studies and reports before the fuel load;

2) On the other hand, from an administrative point of view, the concern centers on the validity period of approvals of casks in the storage and transport field. There is a different approach in this point due to the fact that the revision cycle of transport Regulations is very short, so the validity period for package approvals has to be shorter (about 5 years) than for storage approval (20 years).

Although changes in last editions of transportation regulations have not affected current package approvals so far, this is an issue that should be considered especially if the period of storage until transportation is increased. It could be that a cask approved for transportation at present may not meet the future transportation requirements when it has to be shipped.

Although a possible solution for those cases would be to transport the packages under special arrangements, the regulatory body considers that special arrangements should only be used for exceptional cases and not to be used as a regular procedure to carry out a transport, as is clearly defined by the transport Regulations.

Anyway, when the package approval must be renewed, the assessment procedure takes into account the regulatory changes happened in the period from the last approval review in order to identify this kind of problems.

XII–4. CONCLUSION

CSN has addressed the interfaces issues between storage and transportation design features including in its regulations the need to account for transportation requirements during the licensing of storage casks. Since storage and shipping casks follow different licensing processes in practice the evaluation and assessment of storage is done in parallel with transport licensees. As a result of these assessments complementary license conditions have been issued in order to verify that both storage and transportation requirements are met before casks loading.

The different boundaries arising during the transfer activities from wet to dry storage have also been accounted for in terms of spent fuel and non-fuel hardware characterization, enhancing databases, relevant data retrieval, improving bookkeeping and physical inventories.

Recent experiences on fuel characterization have raised some concerns on non-analyzed situations on safety assessments for transportation.

There are also some concerns about the future transportation of spent fuel assemblies after long periods of time, especially considering the likelihood of extended storage, and this particularly stands out for high burn-up spent fuel.

Annex XIII

SPENT FUEL MANAGEMENT IN SWITZERLAND

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XIII−1. INTRODUCTION AND LEGISLATIVE FRAMEWORK

In Switzerland, five nuclear power reactors (3 PWR, 2 BWR) at four sites (Beznau, Mühleberg, Gösgen and Leibstadt) are in operation, with a capacity of about 3200 MWe. A total amount of about 4200 tHM of spent fuel (SF) is expected, conservatively assuming 60 years of operation for each nuclear power plant (NPP).

The management of SF and radioactive waste is governed by the federal legislation on nuclear energy. This legislation consists mainly of the following acts and ordinances: Nuclear Energy Act (2003), Nuclear Energy Ordinance (2004) and Ordinance on the Decommissioning and Waste Management Funds (2007, in force since 1st February 2008). Further requirements are detailed in regulatory guidelines. In the year 2000 Switzerland has ratified the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management and has, thus, committed itself to the fulfilment of respective obligations.
Handling of SF may only take place in nuclear facilities. The Federal Council has appointed the Swiss Federal Nuclear Safety Inspectorate (ENSI) as the supervisory authority (regulatory body) for nuclear safety, physical protection, and radiation protection for nuclear facilities as well as the Swiss Federal Office of Energy (SFOE) as the supervisory authority for safeguards. ENSI also supervises the preparatory activities for the disposal of radioactive waste and the transport of radioactive material from and to nuclear facilities. In addition, ENSI is the competent authority with respect to the safe transport of dangerous goods of Class 7 in Switzerland.

The Nuclear Energy Act establishes the need for a series of licenses regarding the implementation and operation of nuclear facilities and the handling of nuclear materials and radioactive waste (SF is considered as a nuclear material, as long as it has not been declared as radioactive waste by its owner). The licensing authority may transfer a license to a new holder if the latter meets the specified requirements.

Handling of SF outside nuclear facilities also requires a license. This obligation applies, for instance, to the transport and to the export for reprocessing of SF. The licensing authority for such licenses is SFOE. Licenses for handling nuclear goods and radioactive waste are non-transferable.

According to the Nuclear Energy Act the NPP operators (licensees) are responsible for the management of their SF as well as for the management of all the radioactive waste produced in their respective facilities, including reprocessing wastes. Since reprocessing is not an option to date in Switzerland (10-year moratorium, specified in the Nuclear Energy Act), SF elements are stored in the appropriate at reactor site (AR) and away from reactor (AFR) interim storage facilities. The NPP operators have to ensure the integrity and transportability of the stored SF elements and high level reprocessing waste and have the obligation to dispose of these items in a geological repository at their own costs.

Moreover, the owner of SF and radioactive waste is committed to keep records of his/her respective material data, including the SF history. When SF is transferred from the reactor pool to a reprocessing or to an interim storage facility, its data are entered into a central data base (ISRAM). For each storage cask, stored in an interim storage facility, the data of the cask itself as well as those of its content, i.e. each individual SF element (or CSD-V canister with vitrified reprocessing waste), are registered in the ISRAM data base. Once the SF/waste is in the interim storage facility, the responsibilities related to record keeping are shared between the respective owner and the licensee of the storage facility, unless both are identical (e.g. AR facility ZWIBEZ).

To ensure the availability of sufficient financial resources for the decommissioning of nuclear facilities and for the disposal of SF and radioactive waste, two respective funds have been established according to the provisions of the Nuclear Energy Act. The Waste Management Fund was established to cover the management costs arising after shutdown of the NPPs. The two funds are independent legal entities. The ordinance on the 'Decommissioning and Waste Management Funds for Nuclear Facilities' defines:

- The allocation of financial resources for the decommissioning and for the disposal of decommissioning waste;
- The allocation of financial resources to cover the costs for the final management of spent fuel and radioactive waste from NPP operation.

Current expenditures related to SF reprocessing and storage of SF and radioactive waste, as well as to research and development, planning, geological investigations and, eventually, construction and operation of disposal facilities, are continuously paid for by the NPP operators.

XIII–2. THE STAGES OF THE SPENT FUEL MANAGEMENT SYSTEM IN SWITZERLAND

XIII–2.1. Reprocessing

According to the Swiss legislation, the NPPs are in principle free to choose between reprocessing and direct disposal of SF. Reprocessing is therefore considered as part of SF management by Switzerland. As there are no reprocessing facilities in Switzerland, reprocessing takes place abroad (France and UK). The 'Nuclear Energy Act' states a series of conditions which must be fulfilled before an authorization of the export of SF for reprocessing can be granted. The conditions include an international agreement with the country of destination, the existence in that country of an adequate facility corresponding to the international standards and that the country of destination has ratified the 'Convention on Nuclear Safety' and the 'Joint Convention'.

Plutonium and uranium recovered from reprocessing are used for fuel fabrication and recycled in Swiss NPPs. The radioactive waste arising from reprocessing is returned to Switzerland where it is subject to interim storage with a view to final disposal. The current contracts between the Swiss NPP operators and the foreign reprocessing companies (AREVA NC in France and NDA in the UK) cover roughly 1200 tU of SF.

By July 2006, when reprocessing of SF was suspended by law for a period of 10 years (moratorium), about 1139 tU of SF had been shipped from the Swiss NPPs to the reprocessing facilities in France and the UK.

By the end of 2008 about 50 % of vitrified high level waste from France has been returned to Switzerland for interim storage. The operator of the reprocessing facility in the UK intends to retain low and intermediate level waste resulting from reprocessing of the Swiss nuclear fuel, and instead to return a radiologically equivalent amount of their own high level waste (substitution). The purpose of this substitution policy, which is intended to be environmentally neutral, is to facilitate and accelerate the return of reprocessing waste from SF of foreign origin and to reduce the number of shipments required. The technical correctness of the equivalence principle and the environmental neutrality with respect to storage and disposal has been verified and confirmed by ENSI. This substitution principle has been approved with regard to the legal aspects by SFOE and signed by the Swiss nuclear utilities in 2007. The return of compacted metallic waste (hulls and ends) from France in CSD-C canisters has started in October 2009. After receipt, the CSD-C is unloaded from the transport casks into purposely designed racks and stored in the AFR interim storage facility.

Since the entry into force of the 10-year moratorium on the export of SF for reprocessing as of July 1st 2006, SF is managed by interim dry/wet storage either with a view to later reprocessing or final disposal. As future reprocessing is still an option, no SF has been declared as waste for direct disposal yet.

XIII–2.2. Interim storage

Since a SF repository is not yet available, SF has to be kept in interim storage at first. Usually SF is held in the on-site reactor pools for some years to allow for decay and cooling before being transferred to a dedicated interim storage facility (unless being reprocessed). Currently there are 3 interim storage facilities for SF in Switzerland: the AFR central dry storage facility (ZZL), the AR Beznau NPP dry storage facility (ZWIBEZ) and the AR Gösgen NPP wet storage pool.

The central storage facility ZZL in Würenlingen is an AFR dry storage facility owned and operated by the ZWILAG Company, a subsidiary of the NPP companies. This AFR facility started storage operation in 2001; it is conceived for the storage of casks with SF or vitrified high level reprocessing wastes from all Swiss NPP. Its storage hall for dry storage of SF and vitrified high level waste can accommodate 200 transport and storage casks. By October 2009, 24 casks with SF and 8 casks with vitrified high level reprocessing wastes (CSD-V) were stored in the ZZL facility.

The ZWIBEZ storage facility of the Beznau NPP is an independent on-site building which has been commissioned in 2008. This AR facility is conceived for dry storage of SF from the two Beznau NPP reactors; it has a capacity of 48 transport and storage casks. By October 2009 two casks with SF were stored in the ZWIBEZ facility; the storage of a third cask with SF is foreseen in February 2010.

The wet storage facility of Gösgen NPP is an independent on-site building which has been commissioned in 2008. This AR facility is conceived for wet storage of SF from the Gösgen NPP reactor; it has a capacity of 1008 SF elements. The SF elements are transferred from the reactor building to the wet storage building in a transport cask. Since the commissioning of the wet storage facility, five transfers equal to a total of 60 SF elements have been performed.

During interim storage the SF and the reprocessing wastes remain in the ownership of the NPP operators. This also applies to the storage casks (and their content) stored in the ZZL facility.

For each type of storage cask the suitability for storage in a specific interim storage facility has to be assessed and confirmed. The assessment is performed by ENSI based on the storage safety case and the requirements that ENSI has set in a dedicated guideline (G05 "Transport and storage casks for interim storage"). Further, according to the Nuclear Energy Ordinance, the storage of each individual cask loaded with SF or vitrified high level waste from reprocessing waste in an interim storage facility needs prior approval from ENSI. In this case ENSI verifies that the radioactive content of the cask complies with the respective specifications set in the safety case and that all conditions resulting from the above-mentioned general storage ability assessment are met.

As the casks are transferred to a dry storage facility, they must have a valid transport license; during dry storage the transport licence is allowed to lapse. However, the forthcoming regulatory guideline (G04 "Requirements for the storage of radioactive waste and spent fuel assemblies") provides for the operator of the storage facility to care for maintaining the storage and transport ability for each cask stored in his facility. Correspondingly, the operator of a wet storage facility shall provide by appropriate means for the integrity of the SF elements and for their transportability. These provisions shall be assessed by the operator of the storage facility in the frame of a Preliminary Safety Report (PSR) scheduled every 10 years.

XIII–2.3. Final disposal

According to the Nuclear Energy Act, anyone who operates or decommissions a nuclear facility is obliged to dispose of all radioactive waste produced at that facility, at his own cost. The obligation to dispose of radioactive waste shall encompass the necessary preliminary activities such as research and geological studies, as well as the provision of a deep geological repository. If a general license for a nuclear power plant has been transferred to another license holder, the previous and the new license holder shall be responsible for the disposal of all waste matter and spent fuel elements produced up to the time of transfer of the license. The disposal obligation shall be deemed to have been met if the radioactive waste has been transferred to a deep geological repository and the funds required for the monitoring period and subsequent sealing, if applicable, have been secured.

According to the current national waste management plan, final disposal of SF shall be performed in a deep geological repository, together high level and long lived intermediate level radioactive wastes. However, no such repository is yet in operation in Switzerland. Siting has started in the year 2008 and is planned to be accomplished in about 10 years by granting a general license by the Federal Council which has to be ratified by Parliament and is subject to an optional national referendum. The site selection procedure for deep geological repositories is specified in the 'Deep Geological Repositories Sectoral Plan' which is coordinated by SFOE. The deep geological repository for the high level and long lived intermediate level wastes is planned to be commissioned by the year 2040 at the earliest.

The intended repository will be a nuclear facility subject to the 'Nuclear Energy Act'. It shall be operated by a dedicated organization (licensee) which will be responsible for its safety and operation.

Before disposal in a deep geological repository, the SF elements and the canisters with the vitrified reprocessing wastes (CSD-V) shall be (re-)packed (encapsulated) into special repository containers which are currently designed to hold two CSD-V canisters, nine BWR SF elements or four PWR SF elements, respectively. One option is that encapsulation will take place in a dedicated surface facility at the repository site; however, no final decision has been taken yet regarding the encapsulation modalities and location.

In order to meet their obligations regarding the disposal of SF and radioactive waste, the NPP operators and the Confederation, which is responsible for managing the wastes from medicine, industry and research, have established the national cooperative for the disposal of radioactive waste (Nagra). Nagra is responsible for developing the disposal routes of all kind of radioactive waste, including SF if declared as such. To make sure that SF (once declared as radioactive waste and subject to disposal) will meet the acceptance criteria of the repository, the licensing procedure of a new type of nuclear fuel includes a statement of Nagra regarding its suitability for disposal; so far, this assessment is based on provisional waste acceptance criteria set by Nagra. Nagra has also access to the central SF/waste data base ISRAM and is involved in the management of this data base.

XIII–2.4. Possible interface issues in spent fuel management

From the above SF management procedures in Switzerland the following potential issues have been identified:

- Need of evacuation (transportation) of a loaded cask from the interim storage facility (e.g. to a reprocessing or disposal facility) while the cask transport licence has expired ;
- Lack of suitable, licensed transport casks for high burnup, MOX or defect SF elements;
- Delays concerning the licensing and/or realization of the deep geological repository leading to durations of interim storage in excess of the planned periods, with possible impact on the integrity of storage casks and/or SF elements;
- Liability issues related to the transfer of responsibility and ownership, which may arise, for instance, i) after delivery of SF (and radioactive waste) to the disposal facility, or ii) in the case of unforeseen events, until respective legal provisions are made and set into force;
- Issues related to the documentation of SF (properties and traceability of history along the SF management chain), particularly with respect to the reliability of and access to the (digitalized) data;
- Issues related to the encapsulation (conditioning for final disposal) of SF and high level reprocessing waste, until decided.

Most of these potential issues shall be deemed to be of general nature and not necessarily specific to the Swiss spent fuel management system.

Annex XIV

MANAGING THE INTERFACES BETWEEN THE FRONT END AND THE BACK END OF THE FUEL CYCLE IN THE UK

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XIV−1. INTRODUCTION

An interface in the physical sense is a transition between one phase and the next; for example moving spent fuel from At Reactor (AR) storage to Away From Reactor (AFR) storage. This transition, however, is not just a physical transition, but is reliant upon a number of interfaces being managed which range from working level interfaces all the way up to satisfying International regulations and Government Policies.

It is not possible in a short paper on the interfaces between the front end and the back end of the fuel cycle to do all of the interfaces justice. In the following paper we start from the various fuel cycle strategies being adopted for spent fuel in the UK and then provide examples of how the interfaces are managed and some of the issues for selected fuel types.

XIV−2. FUEL CYCLE POLICY

The UK Government believes that spent fuel management is a matter for the commercial judgment of the owners of the spent fuel, subject to meeting the necessary regulatory requirements. The exception is for Magnox fuel, where Government Policy is to reprocess.

In the case of non-UK owned fuel (i.e. overseas contracted spent fuel management services), it is UK law that the fuel can only be brought into the country with the intention of reprocessing.

Table XIV−1 summarizes the current fuel cycle policies or options or strategies that have been adopted [XIV−3] in the UK.

TABLE XIV−1. FUEL CYCLE STRATEGIES IN THE UK

Fuel Type	Fuel Cycle Strategy/Option/Policy
Magnox Fuel	Recycle/reprocess
LWR Fuel (UK)	Storage pending direct disposal
AGR Fuel (Historic contracts)	Recycle/reprocess
AGR Fuel (New contracts)	Recycle/reprocess & Storage pending direct disposal[1]
LWR Fuel (Overseas contracts)	Reprocessing, returning products and wastes to customer
New Build (UK)	Planning basis to be used [4], storage pending direct disposal

XIV–3. MANAGING THE INTERFACES BETWEEN THE FRONT END AND BACK END OF THE FUEL CYCLE

The interfaces between the front end and the back end of the fuel cycle are varied and complex reflecting the different fuel types to be managed.

Some examples are provided for a number of the interfaces, outlining the issues and how they are managed.

XIV–3.1. MAGNOX

Fig. XIV–1 shows the Interfaces between the front end and the back end of the Magnox Fuel Cycle. These interfaces are managed by what is referred to as the MAGNOX Operating Plan (MOP).

The MOP is an integrated programme [XIV–2] covering all business areas associated with the cost-effective use and safe post-reactor management of spent MAGNOX fuel.

The MOP was first introduced in 2001 in response to the operational and transportation difficulties that were being experienced, which had led to a build-up of fuel stocks in AR and AFR storage ponds. As MAGNOX fuel cannot be stored 'wet' indefinitely this led to some fuel corroding. The net effect was radiological issues in storage, issues with transporting corroded fuel and issues with decanning MAGNOX fuel in readiness for reprocessing.

The MOP is probably one of the best examples where the interfaces within the fuel cycle are being managed for the benefit of the whole cycle rather than any one element. This may only be achievable because all the elements in the fuel cycle are under the same ownership.

The one example of the interface issues in the MAGNOX fuel cycle and how it is being addressed is the Wet Fuel Stock Policy.

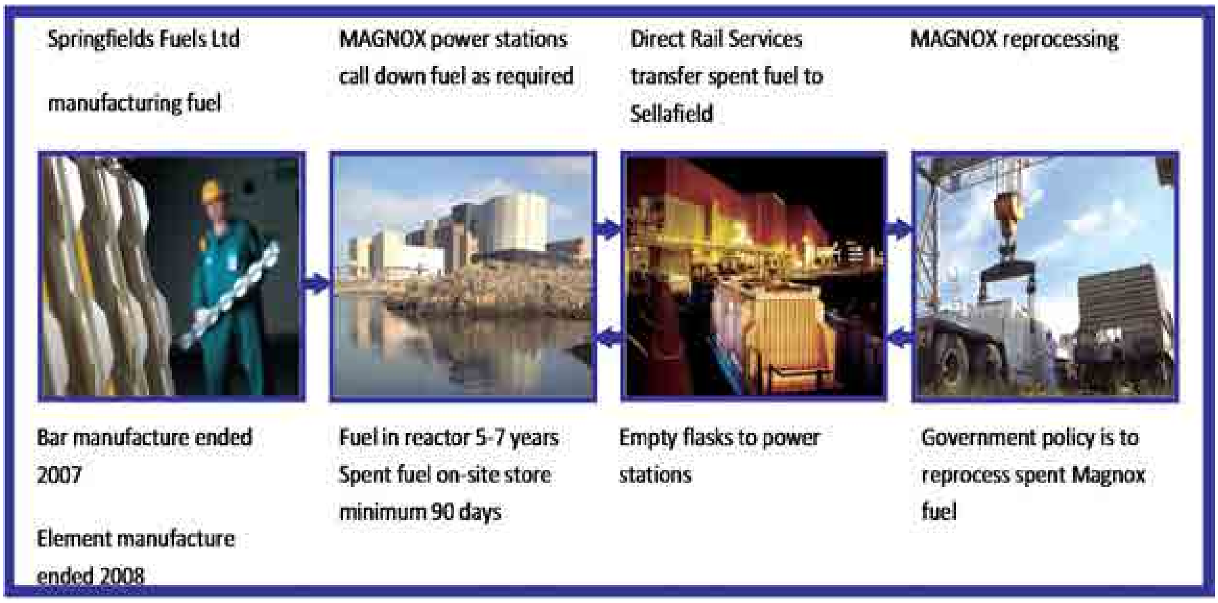

FIG. XIV–3.1. Interface between the front end and the back end of the MAGNOX fuel cycle.

XIV–3.2. Wet fuel stock policy

To avoid the accumulation of 'wetted' MAGNOX fuel and the issues associated with this a Wet Fuel Stock Policy has been derived. It is a mechanism for controlling the amount of Magnox fuel committed to wet storage through balancing the benefits of early defueling of closed reactors against the risk of fuel corroding in storage.

The interface issues are:

- Early defueling of closed reactors and the removal of fuel from the station site enables substantial cost savings through a reduction in resource requirements (that are necessary for fuelled sites);
- Having enough fuel in storage to reprocess as reliability issues of the fuel route equipment arise if they are not regularly used and operators lose their familiarity with the equipment and procedures;
- Too much fuel runs a risk of fuel corrosion if there is a failure of MAGNOX Reprocessing.

A wet fuel stock limit of 800tU +/- 50tU has been concluded from scenario modelling. To control the process a number of action levels have been incorporated into the planning process to prevent the limit being exceeded; these are shown in Fig. XIV–2.

XIV–3.3. Advanced gas reactor fuel

Fig. XIV–3 shows the current interfaces between the front end and the back end of the Advanced Gas Reactor (AGR) fuel cycle.

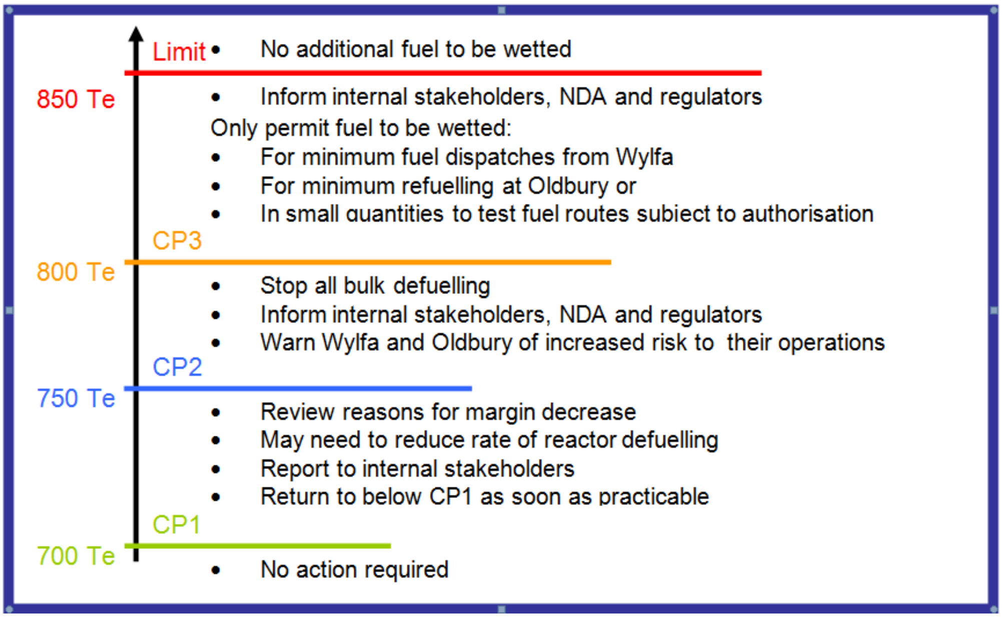

FIG. XIV–2. Wet fuel stock policy control point actions.

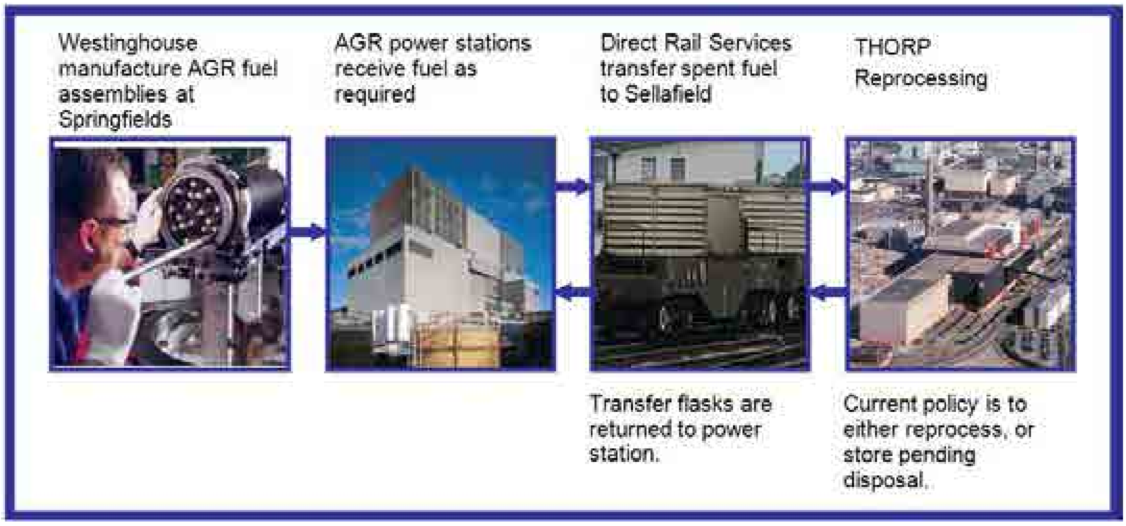

FIG. XIV‒3. Interfaces between the front end and the back end of the AGR fuel cycle.

Unlike the MAGNOX fuel business, there is no equivalent integrated plan, such as the MOP, for AGR fuel; although there are many interfaces to be managed between the front end and the back end of the cycle to ensure successful delivery.

One example of a coordinated approach is the AGR fuel design approval process.

XIV‒3.4. AGR design approvals

The AGR design approval process is a mechanism whereby changes in terms of fuel development, process changes, changes in materials, supplier/sub-contractors are not only assessed in terms of impact on power generation activities/safety cases, but also impacts on the back end of the fuel cycle.

Whilst bounding parameters (maximum fuel enrichment and fuel burn-up) limit what is acceptable in the back end of the fuel cycle, changes within these parameters potentially can also impact on the plants and processes at the back end of the fuel cycle. An AGR fuel element (see Fig. XIV‒4) consists of 36 stainless steel clad pins, a stainless steel central guide tube, two stainless steel spacer braces, a grid which the pins are fixed into and a graphite sleeve.

Fig. XIV‒5 outlines the management of fuel once received at the Sellafield site. Changes, for example, to:

- Impurities in materials;
- Hardness of metals;
- Corrosion performance of materials;
- Changes in pin pitches;
- Changes in how burnable poison cables are attached.

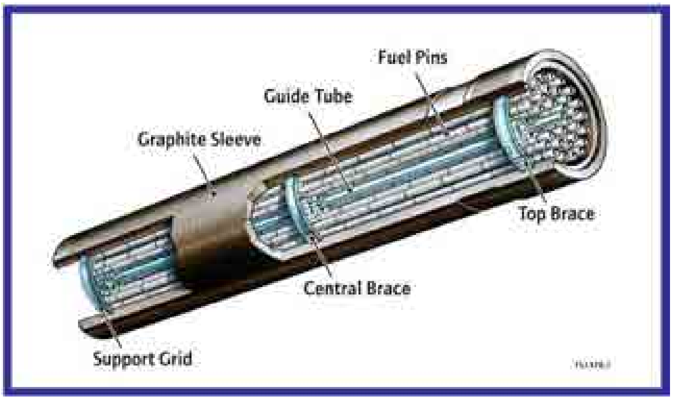

FIG. XIV–4. AGR fuel element.

These can impact on plant specific radiological safety cases, the ability to handle wastes, the ability to store the fuel, the ability to dismantle AGR fuel, performance of compaction equipment, and so on in the back end of fuel cycle.

XIV–3.5. Transition from storage in support of reprocessing activities to one of interim storage

Another interface issue in the AGR fuel cycle is the future transition from one in support of reprocessing activities to interim storage.

The Sellafield Site Life Time Plan [XIV–7] identifies Thorp Receipt & Storage (TR&S) as an option for the future management of unprocessed AGR fuel pending availability of a UK repository. Although this is an existing operational facility with a valid safety case and ‘operating licence’, the forward strategy impacts on continued operations. This transition is an interface between current reprocessing operations and storage/disposal. The impact of this can be separated into two main issues; physical compliance and stakeholder consultation.

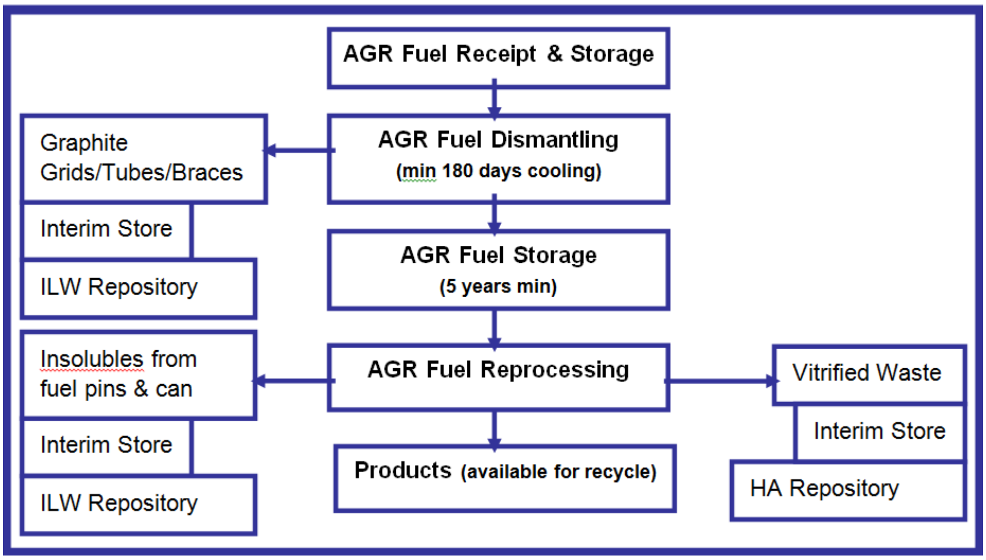

FIG. XIV–5. Management of AGR fuel for reprocessing at Sellafield site.

Physical compliance:

- Planning permission (consent to operate) for AGR Interim Storage. It is likely that the use of TR&S as an interim store for AGR fuel may be considered a material change of use and hence require planning permission;
- Requirements for a PSR and revised Continued Operations Safety Case (COSC) in support of the revised duty;
- Development of a safety case for the interim storage of AGR fuel in dosed water.

Stakeholder consultation:

The Nuclear Decommissioning Authority (NDA) Strategy [XIV−5] made a commitment to explore the long term management options for AGR fuel. This process has involved examining the options available and has involved consulting with key stakeholders in one or more forums as outlined below:

- A Topic Overview Group has been established to review and advise on the strategy development and optioneering studies. This membership consists of interested parties such as Regulators, Government representatives, Department for Transport, Ministry of Defence and the Radioactive Waste Management Directorate;
- A stakeholder engagement plan is being prepared for consultation at the end of 2009. Updates of the development of the Oxide Fuels Management Strategy are given at the bi-annual meeting of the National Stakeholder Group (NSG) and are available for discussion. The NSG consists of 67 different organizations, including representatives from local communities, local authorities, the Government, Regulators and trade unions;
- There are also Site Stakeholder Groups that act as the interface between the local community, the site operator and the NDA. The West Cumbria Sites Stakeholder Group will play an important part during the strategy development for the transition of TR&S to an interim store.

Pending an outcome from the above consultation and the development of a spent fuel management strategy, Sellafield Limited is developing the interim use of TR&S for the bulk storage of AGR fuel to ensure that power generation activities are not compromised.

XIV−4. NEW COMMERCIAL REPROCESSING BUSINESS

An example of a generic overseas fuel cycle is provided in Fig. XIV−6.

The future of Thermal Oxide Reprocessing Plant (THORP) at Sellafield is dependent on the future Government policy [XIV−5]. Any extensions to existing contracts or new reprocessing business are dependent upon Government approval. The Nuclear Decommissioning Authorities (NDA) role would be to inform the decision making process. The likely range of criteria that would have to be met is provided in the Section 4.1 of the NDA Strategy [XIV−5].

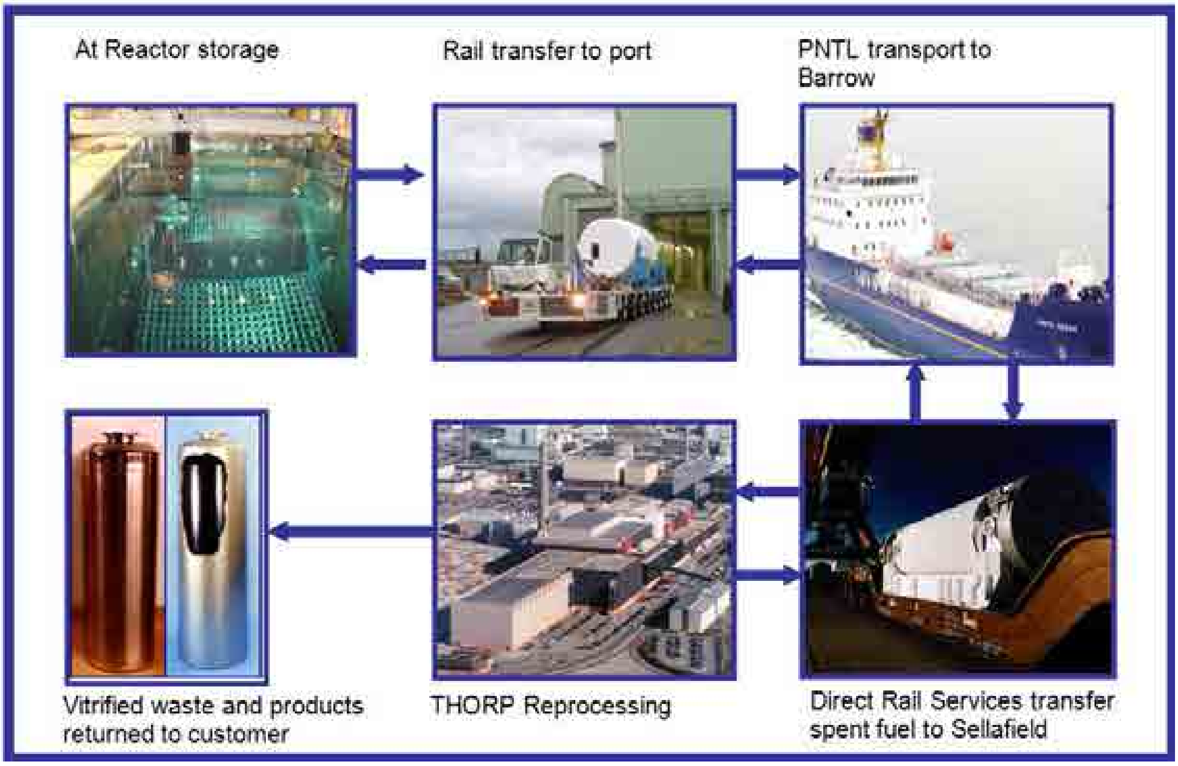

FIG. XIV–6. Generic overseas fuel cycle.

Assuming that the above can be satisfied the immediate interface issues are not associated with the ability to reprocess fuel as this is an ongoing activity, but the ability to transport and receive fuel from overseas. The last international fuel receipt into Thorp Receipt & Storage at Sellafield was in February 2005. The additional requirements requested by customers and to meet individual countries regulatory requirements when handling international fuel receipts compared to national or internal fuel receipts are described in reference [XIV–6].

As stated in the MOP, fuel route equipment works most reliably when it is regularly exercised. Changes in operating regime and improvements in safety related standards necessitate investment in TR&S for plant modifications to allow external flasks to be received and handled safely.

Investment is ongoing in the Pacific Nuclear Transport Limited (PNTL) shipping fleet; the Pacific Heron was launched in May 2007, and a further two vessels are to be launched in 2010/11 to replace those ships that have exceeded their design lifetimes [XIV–8]. Investment will also be required for dockside equipment at foreign ports to bring them up to the same standard as Barrow docks to ensure safe and reliable handling of flasks from train to ship.

A project is ongoing to refurbish the flask maintenance facility and prepare flasks to support future demand.

XIV–5. SUMMARY

This paper has taken examples from the UK to demonstrate management of interface issues in spent fuel management.

For MAGNOX fuel, the interfaces are managed by the MOP, a fully integrated plan that works for the benefit of the whole fuel cycle. The wet fuel stock policy was introduced to balance the benefit of early closed reactor defueling against the risk of corrosion during storage.

There is no integrated plan for AGR spent fuel management, despite there being many interfaces between the front and back ends of the cycle. The AGR design approvals process is applied to ensure changes in fuel fabrication and manufacture do not present a negative impact on power generation or back end activities.

Transition from storage for reprocessing activities to an interim storage facility will encounter many issues that fall under the categories of physical compliance and stakeholder consultation.

New commercial reprocessing business is dependent upon future Government policy. Investment will be required in plant and transport assets to ensure efficient ongoing operations.

ACKNOWLEDGEMENTS

Images used courtesy of the NDA and Sellafield file photographs.

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Annex XV

KEY INTERFACE ISSUES IN SPENT FUEL DRY STORAGE AND TRANSPORT IN THE U.S.A.

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XV–1. BACKGROUND

Spent fuel management options in the U.S.A. have expanded significantly over the last 25 years as the dry storage of spent fuel has increased dramatically. After a period of 12 years or so (from the mid-1980s to the late-1990s) of moving spent fuel from wet storage into dry storage-only systems (licensed only for storage), the U.S.A. nuclear utility industry now places most of its dry-stored spent fuel into technologies that are certified by the U.S.A. Nuclear Regulatory Commission (NRC) for both spent fuel storage and transport, typically called multipurpose or dual purpose systems in the U.S.A.

This places a significant focus at U.S.A. nuclear plants on planning for transitioning spent fuel from wet storage to dry storage, then, hopefully, from dry storage to dry transport. NAC has provided multipurpose canister system (MCS) technology for dry storage of spent fuel, as well as transportation services for spent fuel using its fleet of spent fuel transport casks, for many years. This experience has identified some very important, or key, spent fuel management interface issues, which all organizations planning to transition to dry spent fuel storage or to perform spent fuel transport might wish to review and consider. This paper and presentation offer brief highlights of these key interface issues.

XV–2. OVERVIEW

The task of transitioning from wet storage of spent fuel to dry storage at a nuclear power plant involves a number of activities during the process from conceptualization to actualization. Of supreme importance in this process is the determination of how the dry storage technology will interface with the nuclear plant and its site – what impact will this new storage approach have on plant operations and vice versa, how does the technology fit within the plant and site physical, environmental. and licensing constraints, how will dry storage affect planning for transport of spent fuel to the next stage in that country's nuclear fuel cycle.

In the U.S.A., the dry storage technology of primary choice has become what has been termed canister-based, concrete storage technology. It is typically referred to as MCS technology and uses a thin-shell canister containing a basket structure to hold the spent fuel. The canister has a closure lid that is welded to the canister at the plant, once the canister is loaded with spent fuel, drained of water, dried, and backfilled with an inert gas. Once the canister is fully prepared, it is transferred into a concrete cask or module for long term dry storage. The canister is handled by a transfer cask system during all operations at the plant, from placement into the plant's spent fuel pool, to loading the spent fuel, to the complete closure process, to the transfer of the canister to the concrete storage module. The accompanying presentation presents a summary of MCS operations in proceeding to dry storage and subsequently to off-site transport. Figs XV–1 and XV–2 herein also show these operations summaries for vertical MCS technologies in the U.S.A.

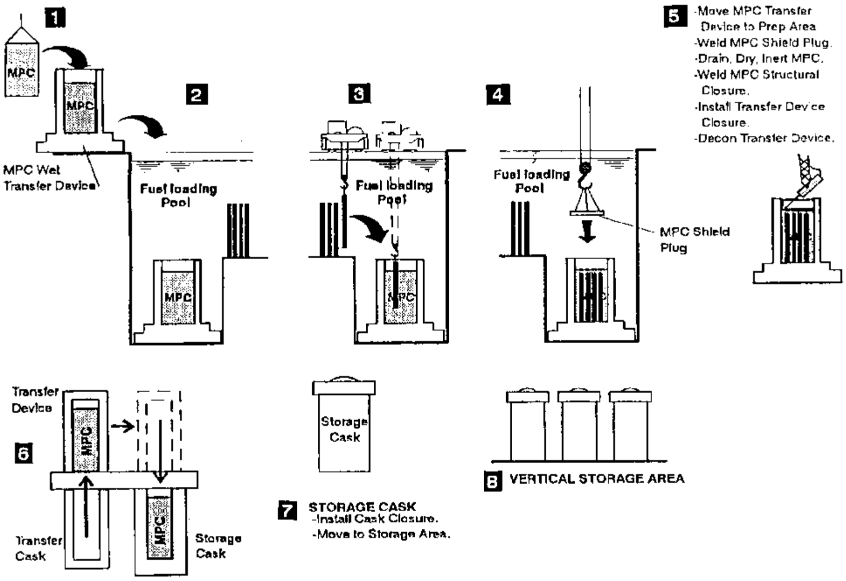

FIG. XV–1. Vertical MCS operations, loading and handling on-site storage.

For the nuclear plant that anticipates the transition to dry storage in MCS or dual purpose systems, there are two principal areas of interfaces that NAC has found to be to be of great importance in planning for dry storage and for transport of spent fuel. The rest of this paper and the accompanying presentation will address these areas at a very high level.

XV–3. KEY INTERFACE ISSUES

The two key interface areas of consideration in this paper and presentation may be summarized as follows:

- Plant Interfaces in the Transition from Pool Storage of Spent Fuel to Dry Storage Systems in the U.S.A.;

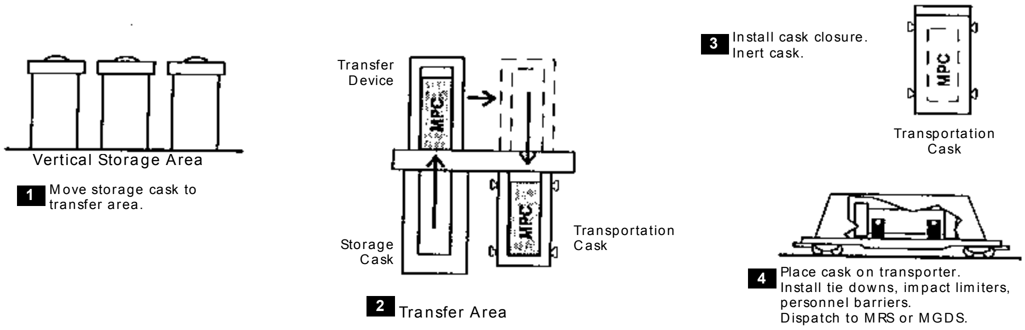

FIG. XV–2. Vertical MCS operations, transport and loading.

- Spent Fuel Transportation and Intermodal Transfer Interfaces for Transport Planning.

The remainder of this paper will address some of the issues associated with these interface areas in the U.S.A.

XV–3.1. Plant interfaces in the transition from pool storage of spent fuel to dry storage systems in the United States of America

In transitioning to dry storage, the importance of the effort spent on the examination and resolution of plant and site interface issues with the proposed dry storage technology cannot be overstated. Additionally, these plant and site interfaces can have flow-down effects on later spent fuel management options, as well.

Plant and site interfaces can be grouped into several classes or categories for the consideration of transitioning from wet spent fuel storage and performing dry storage (or transportation):

- Site environmental conditions (temperatures, wind, floods, seismic, etc.);
- Site physical conditions and limitations (area/size, roads, preferred locations, etc.);
- Plant facility conditions and limitations.

Specific interfaces for each class or category of interface are summarized further below.

Environmental conditions must be addressed by the selected technology. The most important of these is very commonly the site seismic conditions. Seismic evaluations of dry storage technology must be performed at various key points in the operations of the technology, and these include:

- On the storage pad (soil/structure interaction, dynamic amplification);
- At the drain/dry/closure area on the fuel floor (dynamic amplification);
- At the canister transfer area for “stack-up” of MCS components (dynamic amplification).

Site physical conditions and limitations with respect to interfaces with dry storage technology include:

- Dry storage system storage installation area;
- Ancillary equipment storage area;
- Dry storage system haul path to storage installation area;
- Site access roads/rails for equipment delivery;
- Storage system inventory warehousing.

Within the plant itself, there are numerous interfaces that must be considered, since the plant’s capabilities and limitations will determine if the dry storage system can be brought into the plant; loaded, drained, dried, and closed and moved outside to the storage area. Some of these key interfaces include:

- Crane capacities and qualifications;
- Floor loadings (fuel floor, pool, rail bay);
- Plant drop accident licensing bases;
- Wetting of crane hook;

- Refueling floor space for equipment laydown;
- Available utilities (air, water, power, gas bottles, etc.);
- Crane lift height restrictions;
- Rail bay door and airlock clearances;
- Cask handling clearances: pool, rail bay, lift hatch (if applicable), drain/dry/closure/decon area.

For the plant and site interface assessment process, NAC would suggest the following reviews and actions be considered before the dry storage system is selected:

- Review site access roads and roadways to the storage facility;
- Physical features;
- Interferences;
- Subsurface structures (water, electrical, fire);
- Required modifications or upgrades;
- Determine new equipment storage needs, new storage facilities;
- Review/define cask handling facilities and physical interfaces;
- Perform plant/site walk downs of all facilities and haul route;
- Confirm clearances, lift heights turning radii;
- Ensure no interferences or plan to address corrections;
- Review design/licensing documentation for technology;
- Weight limits, plant drop analyses, heavy loads requirements, fuel pool limits;
- Technical Specification requirements for fuel handling;
- Identify potential plant modifications and changes, which may be necessary to accommodate dry cask storage system, for example;
- Single failure proof crane;
- Rerouting of buried utilities along haul routes;
- Cask handling equipment (size, laydown areas);
- Develop design basis publication addressing dry cask storage system handling and placement;
- Prepare specifications for dry storage system procurement.

XV–3.2. Spent fuel transportation and intermodal transfer interfaces for transport planning

In preparing for dry spent fuel storage, the ultimate objective is to transport the spent fuel to the next stage in fuel cycle closure. So, in planning for ultimate spent fuel transport, each plant needs to evaluate how that transport might be accomplished and what considerations for future transport will need to be addressed well-ahead of the actual transports.

Over many years of performing spent fuel transports both in the U.S.A. and globally, NAC has determined that intermodal transports of spent fuel casks are often required. Intermodal transports are those that require a transition from one mode of transport (road, rail, marine, or air) to another before the ultimate delivery of the spent fuel cask.

In the U.S.A., the use of MCS or dual purpose cask systems suggests the strong desire by the purchaser to ultimately perform spent fuel transport directly from dry storage, since MCS and dual purpose systems are more expensive than storage-only systems. Therefore, NAC suggests that each user of dry storage systems review the need for intermodal transfers as part of shipping spent fuel off-site. Such a review may be supported by earlier studies for the plant, but it is recommended that this review be accomplished at planning stage for dry storage of spent fuel at the plant. In this regard, NAC's experience may be instructive and helpful for intermodal transfer considerations, and the following summarizes NAC's experience in the need for intermodal transfers.

For more than 20 years, NAC has performed hundreds of shipments of spent fuel casks around the world. During the period from the mid-1980s through 2007, NAC performed more than 400 spent fuel cask transports. Table XV−1 provides a breakdown of the timing of these transports, how many were intermodal transports, and how many intermodal transfers were required. In general, this spent fuel transport experience may be summarized, as follows:

- Since 1985, NAC has transported 401 spent fuel casks in the U.S.A. and internationally;
- During that period, 232 casks required intermodal transport (58%);
- Many of these transports required multiple intermodal transfers, so that 625 intermodal transfers were required, which is 50% more than the total casks transported and 170% more than the total casks requiring intermodal transport;
- This is an average of 10 intermodal cask shipments each year, and 27 intermodal transfers each year, for 23 years;
- Peak years for intermodal transport were 1988 and 1989, with 20 intermodal cask shipments, and 60 intermodal transfers each year.

What is clear from NAC's experience, based on the amount of spent fuel that is transport internationally, is that global experience may be three times that of NAC. Additionally, intermodal transfers of more hazardous and heavier cargo occur regularly throughout the world and all of this experience shows that correct planning and proven implementation make intermodal transports of spent fuel casks straightforward.

From this record, however, it seems clear that nuclear utilities planning for dry spent fuel storage should begin to evaluate the potential need to transport spent fuel from their sites using intermodal transport. If the need for intermodal transport is confirmed, then NAC suggests spent fuel transport intermodal transfer performance protocols be identified, selected, and developed to an appropriate state of implementation readiness during a transport planning phase, which should be initiated well before actual transports are planned. This intermodal transport planning should include assessment of transfer areas/facilities to determine cask transfer methods, resources required, equipment needs and development lead times, and quality assurance and oversight requirements. Suggested performance protocol considerations include:

- Infrastructure;
- Cranes, fork lifts, man lifts (capacity, lift height, load path, etc.);
- Services (air, water, electricity, etc.);
- Resources (operators, mechanics, security, etc.);
- Access and work space;
- Fences (security control, vehicle access);
- Facility/crane openings (access for forklifts, trailers, mobile cranes, etc.);
- Facility operating space (equipment lay down areas, etc.);
- Quality Assurance and Oversight;
- Procedures/training/testing/inspection documentation;
- Facility work process control.

TABLE XV–1. NAC INTERMODEL SPENT FUEL TRANSPORT EXPERIENCE

Decade	Spent fuel casks transported	Spent fuel casks in intermodal transports	Intermodal spent fuel cask transfers
Late 1980's	161	80	240
1190's	146	99	269
2000's	94	53	116
Total	401	232	625 393 in U.S)

Simply applying the planning, implementation methods, and quality considerations for which the nuclear industry is known, intermodal spent fuel transport can be part of the spent fuel management system without undue risk and without pushing the experience envelope.

XV–4. CONCLUSION

Spent fuel management includes dry storage and transport of spent fuel, and the detailed review and consideration of key system, facility, site, and transport route interfaces remain a most significant part of successful implementation.

Annex XVI

TRANSPORT CONSIDERATIONS IN SPENT NUCLEAR FUEL

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XVI–1. SPENT FUEL MANAGEMENT STRATEGY OPTIONS

The three fuel cycle options are the once through cycle where spent nuclear fuel (SNF) is regarded as waste destined for final disposal, the open cycle in which the fuel is stored on an interim basis and the closed cycle where the SNF is reprocessed and recycled, recovering the uranium and plutonium for further use.

In each case the interfaces between the various stages in SNF management options all involve packaging and transport; namely from short term storage to longer term storage (until the fuel cycle option has been determined), from storage to reprocessing and from storage to conditioning for disposal.

Packaging and transport are essential components of the management of the wastes associated with any of these options. Effective packaging and transport are therefore vital to keeping the nuclear fuel cycle flowing.

XVI–2. REGULATION ISSUES

As for all radioactive materials, the transport of spent fuel at each stage will be governed by the comprehensive regulatory regime which sets the standards for packaging and transport and which is vital to ensure safety. Compliance with the transport regulations is therefore essential in addition to any requirements for storage and disposal and transport considerations must be a vital and integral part of the overall spent fuel management strategy.

Although there are common factors in 'radioactive waste handling', 'storage', 'spent nuclear fuel transport', and 'radioactive waste transport', the risks involved are not necessarily always the same and there are separate regulations (safety standards) for each. It is therefore important to ensure that the regulations governing SNF and waste management and transport are appropriate -, with any conflicts and inconsistencies obviated or corrected.

From the public and industry perspective it is important that the relevant regulatory requirements should meet their safety objectives without imposing unnecessary operational or economic burdens.

XVI–3. ISSUES RELEVANT TO SPENT FUEL AND TRANSPORT

Safety case considerations, the safety case to demonstrate compliance with the regulations for storage or transport will depend on many factors and assumptions, including:

- Developments in fuel design (mechanical, enrichment, MOX, etc.);
- Reactor operating strategies (burn-up, core management, etc.);

- Basis of Safety Case (e.g. methodology, burn-up assumptions, water ingress, enrichment mapping, accident scenarios, etc.);
- Storage time (fuel degradation and containment, deterioration of mechanical properties of packaging, etc.).

The optimum strategy will depend on individual circumstances.

Storage time issues, an increasingly popular option for the management of spent fuel, and some types of waste, is to use dual-purpose casks for both transport and storage, and possibly also for disposal. This may offer significant operational and economic benefits in some situations. The time scales for which regulatory approvals for storage and transport are valid are normally quite different (storage approvals for a long period and transport approvals for a relatively short period). At the time transport takes place compliance with both transport and storage regulations would be required and the safety criteria would have to be demonstrated. There are a number of possible approaches which need to be considered so that operational and safety issues are both effectively catered for.

Interface Issues, the preparation and execution of a major spent fuel transport project often involves many interests such as commercial departments, clients, project managers, regulators, operators, shippers, drivers, emergency response, legal, security, maintenance, QA, purchasing, procurement contractors and sub-contractors.

The extent of so many interfaces clearly demonstrates the need for these to be bound together through robust management arrangements. This, together with a stable and - efficient regulatory regime, facilitates safe and effective transport in which heavy infrastructure investment has been made.

XVI–4. RELATED WASTE MANAGEMENT ISSUES

In addition to the packaging and transport of spent fuel the success of nuclear power operations depends on the effective management of a variety of radioactive waste streams. Issues of importance to safe transport, and which need to be considered in the review of the transport safety regulations to ensure that wastes from nuclear power and fuel cycle plants, including decommissioning, can be accommodated in ways which will continue to ensure both safe and cost-effective packaging, storage and transport include:

High level vitrified waste, high level waste (HLW) from reprocessing of SNF has important characteristics similar to SNF in terms of activity, physical properties, design of containment casks, storage regimes, etc. and similar considerations apply.

Decommissioning wastes, increasing volumes of wastes are arising from decommissioning operations in many countries to clean up the legacy of past nuclear programs and to prepare existing nuclear sites for new nuclear build. The transport of large objects, such as decommissioned reactor components, process vessels and pipe- work, is an important example. Normal types of packaging clearly are not appropriate or always necessary for very large objects and these need to be effectively catered for in the IAEA transport regulations.

Process wastes, a large variety of process wastes have arisen in the nuclear fuel cycle industry, from mining, conversion, enrichment and fuel fabrication facilities to reactor operations and reprocessing. Some of the wastes are not easily characterized.

As in the case of SNF the focus in waste management so far has been mainly on the need to package the wastes to ensure safe, interim or long term storage, in the raw or conditioned state; but eventually the wastes will have to be prepared and packaged in a form suitable for final disposal.

XVI–5. OPERATIONAL EXPERIENCE

Spent fuel has been routinely and safely transported, both nationally and internationally, by road, rail and sea for some 50 years without causing damage to man or the environment. This is the result of an effective transport safety regulatory regime in combination with the professionalism and commitment of the nuclear industry and its transport service providers. The safe international transport of HLW also has a well-established track record. The same is true for operational waste and waste arising from decommissioning activities.

In the case of both SNF and HLW transport public concern and opposition in some quarters has been a serious issue. There is a need to continue efforts to allow the public to understand that such transports are necessary, safe and secure.

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