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SPENT FUEL MANAGEMENT STRATEGIES

(Session 1)

Chairpersons

D. FOX

United Kingdom

INVITED SPEECH 01: NUCLEAR POWER AND THE FUEL CYCLE OPTIONS FOR JORDAN

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Abstract

The Government of Jordan has taken a decision in August 2013 to adopt nuclear energy as part of its energy mix, and the Jordan Atomic Energy Commission has selected RUSATOM Oversees (RAOS) as its strategic partner for the operation of Jordan's first nuclear power plant (JNPP), which consists of two 1000-MW(e) VVER Russian PWRs (based on the AES-92 design). During the next 18 to 24 months, final negotiations will continue with RAOS to finalize all agreements specified in the Project Development Agreement (PDA) and Inter-Government Agreement (IGA) and complete all studies necessary to make a final decision on the construction and operation of Jordan's first NPP. Of these agreements important to the nuclear fuel cycle are the Nuclear Fuel Supply, Radioactive Waste Management, Spent Nuclear Fuel Management, and Decommissioning agreements. The signed PDA and IGA have been formulated to ensure allocation of funds for spent nuclear fuel management, radioactive waste management, and decommissioning of the JNPP. Furthermore, Jordan has obtained commitments to ensure timely supply of nuclear fuel and control rod assemblies during the life time of the JNPP. An option was also obtained by Jordan for the return of spent nuclear fuel to the Russian Federation – an option that will undergo detailed studies to determine its feasibility. Jordan also plans to utilize its vast uranium deposits to provide feed material to fuel its planned JNPP. Forging of the forthcoming agreements with RAOS will also focus on the potential utilization of uranium from planned uranium mine in Jordan that is currently scheduled for operations in 2019, in due to time that hopefully will enable fueling the JNPP with Jordanian uranium. As a newcomer to nuclear power, Jordan intends to gain from past experiences of other countries in order to avoid pitfalls associated with nuclear power plant operations, especially those related to long term management of radioactive waste and spent nuclear fuel. The International Atomic Energy Agency has been and continues to be instrumental in providing Jordan with support to establish a strong, safe, and secure nuclear energy program. This paper intends to present Jordan's efforts and current vision in establishing a strong foundation for its nuclear power plant operations and discusses challenges associated with nuclear fuel cycle management in general and those that are specific to Jordan, especially those related to the agreements that will be forged with the Russian Parties.

POTENTIAL INTERFACE ISSUES IN SPENT FUEL MANAGEMENT

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Abstract

Efficient spent fuel management (SFM) requires an evaluation of the potential interface issues among phases of the nuclear fuel life cycle including: process and equipment compatibility; policy considerations; and different positions of the many stakeholders that influence management options and decisions. Because many issues affect multiple stakeholders and may require long lead times to resolve, it is important to identify interface issues early and solve them in a timely manner. Opportunities are lost if interfaces are not identified and addressed in the early stages of each of the back-end of fuel cycle (BEFC) phases. The objective of this paper is to suggest a process for systematically identifying and evaluating the potential interface issues in SFM, and to recommend effective management based on the experience of Member States before losing timely resolution opportunities. The principles presented in this paper emphasize the importance of systematically identifying and managing interface issues within the BEFC. Because of the complexity of the issues and interfaces, a process is provided to help ensure exact identification of applicable interface issues and consideration of the associated issues and opportunities. Of the conclusions drawn; the most important are: assuring compatibility of schedules, equipment, and acceptance criteria are key to solve interface issues; Interface issue will take on increasing importance as storage periods are extended and countries plan consolidation into regional or centralized dry storage facilities, 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. Accurate information must be provided in a user-friendly format.

1. INTRODUCTION

Most of the spent fuel (SF) from current nuclear power production is stored while awaiting national and international decisions on final disposition. To address a number of potential issues and opportunities, careful consideration of the challenges associated with interfaces in SFM, including extended storage and subsequent transport is appropriate. As an example, IAEA Member States opting for the 'wait-and-see' approach to the backend of the fuel cycle (BEFC) have indicated that one benefit of waiting. A systematic approach is needed to ensure that influences from and impacts on all phases of fuel cycle are taken into account when making decisions. Opportunities are lost if interfaces are not identified and addressed in the early stages of each of the BEFC phases[1]. Because many issues affect multiple stakeholders

[1] A phase is a distinct operation within the BEFC such as wet storage, dry storage transportation, etc.

and may require lead times to resolve, it is important to identify interface issues early and solve them in a timely manner.This paper is an outgrowth of an IAEA Consultancy to address interface issues on the BEFC. Additional detail of this work, such as criteria to identify an interface that are divided into operational, regulatory, safety and security cosiderations, and metholody of gather information about experiences with interfaces can be found in the TECDOC associated with this Consultancy, which should be published shortly. This TECDOC describes a process for identifying and managing interfaces to achieve SFM objectives. The phases and interfaces identified in this TECDOC are not intended to produce a conclusive or fully developed list of issues and recommendations. 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. Extensive exampples of sucesses and failures solving interface issues are also presented in the TECDOC.

The objective of this paper is to suggest a process for systematically identifying and evaluating the potential interface issues in SFM, and to recommend effective management based on the experience of Member States before losing timely resolution opportunities

2. PROCESS

2.1. Development of the matrices

Fig. 1 is a flow diagram showing the major steps of the BEFC. 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. No attempt was made to determine which phase was the driver for the interface, only that an interface existed. Transfer and transport of SF are differentiated. Those activities that move fuel within a site boundary are considered transfers and are not included in the scope of this document. Matrices were developed to facilitate determination of the interfaces associated with the BEFC. Tables 1 and 2 show, respectively, potential interfaces among the phases, and among the participants. The phases were identified from the flow diagram (Fig. 1). There may be additional phases and/or participants specific to a country or a BEFC management strategy. Further, additionally, phases such as transport may have many sub-phases such as sea, ground or intermodal. The methodology provided in this document can easily be expanded to address these additional considerations. In the matrices (Table 1 and 2), a cell entry refers to the section in this paper that discusses associated interface issues.

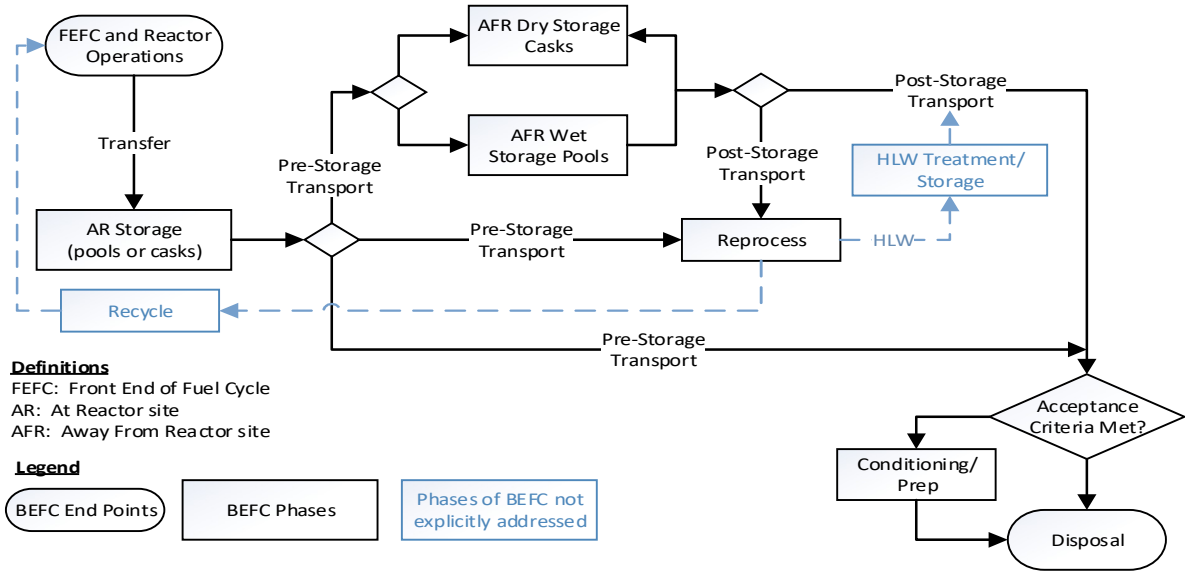

FIG. 1. Flow diagram of interfaces with and within the back-end of the LWR reactor fuel cycle.

3. IDENTIFICATION AND EVALUATION OF INTERFACES

3.1. Phase–phase matrix

The following matrix 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.

TABLE 1. THE PHASE–PHASE MATRIX

Phases \ Phases	Wet storage (AR, AFR)	Dry storage (AR, AFR)	Transport	Reprocess and disposal	Direct disposal
Front end of fuel cycle and reactor operations		§3.1.1			
Wet storage (AR, AFR)				§3.1.1	
Dry storage (AR, AFR)	§3.1.2.1		§3.1.2.2		
Transport					

Numbers in matrix refer to section of paper dealing with this issue.

3.1.1. Cross cutting issues affecting many phase-phase interfaces

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

Fuel design requirements should include BEFC performance considerations. Fuel design and irradiation conditions affect the SF characteristics – which in turn affect the risks, available options and costs of handling, transportation, storage, reprocessing, and disposal. For example, although higher burnup typically reduces front-end costs, it adds cost on the backend by increasing storage times and the potential for personnel exposure during storage, handling, reprocessing, and disposal. High burnup has been found to affect all phases of the management of SF, particularly through increased heat loads and increased susceptibility to degradation [1]. Another example is the affect of fuel design and burnup on cladding condition and its susceptibility to failure during storage and transportation.

Hence, it is in the interest of fuel vendors, 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.

3.1.1.2. 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, equipment compatibility, and information and records needed. The fuel owner is responsible for defining and implmenting a SFM strategy and for ensuring that the SF and any associated packaging fulfill 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;
- The need for further treatment or conditioning of SF;
- Identifying and placing damaged fuel into a condition that meets acceptance criteria for future phases of the BEFC.

3.1.1.3. Major considerations relative to ensuring compatibility with acceptance criteria include:

- Thermal load and dose limits;
- Cladding integrity;
- Dryness;
- Undefined or changing disposal criteria.

3.1.1.4. Knowledge management/adequacy of records

Spent fuel storage periods must be extended until sufficient reprocessing or disposal capacity becomes available. This may take several decades and, as a result, maintaining records needed to demonstrate compliance with technical and safety requirements, facility acceptance criteria, and to facilitate changes in fuel ownership as SF progresses through the BEFC will become increasingly important. During all BEFC phases, responsible management should:

- Maintain a complete information chain of the SF history (fabrication, operations, and storage);
- Establish and conduct an operating experience feedback (OEF) program to collect, screen, analyse and document operating experience and events at the storage facility in a systematic way;
- Gather, safeguard, and periodically up-date digitalized media containing any information considered useful for the future SFM.

3.1.2. Specific evaluation of phase/phase matrix interfaces

3.1.2.1. Wet storage and dry storage

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 to minimize 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 storage, treatment and disposal of the SF;
- In dry storage, control and maintain casks to assure that their safety functions will be not compromised during 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. Examples include: 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;
- At present, there has been little need for moving fuel from 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. Degradation during dry storage is a potential concern.

3.1.2.2. Transport and storage

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). Canisters may degrade due to chloride SCC and jeopodize their abity to act as a barrier for moderator exclusion during transport or use as a containment barrier during future storage. Metal gaskets for dual purpose casks may lose its springback force by creep of aluminum or silver during storage. Metal gaskets should thus be evaluated to ensure they maintain the required containmnet performance for post storage transport [2]. Transportability should be periodically evaluated and maintained.

Pools (and/or large hot cells) may be necessary if they must be opened for unloading, repackaging, inspection, or repair/replacement of components. Pools or hot cells may be needed for planned or unplanned maintenance or inspection of casks and their inventory (e.g. gaskets on primary lids, conditions of SF, etc.). If reflood occurs when the fuel is hot there may be property changes that affect futur behavior in the BEFC. As the lifetime of SF 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.

Damage during pre-storage transport due to mechanical vibration may compromise the containment performance of the storage cask [3]. 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. Likewise these vibrations may challenge the integrity of the fuel cladding if afetr long-term storage the fuel has significantly cooled. Transport licenses may lapse during the duration of the storage license. If the transport regulations have changes the cask or canister may no longer be licensable.

3.2. Participant–participant matrix

The 'participant–participant matrix', Table 2, illustrates potential interfaces among key industry and national Governments participants in the BEFC. Resolution of issues requires interaction among various participants. Potential issues to be addressed at these interfaces are provided along with potential solutions. Addressing BEFC issues is often a costly activity that may require years of persistent effort and investment. 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.

Several participants were combined into one column entitled 'Industry', which includes: Fuel vendor/fuel owner; Cask vendor/cask owner; NPP operator/storage facility owner/operator; 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. 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.

TABLE 2. THE PARTICIPANT–PARTICIPANT MATRIX IN SFM

	Regulator	Bordering/Transit countries	Industry
Policy maker	§3.2.2.1	§3.2.2.4	§3.2.2.2
Regulator	§3.2.2.3		§3.2.2.5
Bordering/Transit countries	§3.2.2.4		§3.2.2.6
Industry	§3.2.2.5	§3.2.2.6	§3.2.2.7

3.2.1. Cross cutting participant−participant 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 SF. The technical, jurisdictional, legal and financial obligations will have to be thoroughly analysed in order to clearly identify the scope of responsibilities of each party. A long-term vision and commitment to implementation is essential for success.

3.2.2. Specific participant–participant issues

3.2.2.1. 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.

3.2.2.2. 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. The back end policy defined by the policy maker determines the waste and SFM paths available.

The availability (or lack thereof) of an endpoint (reprocessing and/or disposal) affects the need for storage capacity and its duration. In countries where a closed fuel cycle is a well defined national policy, indefinite storage of the SF 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 SFM and radioactive waste management (DIR2011/70/Euratom) requires EU Member States to develop their national policy and programme plan by 2015.

Changing or uncertain policies present difficulties. 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 SF 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 SF 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.

3.2.2.3. Regulator — 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.

3.2.3.4. Regulator and border/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 international shipments of MOX fuels or SF 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.

3.2.2.5. Industry and regulator

Regulations that ensure safety and security are essential for effective operation of the BEFC. Transport regulations [4] are available for cross border transportation and are applied for domestic transport in many countries as well. Several IAEA guidance documents are available for use by countries developing storage regulations [5].

Regular interaction between the regulator and industry participants occurs to ensure necessary licenses are in place and followed. 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

Licenses are usually valid for a limited period of time, after which the license must introduce an application for extension or renewal as a condition of continued operation. 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 re-licensing approach. Relicensing considerations include:

- Establishing a path forward for relicensing that assures that storage facilities are either relicensed or SF 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.

3.2.2.6. Industry with bordering and 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.

3.2.2.7. Industry — industry

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. In case where the reactor cooling pool is nearing its capacity, 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.

Effective planning and preparation:

- Participate in user groups to address common concerns and to share technical information, lessons learned, and solutions;
- Transport coordination services:
  - Assure availability of authorized transport routes;
  - Verify operator and carrier organization qualifications, insurance contracts, equipment maintenance, etc.;
  - Organize coordination meetings on a regular basis and keep in close contact with the authorities.

Contingency planning:

- Consider multiple vendors to reduce risks associated with cask availability;
- Plan alternative routes for transport of SF and HLW in order to offer the flexibility to the transport.

Careful coordination and contractual arrangements:

- Ensure clear accountability is maintained throughout the all phases of the BEFC;
- Transfer of ownership is a significant activity involving many parties representing the industry, the regulator, and other affected stakeholders. Transfer of ownership must address 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.

Execution and follow-up:

- Hold regular debriefing and feedback meetings;
- Report off-normal events and other lessons learned that can benefit other industry participants.

3.3. PUBLIC CONFIDENCE

Managing interfaces that affect public confidence in safe and effective SFM 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 SF or waste storage facilities and have even resulted in closure of existing facilities. Conversely, public support can encourage favorable conditions/politics for BEFC nuclear facilities.

- 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;
- Effective communication of the resulting societal and community benefits.

3.3.1. Policymaker/public interaction

Availability of an endpoint (direct disposal or reprocessing and disposal) affects the necessary storage capacity and duration. Commitment to a clear and achievable path to an endpoint gives the public confidence that interim storage facilities will not become permanent and that SF can be successfully managed. 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. Identifying and successfully implementing an endpoint is likely to reduce the perceived risk associated with SF and thus facilitate siting of new facilities.

3.3.2. Regulator/public interaction

Public trust in the competency and effectiveness of the regulatory process is essential for achieving and maintaining stability in a nuclear program. 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. 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. Trust is easily lost. Good practices include for example:

- Maintaining an inspection program that aligns with national standards and practices that are internationally endorsed to ensure operational compliance;
- Hiring and retaining staff with direct nuclear experience;
- 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.
- Explaining and discussing all changes in regulations or guidance with the affected public prior to implementation;
- Balancing transparency/openness with the need to maintain independence and to protect security considerations and proprietary information.

3.3.3. Industry/public interaction

Lack of public acceptance of a path forward (issues with NIMBY) may prevent new facilities and thus results in defacto 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. Providing access to information and facilities is a key to effectively communicating with concerned members of the public. Below are several examples that have proven effective:

- 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; capitalize on economic development/infrastructure attending the facility, etc. Actively engage with the native and local populations to identify and address impacts;
- Encourage beneficial development within the community and region by:
  - Increased employment and local development opportunities to communities willing to host BEFC facilities;
  - Create a citizen's association and local partnerships 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, have helped to reduce the perceived liability of new build;
- Methods tailored for addressing each group help to ensure that their core concerns/interests are fully understood and addressed;
- 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.

3.3.4. Special cases

Industry and regulator with 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. However, citizens and organizations in bordering areas 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.

Transport entities with public:

Public demonstration on the transport route could result in unplanned conditions with the potential for compromising safety and/or security controls[2]. To ensure safe and secure transport, planning should take potential public actions into consideration. Some examples of practices that are considered prudent are: 1) ensure that transport documentation clearly defines the entities responsible for the various transport activities and interfaces; 2) allow time for public in-route interactions with the local population.

4. KEY CONCLUSIONS

- Assuring compatibility of schedules, equipment, and acceptance criteria are key interface issues;

[2] Note that many countries keep transport movements confidential, while in other countries transparency is an obligation except for the transport of materials listed in INFCIRC 225 which must remain secret for security reasons.

- 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;
- Record keeping is an important issue for each interface;
- At present, there is limited experience transporting fuels from a dry storage facility to a subsequent dry storage facility. 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;
- The biggest uncertainty in successfully integrating the BEFC is the uncertainty relative to the endpoint of the fuel cycle;
- The existence and importance of interfaces depend upon the national energy policy, objectives of the nuclear program, maturity of the nuclear program, size of the program, 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.

The principles presented in this paper 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.

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LESSONS LEARNED FROM A REVIEW OF INTERNATIONAL APPROACHES TO SPENT FUEL MANAGEMENT

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Abstract

Worldwide, slow progress in the deployment of geological disposal facilities and reduced use of reprocessing has led to the need to increase the inventory and duration of spent fuel storage. A range of approaches to the management of spent fuel in storage have been adopted in different countries. A review of approaches adopted internationally has been undertaken to inform decision making on spent fuel management. The review surveyed current spent fuel storage and disposal practices, standards, trends and recent developments. National strategies for spent fuel storage and disposal in 16 countries were surveyed and more detailed studies were carried out into the evolution of spent fuel storage and disposal strategies and activities in 4 countries. The review highlighted that: Spent fuel management should be aligned to the national policy for final dispositioning of the fuel; given the long timeframes associated with geological disposal facility site selection and the management life-cycle associated with nuclear fuel, selection of national spent fuel storage arrangements should reflect the need for efficiency of delivery of the whole spent fuel management strategy; it should ensure unduly increased costs due to an expedient short term focus are avoided; commercial and financial arrangements should ensure that spent fuel management decisions do not unnecessarily preclude future management options. This will minimise the constraints placed on future fuel handling, packaging and disposal activities; having the capability to allow for extended storage of spent fuel, either at reactor sites or at a centralised facility, potentially over periods of several decades, may give increased flexibility in the design of future packaging or disposal concepts; storage of spent fuel over 100 years or more using existing technologies, or foreseeable evolutions of them, is technically feasible and operationally credible. The use of multiple approaches to fuel storage, and continued evolution of the design of storage facilities, indicates that there is no single best storage technology and that local factors such as existing infrastructure, approach to fuel cycle management, existing experience/capability and short-term cash flow considerations all influence technology selection. Both wet and dry storage systems continue to receive regulatory approval and are generally considered to be acceptable.

Key Words: Spent Fuel, Irradiated Fuel, Storage, Management.

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SUCCESSFUL STRATEGY DEVELOPMENT IN USED FUEL MANAGEMENT: AN INDUSTRY PERSPECTIVE

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Abstract

This paper explores how industry can help stakeholders to develop and implement successful stainable used fuel management strategies, focussing on the work of the World Nuclear Association Working Group on Sustainable Used Fuel Management.

Key Words: Sustainable Used Fuel Management

1. THE WORLD NUCLEAR ASSOCIATION SUSTAINABLE USED FUEL MANAGEMENT WORKING GROUP: OBJECTIVES AND TERMS OF REFERENCE

Global nuclear capacity is expected to increase by~+50% by 2030. Many countries are looking at expanding their nuclear programmes to increase their energy independence, preserve natural resources and decrease carbon emissions. In conjunction with developing new technologies to minimise waste, the industry are keen to ensure that sustainable used fuel management strategies are put in place in all nuclear programmes.

The main drivers behind establishing sustainable and successful used fuel management strategies include risk reduction, non-proliferation & security, nuclear safety, environmental impact and of course public acceptance. Perpetual storage of used fuel and waste infringe one of the nine principles set forth in the Safety Fundamentals approved by the IAEA's Board of Governors in September 2006. In order to aid industry and other key stakeholders in establishing these management strategies, the World Nuclear Association in 2013 established a working group which provides the nuclear industry with a regular forum to discuss the achievements, best practices and challenges surrounding the sustainable management of used nuclear fuel.

The Working Group is designed to support industrial approaches to:

- Create a sound framework with available industrial solutions with the view to avoid 'wait and see' strategies which create an unresolved issue in the long term;
- Share and promote best practice in used fuel management;
- Encourage national efforts and international collaboration on the R&D of advanced nuclear reactor and fuel cycle technologies.

The working group aims to create a group of industry experts:

- To monitor developments and shape industry positions with the view to engage in the international debate that has emerged post-Fukushima on sustainable management strategies for the back-end of the fuel cycle;
- To develop appropriate information documentation, tools, to inform:
  - The nuclear community and stakeholders;
  - More generally, the public, opinion leaders, and media.
- To develop in particular specific information on key issues of the sustainable used fuel management (SUFM) such as:
  - Conditions and criteria to create a final repository for HLW (administrative, technical, financial);
  - Specificity of the very long term aspects of the SUFM;
  - To consider the possibility of recovering valuable material from the spent fuel.

The Group will support industrial approaches by:

- Putting forward the industry view that promoting ideal but politically infeasible solutions may only postpone decisions that are needed to afford predictability in an industry where rapid and effective development is needed to meet human and environmental needs. Perpetual storage of used fuel and waste is not a sustainable practice and offers no solutions for the future but safety, security and financial risks implications which infringe one of the nine principles set forth in the safety fundamentals approved by the IAEA's Board of Governors in September 2006;
- Offering a range of sound, safe, sustainable and proliferation-proof solutions for the management of used fuel and high level waste and support nuclear user countries, including new comers, to create a sound framework with available industrial solutions with the view to avoid "wait & see" strategies which create an unresolved issue in the long term;
- Sharing and promote among members good practices on all approaches to the SUFM and its wider understanding of the different in the context of nuclear as a sound, safe, sustainable and proliferation–proof form of power. Particularly regarding the interim steps needed due to the unavailability of final disposal facility;
- Considering the feasibility of a multilateral approach between nuclear countries for the development of regional waste final disposal facilities. In such a case, should a genuinely international, non-discriminatory and sustainable solution be available?;
- Encouraging national efforts and international collaboration on the R&D of advanced nuclear reactor and fuel cycle technologies taking into account lessons learned from more than 4 decades of operations and continuous improvement of processes in used fuel management.

The third prong of the working group work is to share the industry's views with other international bodies by promoting SUFM towards the main players among the nuclear international bodies such as mainly IAEA, OECD/NEA and reviewing any new initiatives in connection with the used fuel management. In particular the group hopes to develop active participation in the IAEA, OECD/NEA and other International or key organization and ensure industry's views are considered, develop supporting material, and coordinate sustained participation from industry internationally (events, reviews, etc.) and provide industry feedback accordingly.

2. DEFINITION OF A SUSTAINABLE USED FUEL MANAGEMNT

The Group has an established membership from a wide range of countries including Europe, Russia, Asia and North America. This group will share and promote sound, safe, sustainable and proliferation-proof used fuel management strategies and has succeeded in producing an industry agreed definition of sustainable used fuel management. A used fuel management is considered to be a sustainable option if it adheres to the following key criteria:

- It provides a defined reference plan which proceeds forward in a step-wise manner, with open and transparent decision-making processes to define each step;
- It covers all the steps of used fuel management from the generation of used fuel up to and including final disposal in accordance with a practical plan;
- Takes into account the interdependencies among all steps of the chosen option;
- It proves to be feasible with a sustainable impact level;
- It includes a realistic financing plan.

Examples of a Sustainable Used Fuel Management are:

Recycling (closed cycle):

- Interim (wet or dry) storage of the used fuel, if any;
- Treatment of the used fuel;
- Storage of the vitrified and compacted waste (flexibility on the storage period);
- Final disposal of the waste.

Conditioning or Packaging (open cycle):

- Interim storage (wet or dry) of the used fuel, if any;
- Conditioning or packaging of the used fuel;
- Aging management to assure sustainability of storage methods;
- Final disposal of the conditioned or packaged used fuel.

3. ACHIEVEMENTS TO DATE OF THE WORKING GROUP

The group has conducted a comprehensive survey on current practices in used fuel management, in order to provide a global baseline for used fuel management strategies. This industry-wide survey looked at current policy practises, current discourse, current regulations and what industry would like to see in future to ensure that they are able to contribute effectively to developing a successful strategy for used fuel management.

The efforts of the industry are now being turned to the identification of key messages and best practices, which we will spread to all stakeholders to raise awareness of key issues in developing successful used fuel management strategies. The key messages will include industry views on:

- The definition of a Sustainable Used Fuel Management programme (see § II);
- The characteristics of used fuel;
- Strategic requirements for UFM;
- The prospective for used fuel management inside the nuclear fuel cycle;

- How to produce a defined roadmap for sustainable used fuel management plan up to and including final disposal;
- Site selection;
- Public acceptance;
- The possibilities for a regional repository (as opposed to an international one);
- How UFM will be funded over the long term (securing funding and ensuring that it is adequate).

Key success factors identified are as follows:

- How to produce a defined roadmap for sustainable used fuel management plan up to and including final disposal;
- Site selection;
- Public acceptance;
- The funding.

These key messages will be tailored to various different stakeholders: one of the key industry considerations with the development of a comprehensive used fuel management strategy which extends the final disposal of used fuel or any waste produced in the treatment of used fuel is the extensive role which government must play in many countries. The group is aiming to ensure that key stakeholders in governments are aware of the importance of establishing comprehensive used fuel management strategies, and are therefore hoping to further develop relationships with the IAEA and other key groups.

4. THE USED FUEL MANAGEMENT REPORT

The group aims to produce a comprehensive report to compliment IAEA efforts in a similar direction to provide key stakeholders with a clear overview of the current situation and the needs and aims of industry in the framing of strategies. The objective of the report is to provide an overview and a shared vision of the used fuel management perspectives worldwide for the next 20 years. This report will include:

- The definition of a sustainable strategy, as endorsed by the industry (above);
- An explanation of the characteristics of used fuel;
- Quantitative forecasts for yearly used fuel unloading across all civilian reactor types;
- Storage: wet and dry, and on- and off-site options;
- The reprocessing of used fuel;
- Other management options, both long and short term solutions. The report will evaluate the sustainability of these options according to our agreed industry definition;
- The prospects for final disposal;
- The possibilities for a regional deep geological repository;
- Various option for the secure funding of used fuel management for the whole lifecycle.

This report will be used to feed into a communications plan which the World Nuclear Association hopes to promote through its members in the global nuclear industry to encourage the development of a sustainable and successful strategy for used fuel management.

MANAGEMENT OF SPENT FUEL FROM PHWR'S IN INDIA – AN INTEGRATED APPROACH

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Abstract

India has adopted closed fuel cycle to complete the three stage nuclear power programme envisaged by great founder and nuclear scientist, Homi J. Bhabha, which would provide energy security for the country. The spent fuel from Pressurised Heavy Water Reactors, the mainstay of nuclear power production in the country, are processed in reprocessing plants at Tarapur and Kalpakkam. In order to implement stage II of Indian NPP i.e. Fast Breeder Reactors (FBRs) are planned and fuel for these FBRs will be generated at large scale Integrated Recycle plants being set up for the first time. India has mastered the reprocessing and waste management technology indigenously for PHWR and FBR fuels and has been operating plants since last four decades. The large throughput plants are based on this technology incorporating improvements based on the feedback from the operating facilities taking into consideration of economics of construction and operation. This paper describes the details of Integration approach, spent fuel storage, processing and waste management being planned for large scale recycle plant.

Keywords: Nuclear Power Programme (NPP), Spent Fuel (SF), PHWRs (Pressurized Heavy Water Reactors), high level waste (HLW), low & intermediate level (L&IL) long lived solid waste, vitrified waste product (VWP), vitrified waste storage facility (VWSF).

1. INTRODUCTION

India has adopted closed fuel cycle strategy for long term energy security to counter the prevailing depleting fossil fuel reserves. Indian nuclear power programme is based on three stage strategy envisaged by Dr. Homi Bhabha. The PHWR type of nuclear power reactors and the reprocessing of the spent fuel play a big role in this Indian strategy. In this context, an integrated recycle facility at Tarapur is the first high throughput, integrated back end fuel recycle facility in India aimed at processing Spent fuel from PHWRs (both 220 MW(e) and 540 MW(e)) and is presently under construction stage. India has recycle plants located at Tarapur & Kalpakkam and these sites have associated waste management facilities for management of low, intermediate and high level and non-high level wastes co-located with the processing plant. Recycle & waste management plants are being integrated for the first time in a new integrated recycle facility, into a single entity by locating all the civil structures in a single campus with suitable connectivity between them.

This large throughput integrated plant is based on technologies developed in research and developmental efforts at BARC which also incorporates design improvements based on the feedback from the operating facilities. It also gives major considerations with regard to economics of construction and sustained operation. The plant design is based on 'Solid–in & solid–out' concept with spent fuel as input and uranium, plutonium and solid waste as the output from the plant. The facilities in the plant will have common utilities, maintenance section, control laboratory, chemical stores, general stores and administration systems.

The radioactive wastes emanating from these plants will be suitably managed in the various blocks segregated based on radioactivity level. This facility is designed to address higher reliability and plant availability by grouping equipment into appropriate locations for facilitating maintenance/replacement. An example is the location of all waste evaporation systems in the evaporator block. Provisions will be incorporated for partitioning of high level

waste with an aim of reducing, radio toxicity and waste volumes for geological disposal and to achieve resource utilization. The evaporated high level waste / partitioned waste will be taken up for verification in the waste treatment & verification block of the plant. Appropriately designed facilities for intermediate level wastes and organic waste treatment are also considered in this bock. The vitrified wastes will be stored in vitrified waste storage facility and alpha decontamination / storage block Setting up and operation of an Integrated Recycle facility will mark a milestone in the evolution of back end of the fuel cycle as a matured technology in the Indian context.

2. INTEGRATION APPROACH

Following are the concepts used for design of an Integrated Recycle Facility:

- Block concept — process wise segregation for ease of construction and operation;
- Common utilities isolated from main plant buildings for economization and out sourcing of O & M personnel;
- Solid–in and solid–out concept;
- Layout as per material flow and segregated as per radiation zoning concept;
- Standardization of components for lower spare parts inventory;
- Modular construction for remote / quick replacements;
- Improved design features based on R&D experience and feedback from operating plants;
- Process control room, common main control room and block wise local control room concept.

This approach leads to reduction in overall plant area, reduction of manpower, project cost, reduction in operating cost and increase in efficiency as management of different processes is under single entity. Some of technical advancements done are given in proceeding paragraphs.

2.1. Spent fuel storage facilities

The integrated recycle plants will get feed of spent fuel from 220 MW and 540 MW PHWRs. These spent fuel bundles having burn up of about 8000 MWd/t will be transported to an Interim wet type Spent Fuel storage facility using qualified lead shielded shipping cask [1]. In spent fuel storage facilities, the advanced safety features provided are such as: Infiltration system for keeping the ground water table below the raft of the pool; Leak detection system for pool liners; Pool bridge design with automation features for positioning; Locators for seismic stability of stack of trays and single failure proof cask handling crane qualified for seismic event; The pool walls are lined with 3 mm thick SS plate and floor is lined with 6 mm thick SS plate to avoid ingress/egress of pool water; The fuel pool is provided with an elaborate leak collection/detection system.

> "*The leak collection channels on the backside of pool liners play the dual role of supporting the liners and collection of any water leakage through the welding. These channels are connected to pipe headers terminating at a deep leak detection and collection pit. In addition to the above, there are infiltration bore wells/infiltration galleries provided around the SFSF building to maintain the ground water below the pool raft. In the infiltration system, four deep wells are connected to HDPE slotted pipes covered by layers of gravels and soil around the pool. The ground water from the surrounding travels into these pipes and gets collected in these four wells where submersible pumps take care*

of pumping the water to storm water drains automatically. This system is quite useful in keeping the ground water table below the raft of pool in all seasons [2].

Fuel bundles are handled underwater in the fuel pool in following ways:

- *Handling of trays from shipping cask to storage locations;*
- *Handling of tray from one location to another designated location;*
- *Loading of fuel bundles from tray loading platform into charging cask*"

[3].

All the storage locations of the trays in the fuel pool and loading platform are at fixed co-ordinates due to locators provided on the floor for trays. This aspect is also useful for automation of pool bridge for all these operations. The pool bridge has been provided with rack and pinion type positive drive arrangement for long travel and cross travel motions for accurate positioning. The trays are located at the designated position with the help of guiding locators [4].

The stacks of spent fuel trays have been qualified to be stable under design basis earthquake by both analytical method and as well by testing on a shake table using the site specific ground response spectra [5].

2.2. Direct fuel transfer system

"*An innovative concept of direct fuel transfer of spent fuel bundles from pool to hot-cell has been utilized which eliminates the use of heavy charging casks. This gives an added approach towards total automation in fuel handling at recycle plants.*

Direct fuel transfer system is required for remote transfer of the spent fuel bundles from fuel pool of fuel handling area (FHA) to hot cell in an automated way, eliminating manual operation, saving man hours, eliminating dependency on skilled man power and minimizing man·rem consumption. This will be achieved by connecting the fuel pool to hot cell through an underground tunnel and tray transfer trolley running into it from fuel pool to hot cell and vice versa. The system utilizes electro-mechanical wire rope driven trolley for transferring of fuel bundles from storage pool to the cell, motorized x-table for movement of bundles from trolley position to the shear position, Pushing arrangement for transfer of bundles (from trolley to the x-table and later from x-table to the shearing machine) and various in-built interlocks/safety features" [3].

2.3. Fuel feeding system

Transfer of spent fuel bundles from cask to hot cell for shearing the fuel bundles is a time consuming & labour intensive operation involving man·rem consumption also. An automated charging facility (ACF) was developed for charging of spent fuel bundles. It has helped in reducing cycle time of charging, reducing manpower requirement & reduction in radiation exposure to operating staff. Recently an indexing cask also has been developed where in 20 fuel bundles (5 × 4) will be loaded in a magazine and pushed and indexed four times thus reducing the number of trips to the fuel pool.

2.4. Fuel shearing system

Progressive chopping concept was used in earlier plant wherein fuel bundle was pushed progressively & cut by shearing machine. This process was time consuming & there were chances of fuel cut pieces getting pinched at the ends. A gang chopper has been developed & deployed at Tarapur. It cuts spent fuel in multiple segments in a single stroke without any pinching of the cut pieces. It is modular in design & can be taken out for maintenance without requiring any cell entry. Chopping time is reduced by a factor of ten & uniform cutting of fuel bundle has helped in achieving better plant performance.

2.5. Hull tilting

From a manual tilting system using a cell crane at older plants to a PLC controlled mechanized system at new facility has solved many operational and safety related problems. The operator related incidents are avoided while tilting hull pieces into the drum and dependence on cell crane has been eliminated.

2.6. Hull monitoring

The hull monitoring system for measurement of fissile material in the hull waste in a outgoing drum has been developed and installed. The technique is based on detection and counting of passive neutrons.

2.7. Power manipulator

In new recycle facility, the remotely operated 100 kg power manipulator has solved many a operation and maintenance problems inside the dissolver cell with its unlimited reach within the hot cell e.g. hooking of equipment lid, removal of electrical/pneumatic connectors and replacement of filters.

2.8. Improvements in material of construction of equipment and design

The nitric acid poses extreme hazard to the life of piping and equipment in a recycle plant and generally AISI 304L material is susceptible to IGC mode of corrosion which reduces the life of the plant. a low carbon, low impurity improved variety of AISI 304L called 304 L , NAG grade material which has very low rate of inter granular corrosion i.e. less than 8–10 mpy[1], has been used for equipment and piping.The EOT cranes inside the hot cell do require periodic maintenance without which the whole facility may come to standstill. Hence a roll out feature for cranes has been provided so that crane is accessible for maintenance in non-radioactive area.The Joule heated ceramic melters for verification have been designed for higher life and are provided such that a failed melter can be rolled out of the hot-cell and new one can be rolled in for continued operation of the facility

2.9. Waste volume reduction

Gradually waste volume generation per ton of fuel has been reduced by suitable segregation of different process streams, by reducing the salt addition to certain process streams, by increasing the factor of evaporation in evaporators etc. In the upcoming plants, volume of Intermediate Level Waste and high level waste generation per ton of fuel will be egligible as

[1] 0.232 – 0.254 mm/year

the Cs , Sr and minor actinides are being separated from high level waste in a three stage process using specially developed solvents. The Cs recovered would be put to use by production of Cs pencil source and Cs source will be useful for industrial radiography replacing short life source ^{60}Co.

The impacts of partitioning High Level Waste are:

- Reduction in radio-toxicity;
- Minor actinides to be transmuted in fast reactor or ADSS;
- Reduction in requirement of long term storage of waste;
- Recovery of useful fission products for societal application.

The Fig. 1 shows the effect of separation of minor actinides and fission products on release of radiation. The demonstration facilities for separation of minor actinides and production of Cs-pencil source have been commissioned and have given encouraging results to take forward these technologies on commercial scale plants.

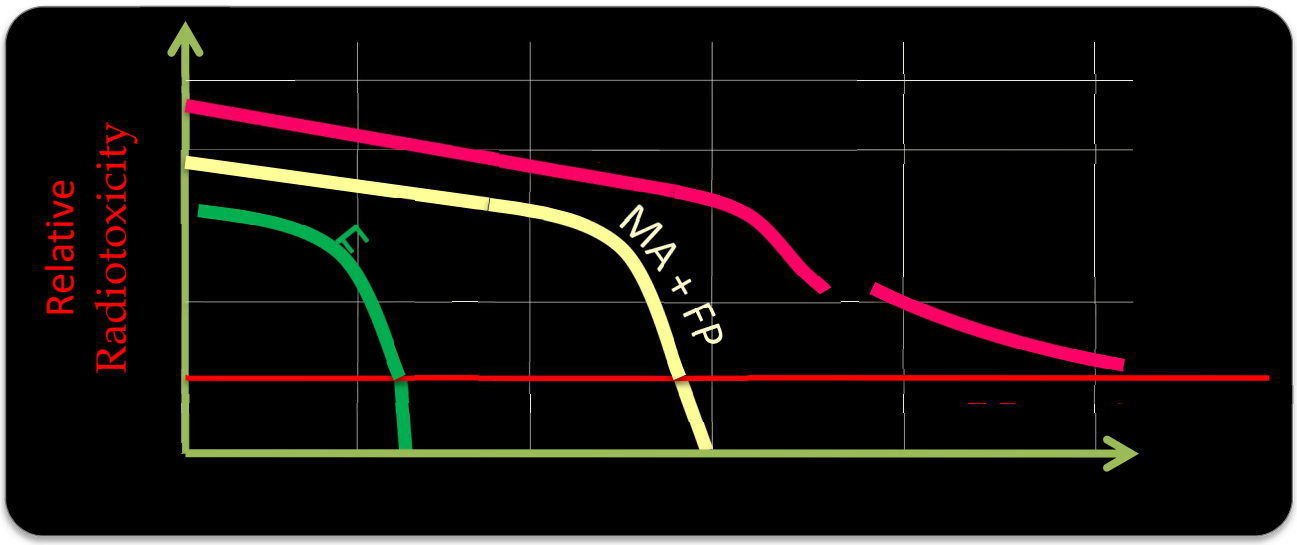

FIG. 1 Effect of partitioning of HLW.

The solid waste volume from zircalloy hulls is also planned to reduce by using a compaction process and packing these waste compacted pucks into a standard canister before transfer for long term storage.

2.10. Control and instrumentations systems

In the older plants, pneumatic instruments were used in field and in control room. Operations of pumps & valves were carried out manually from access galleries.

Gradually electronic instruments were introduced and control panel based operations were planned. Earlier instrumentation lines in out cell areas were of Dacron material and these were cracking due to heat & corrosive environment. Hence these Dacron tubes were replaced with SS tubes. This has resulted in lesser maintenance & uninterrupted process parameter measurements. Pen & paper based recorders were replaced with Smart digital recorders thus avoiding frequent changing of pen & paper.

The latest plants have PC based SCADA (supervisory control & data acquisition system) with standby server and PLCs for operation with standby. These plants also use digital instruments. An emergency control room having the display of important parameters & having the controls to shutdown plant in case of non-availability of main control room has been provided. The emergency control room is located in a seismically qualified building

3. REVIEW OF FACILITIES UNDER EXTREME EXTERNAL EVENTS

The nuclear recycle facilities at Tarapur and Kalpakkam including fuel storage facilities were reviewed by country's regulatory authorities for external events similar to Fukushima and leading to flooding, power unavailability, etc. [6]. The facilities were found safe under the design basis earthquake / and design basis maximum flood level was much below the finished floor level. There was adequate redundancy provided for power or water availability for cooling of fuel pools or and waste storage tanks. Some of the measures such as an additional water hook up line have been provided for filling of pool by fire tenders/water tankers.

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STUDY ON THE MULTILATERAL MANAGEMENT OF SPENT FUEL ACCORDING TO KOREA'S POWER SUPPLY PLAN

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Abstract

Through the 6th power supply plan at 2013, Korean government announced nuclear power plants (NPP) reaches 34 units at 2024. In 2014, it announced through the 2nd energy basic plan that the 43 GW of the power supply will be from the NPP. Just like this, although the operating NPP unit is increasing, there isn't a specific plan on the management or disposal of the spent fuel produced by the nuclear power generation. Recently in November, public engagement commission on spent nuclear fuel management suggested the establishment of permanent disposal facility around 2055. According to the technological assessment, it is predicted that the spent fuel storage amount at NPP site will be started to be saturated from 2025. Normally, it suggests temporal expansion of storage facility in the plant or operation intermediary storage facility as the alternative before the permanent disposal. However, there are more diverse ideas such as overseas reprocessing or domestic reprocessing. This study analysed the pros and cons and the possibility to be realized of each alternatives through multilateral analysis. The interrelation between the spent fuel production amount and management plans among the time difference and system dynamics methodology for the analysis of the pros and cons and feasibility study of each interrelation was used for the management plan assessment. For the interpretation of the system dynamics methodology the analysis tool was made using the Goldsim program to calculate the yearly spent fuel production amount and disposal of each management plans and storage and disposable amount. The main assumption of the assessment is as following: Operating nuclear power plant units: 34 units (light water reactor 30 units, heavy water reactor 4 units); Calculation of the spent fuel production amount: The actual and forecast yearly production amount of each nuclear power plant unit; The cooling time in NPP: 6~7 years; Cooling time before permanent disposal: 40 years;Yearly reprocessing amount: overseas- 360 ton/year, domestic-max. 600 ton/year; Yearly permanent disposal amount: 660 ton/year. In case of the overseas reprocessing, as the amount of the spent fuel is more than the reprocessing amount it is not easy to be the alternative. Also in case of the domestic reprocessing, the time that the technology is completely developed and the amount to be reprocessed can be a problem. Therefore, the temporal expansion of storage facility in the plant or operation intermediary storage facility is thought to be necessary.

Key words: Nuclear spent fuel, management, storage, disposal.

1. INTRODUCTION

The Korean government has two national energy plans, one is the 'Energy basic plan' and other is the 'Electric power supply basic plan'. The energy basic plan is a more encompasssing plan than the electric power supply basic plan. All kinds of energy sources such as oil, coal, gas and electric power, etc. are analysed in the energy basic plan. The electric power supply basic plan is a large sub plan of the energy basic plan, and will be developed into more a detailed plan.

At present in 2014, a second energy basic plan and the sixth electric power supply basic plan have been announced. They have predicted economic growth rate, population (household), long term oil price, and industrial structure in the future. These four indicators are used to predict the energy and electric power consumption in the future.

Nuclear energy policies are established and suggested in both plans. Thirty four nuclear power plants will be required to be in operation by 2024; based on the sixth electric power supply basic plan. About 34 GW electric power will be required to be generated by nuclear power plant by 2035; according to the second energy basic plan.

Increasing the number of nuclear power plants with time will generate alot of spent nuclear fuel. A spent fuel management plan for safety and other important issues will need to be developed.

There is no clear plan on future spent fuel management, but many management scenarios have been considered. This study is one of them, and suggests a spent fuel management method.

This study has developed and modified a back end of the fuel cycle management program utilizing the Goldsim program for considering and evaluating various parameters. Using this program, the study has evaluated the saturation year by storage method and intermediate storage plan until the permanent disposal facility is operational.

2. ENERGY PLANS OF KOREA GOVERNMENT

2.1. Sixth electric power supply basic plan (2013)

The sixth electric power supply basic plan published 2013 is based on the first energy basic plan in 2008. It shows the analysis and predicted electric power consumption between 2013–2027.

The publication forecasts electric power consumption, supply and construction plans for new power plants.

The plan is to construct new power plants, including nuclear power plants, between 2013 and 2027. Fourteen new nuclear power plants up to 2024 were planned in the fifth electric power supply basic plan. As a result of the accident at Fukushima NPP in Japan, the Korea government delayed a decision on building further new nuclear power plants (NPPs). Recently, a decision to construct eleven NPP units (23 NPP units currently in operation) by 2024 has been made, but there is no clear plan after 2024.

A total of 34 NPP units are planned to be in operation by 2024.

2.2. Second energy basic plan (2014)

This publication has five main chapters: Background; Basic direction; Procedure, Prediction with goal; Long term policies. The predictions and developed plan is until 2035. The scenario analysis, used the following assumptions:

- Economic growth rate — will grow by 2.80% per year from 2011 to 2035;
- Population(household) — population will grow by 0.17% per year until 2030 and then fall, the number of household will grow by 0.96%/year (with an increasing number of one person households);
- Oil price — will grow by 1.2% per year to reach 140 US dollars/barrel by 2035;
- Industry structure — the leading industries will be the assembly and metal.

Forecast and resultant policy is:

- Total energy from all of energy resources will need to be increased;
- The proportion of nuclear power energy is estimated at 29% (Fig. 1).

The predicted NPP installed capacity needed would be 43 GW(e) which would require an additional 7 GW(e) of new nuclear power to be constructed between 2024 and 2035.

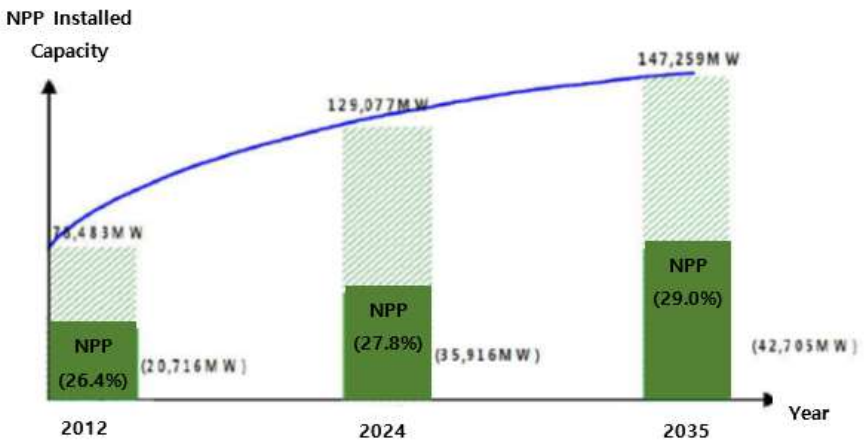

FIG. 1. Forecast of nuclear power plant installed capacity.

3. EVALUATION OF SPENT FUEL MANAGEMENT METHODS

This study has developed and modified the back end fuel cycle management program utilizing the Goldsim program for considering and evaluating various parameters. Using this program, the study has determined the saturation year by storage method and intermediate storage plan until the permanent disposal facility is operational.

The developed program used in the evaluation is configured as shown in Fig. 2.

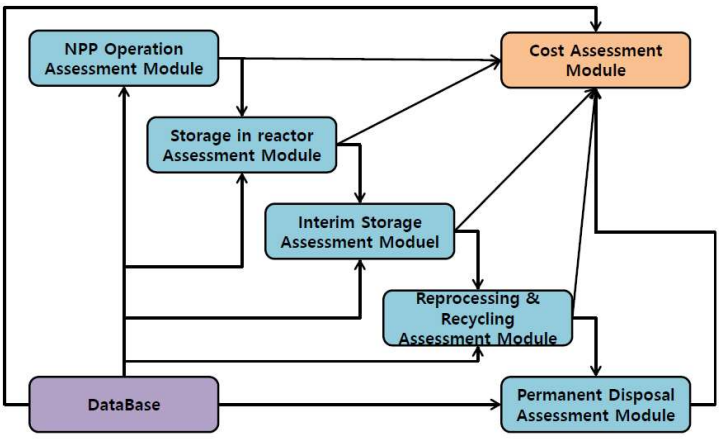

FIG. 2. Assessment program model structure.

In the study, the following parameters were evaluated:

- Amount of spent nuclear fuel generated;
- Saturation year by spent nuclear fuel storage capacity;

- Impact on interim storage facility capacity by spent nuclear fuel management method adopted.

According to the current electricity demand and supply plan, 34 nuclear power plants need to be operational by 2024. The recommended permanent disposal facility is to start operation in 2055. All assessments are based on PWR spent fuel and exclude CANDU fuel; because CANDU spent fuel has not many management options.

3.1. Amount of spent fuel generated

Out of 34 nuclear power plants determined to be constructed and operated, 30 plants adopt light water reactors (PWRs) and 4 plants adopt heavy water reactors (CANDU). Considering that Kori plant number 8, the last nuclear power plant to be constructed, is expected to stop operation in 2085 (no extended designed lifetime is reflected), no more spent nuclear fuel will be generated after 2085. Assuming these conditions, the total amount of spent nuclear fuel generated is shown in Fig. 3.

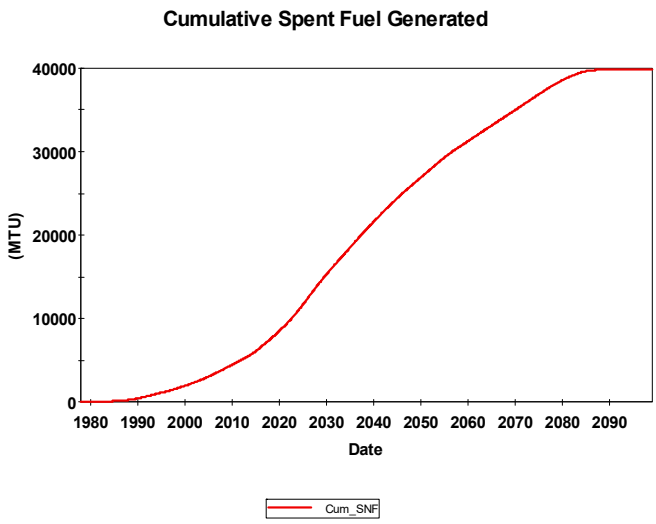

FIG. 3. Cumulative spent fuel of PWRs.

3.2. Saturation year of spent fuel storage in reactor

Each nuclear plant has its own spent fuel pool within the site, and all the spent fuel generated so far is stored in these pools. Because the at station pool storage capacity is limited, the full capacity will be reached if the amount of the spent nuclear fuel grows. If the saturation of the storage occurs before operation of the permanent disposal facility, an additional management facility is required between the saturation point and the operation of the permanent disposal facility. Therefore, it is important to estimate the saturation point of each storage facility.

It is possible to maximize the storage period if you can utilize the at station storage facilities where ther is spare capacity by transporting spent fuel between generators or between sites. Mixed use of at station storage facilities, however, requires consideration of various aspects because there is a technical restriction according to the specifications of nuclear fuel assembly. Therefore, the realistic utilization methods of at station storage facilities are as follows:

- Storing in each nuclear plant unit;
- Storing in each NPP site;
- Storing in the adjacent NPP sites;
- Storing in integrated NPP sites.

Storing in each NPP unit is excluded from evaluation because there already has been transportation of spent fuel between units on the same site, and it is not realistic to store spent fuel in each plant.

The nuclear plants which are under operation or planned are roughly distributed between four regions (Uljin, Wolseong, Kori and Yeonggwang), and NPPs are evenly divided in the same region. Therefore, for evaluating storing in each site, it is assumed that no spent fuel can be transported between sites even in the same region. For storing in the adjacent site, spent fuel can be transported between NPP sites within the same region. For storing in the integrated site, spent fuel can be transported between NPPs sites which are relatively near to one another.

The results of this analysis are shown in Tables 1–3.

TABLE 1. SATURATION YEAR OF STORING IN EACH SITE OF PWRs

Section		Capacity after expansion (MTU)	Saturation year
Kori	Kori	2253	2019
	New-Kori	4789	2036
Hanbit (Younggwang)		3786	2029
Hanul (Uljin)	Hanul	2961	2025
	New-Hanul	2485	2037
Wolseong	New-Wolseong	1068	2033

TABLE 2. SATURATION YEAR OF STORING IN THE ADJACENT SITE OF PWRs

Section		Capacity after expansion (MTU)	Saturation year
Kori	Kori + New-Kori	7042	2034
Hanbit (Younggwang)		3786	2029
Hanul (Uljin)	Hanul + New-Hanul	5446	2032
Wolseong	New-Wolseong	1068	2034

TABLE 3. SATURATION YEAR OF STORING THE INTEGRATED SITE OF PWRS

Section		Capacity after expansion (MTU)	Saturation year
Kori	Kori + New-Kori + New-Wolseong	8110	2034
Hanbit (Younggwang)		3786	2029
Hanul (Uljin)	Hanul + New-Hanul	5446	2032

3.3. Estimation of the scale of interim storage before permanent disposal

According to the above evaluation of saturation point, the at station pool of each nuclear power plant is saturated before operation of the permanent disposal facility; currently recommended to be 2055. Therefore, it is required to build and manage additional storage facility to be operated from the saturation point until operation the permanent disposal facility. Also, reprocessing in overseas countries or domestic recycling can be considered.

Fig. 4 illustrates the comprehensive management plan and Table 4 the scenarios analysed.

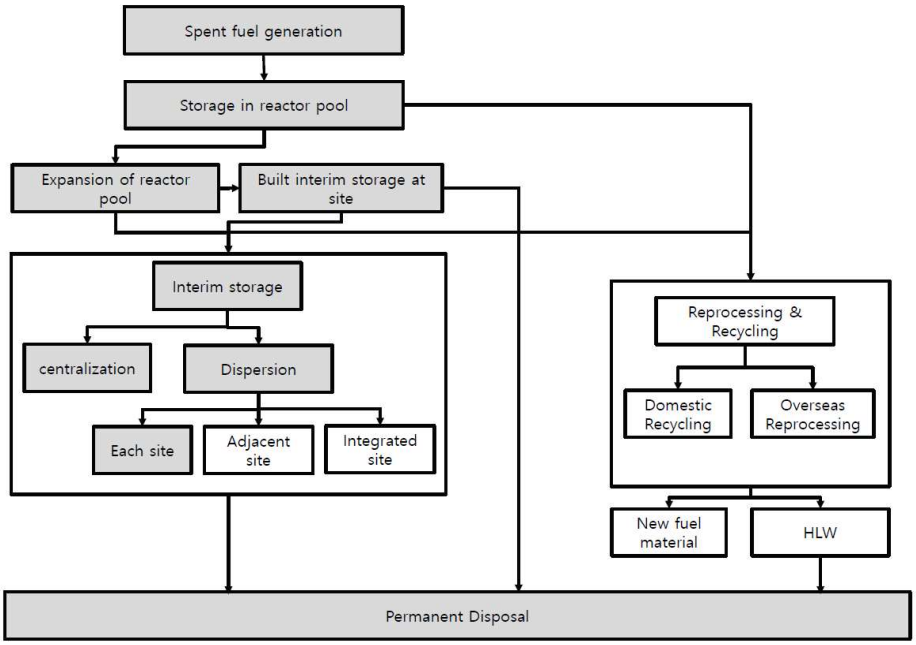

FIG. 4. Spent fuel management plan (grey colours are for CANDU; all processes are for PWRs).

TABLE 4. ASSESMENT SCENARIOS OF SPENT FUEL MANAGEMENT

Section	Management method		Outline of each method
A-1	Temporary facilities		Building of temporary facility at each site
B-1	Interim storage	Centralization	Centralized management
B-2		Dispersion	Independent each site from NPPs, but nearby
B-3			Adjacent site
B-4			Integrated site
C-1	Reprocessing & recycling		Overseas reprocessing
C-2			Domestic recycling
C-3			Link between reprocessing and recycling

The size of the facility required in the interim stage between the saturation of at station storage pool and the permanent disposal facility may be calculated based on the maximum amount of spent fuel generated from the saturation point. Therefore, the size of the facility is calculated for temporary storage, interim storage, reprocessing and reuse based on the management plan suggested in the above figure.

3.3.1. Building of temporary facilities

In this plan, a temporary storage facility is built at each nuclear plant site to store and manage spent fuel. It is the same as building an interim storage facility in the adjacent site except that the facility is built in the nuclear plant site. Because the temporary storage facility is built in the nuclear plant site, various requirements, such as setting of control area, are different from those of interim storage.

The maximum spent fuel storage capacity for each temporary storage facility is given in Table 5.

TABLE 5. MAXIMUM CAPACITY OF TEMPORARY STORAGE

Section		Storage capacity (MTU)
Kori	Kori	435.2
	New-Kori	8489
Hanbit		1154
Hanul	Hanul	1689
	New-Hanul	4897
Wolseong	New-Wolseong	1191

3.3.2. Centralized management

For the centralized management plan, a single site is selected in the Korean territory to manage spent fuel from all the nuclear plants.

In this case, the maximum size of the interim facility must be 17 389 MTU.

3.3.3. Distributed interim storage in independent site of each nuclear plant

In this plan the facility requirements are similar to those for the temporary facility plan except that the facility is built in a separate site adjacent to the nuclear plant.

Therefore, the size of facility is same as that of the temporary storage facility.

3.3.4. Distributed interim storage in the adjacent site

In the adjacent site plan, control areas of the nuclear plant sites in the same region are integrated. Because Kori and New-Kori, Hanul (Uljin) and New-hanul are adjacent to each other, it is easy to transport spent fuel between NPPs. Considering this, a integrated interim storage is built outside of a nuclear plant site. In this case, the maximum capacity of the facility is shown in Table 6.

TABLE 6. MAXIMUM CAPACITY OF INTERIM STORAGE IN THE ADJACENT SITE

Section		Storage capacity (MTU)
Kori	Kori + New-Kori	8905
Hanbit		1154
Hanul	Hanul + New-Hanul	6619
Wolseong	New-Wolseong	1194

3.3.5. Distributed interim storage in the integrated site

Kori and Wolseong are a relatively short distance from one another. Therefore, interim storage can be built in a separate site if there is no problem for transportation. In this case, the size of the facility is given in Table 7.

TABLE 7. MAXIMUM CAPACITY OF INTERIM STORAGE IN THE INTERGRATED SITE

Section		Storage capacity (MTU)
Kori	Kori + New-Kori + New-Wolseong	9928
Hanbit		1154
Hanul	Hanul + New-Hanul	6421

3.3.6. Overseas reprocessing

Spent fuel generated exceeding the at station storage capacity could be sent to an overseas reprocessing facility for reprocessing and management for a certain period. However, due to a problem of overseas transportation and limited annual capacity of the overseas reprocessing facilities, it is not possible to reprocess the entire spent fuel in a short period of time. According to the related research, the amount of spent fuel that can be reprocessed in a year is about 360 MTU. In consideration of this plan, the amount of spent fuel exceeding the storage capacity is decreased considerably, but it is not possible to ultimately process the entire spent fuel. Therefore, it is hard to apply this plan as an alternative management plan, and it is required to establish realistic plan by reviewing various factors if required.

3.3.7. Domestic recycling

Research on recycling of spent nuclear fuel is continued by KAERI Korea Atomic Energy Research Institute (KAERI). This plan is in the research stage, and a considerable time is required until it can be used as an actual management plan for spent fuel. 600 MTU is used as the annual processing capacity of the facility, assuming that domestic recycling is possible from 2028 according to the current long term plan. As in the case of overseas reprocessing, the amount of spent fuel exceeding a certain amount of storage capacity can be decreased, but it is not possible to ultimately process the entire spent fuel.

3.3.8. Link between overseas reprocessing and domestic recycling

It is not possible to use the overseas reprocessing plan and the domestic recycling plan as the ultimate management plan due to the limited annual capacity. When associating the two plans, it is possible to process the spent fuel exceeding the at station storage capacity. However, reprocessing and reuse require consideration of various domestic/overseas conditions. Therefore, more diversified analysis is required for this plan in the future.

4. INTERIM MANAGEMENT OF SPENT FUEL

The saturation year of each at station pool and the size of the facility for each management plan between the saturation point and the permanent disposal were calculated. However, in order to establish a realistic management plan, it is necessary to check if the plans are appropriate in terms of time.

For operation of temporary storage facility and interim storage facility, it is required to complete various preliminary processes, such as securing of land, building of facility and

licensing. Therefore, an analysis was made to check if the conditions for each management plan are fulfilled.

The pre-operation stage of the interim storage facility is roughly divided into securing of land, building of facility and licensing. According to overseas and domestic cases, it takes at least 4 years to secure the land, and 5.5~7.5 years to build the facility. Licensing can be conducted in parallel with securing of land and building of facility. Therefore, it takes at least 10~13 years in the pre-operation preparation stage, which means that preparation for operation must be started at least 10 years before the saturation point.

The Figs 5–7 show the time required in the preparation stage versus the saturation point for each management plan.

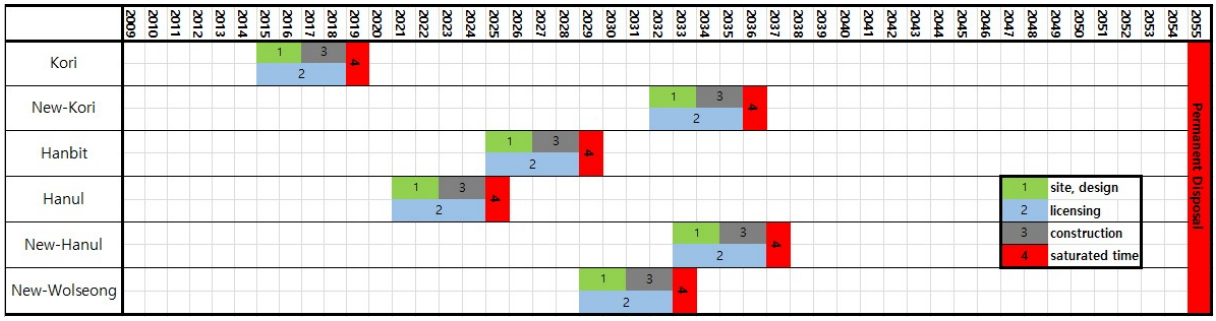

FIG. 5. Temporary storage management plan.

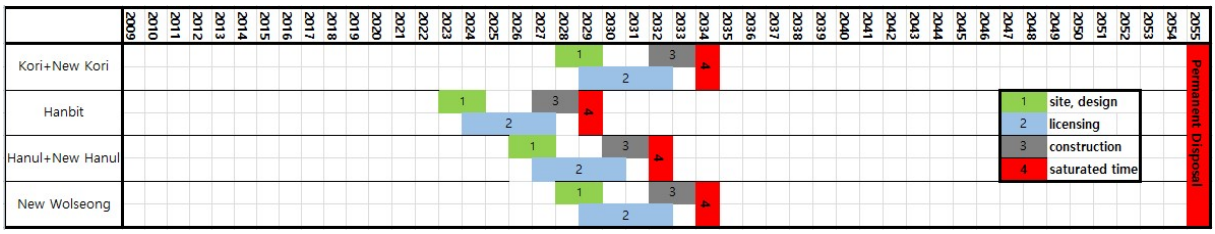

FIG. 6. Management plan of interim storage in independent site of each nuclear plant.

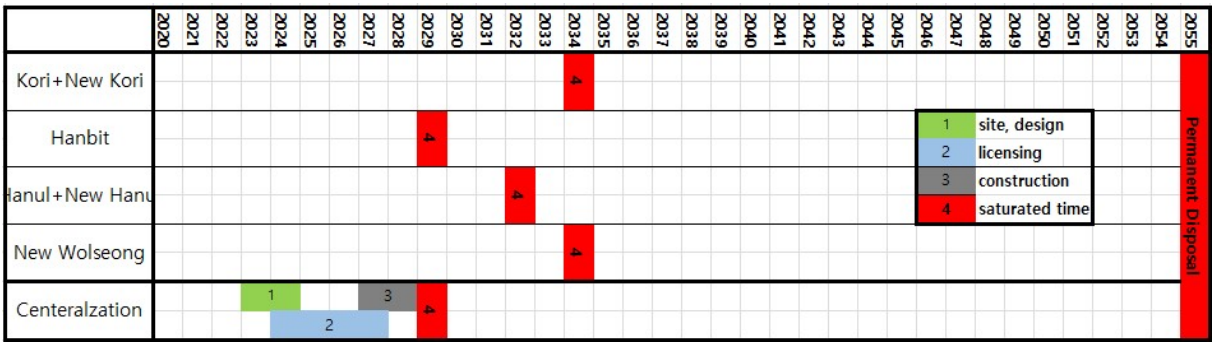

FIG. 3. Management plan of centralized facility.

Interim management by site is not realistic because the saturation point of Kori nuclear plant is 2019, and just 4 years is left as of today.

For the temporary facility plan, because the facility is to be built in the already secured nuclear plant site, it is possible to reduce time to secure the land. Considering the time required for construction and licensing, however, it is required to start preparation in 2015.

For the centralized management plan, it is required to extend the saturation point by transporting part of the spent fuel from Kori to Shingori nuclear plant. In this case, it is required to determine whether to adopt this management plan in not later than 2018.

For management in adjacent site and in integrated site, it is Hanbit nuclear plant that is expected to be saturated first. Based on the case of Hanbit nuclear plant, it is required to establish the management plan in not later than 2018.

Overall, the management plan must be determined in not later than 2018, and survey of land and building of the facility must be started in time to start managing spent fuel at the saturation point.

5. CONCLUSION

This study calculated the amount of spent fuel generated from 34 nuclear plants, reflecting the nuclear plant construction plan suggested by the electricity demand and supply plan in Korea, and calculated the saturation point by comparing with the capacity of the in-plant storage pool. Assuming the operation of the permanent disposal facility in 2055, it analyzed the feasibility of the spent fuel management plan from saturation of the at plant storage to operation of the permanent disposal facility, and suggested the size of the facility. This study also suggested the realistic management plan in consideration of the preparatory period required before operation, and the start point to meet fulfill the requirements.

In order to analyze the spent fuel management plans from various viewpoints, a dynamic analysis system was established by using the system dynamics tool, and evaluation was made for numbers of management scenarios. The program is designed to ultimately calculate the cost of each management plan based on the size of facility of each management plan. In this report, however, the program was used to calculate the size of the facility, but evaluation of cost is to be added later.

Various spent fuel management scenarios have been reviewed in many researches, and most of the realistic plans for review have been suggested. Evaluation of the suggested plans may provide different results depending on assumptions and conditions applied to the scenario. In the static calculation method used so far, it was difficult to analyze complex relations, and evaluation has been performed based on relatively realistic assumptions and conditions. The method suggested in this study uses the same evaluation logic despite verified assumptions and conditions, providing analysis in a short time.

The dynamic analysis program suggested in this study and the result can review more scenarios in the same time period under the same conditions, enabling operators or policy managers to review the suggested scenarios and understand the trend immediately. Due to the characteristics of the dynamic analysis method, it is difficult to show all the advantages in the fixed report frame. Diversification of analysis and review of more scenarios are required to verify the advantages of the program through the outputs, and to support review of more scenarios to determine the spent fuel management plan in the future.

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INVITED SPEECH 02: SPANISH STRATEGY FOR THE MANAGEMENT OF SPENT NUCLEAR FUEL — ATC PROJECT

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Abstract

The radioactive waste management policy is established by the Spanish Government through the Ministry of Industry, Tourism and Commerce. This policy is set at the General Radioactive Waste Plan (GRWP), whose revision currently in force was approved by the Cabinet of Ministries in 2006. A new revision of this document is being discussed for further approval. ENRESA is the Spanish organization in charge of radioactive waste (including spent nuclear fuel) management and nuclear installations decommissioning. The priority goal in spent fuel management is the construction of a Centralized Interim Storage Installation (ATC), whose generic design was approved by the Regulatory Body, Consejo de Seguridad Nuclear, in 2006. This facility is planned for some 6.700 tones of heavy metal, so the total spent nuclear fuel (SNF) generation expected for 40 y of Spanish NPPs operation. The ATC site selection process, based on a volunteer community's process, was launched by the Government in December 2009. The municipality of Villar de Cañas (Cuenca) was selected among other candidates in December 2011. ENRESA is now performing the site characterization and licensing activities in view to get the preliminary and construction permits according to the Spanish regulation (Reglamento sobre Instalaciones Nucleares y Radiactivas–RINR). Meanwhile, extension of the onsite storage capacity was implemented at various NPPs, including past reracking at all sites. In any case, the long-term management of spent fuel is based on direct disposal. Regarding the long term management of the SNF, The GRWP recognises that deep geological disposal is internationally recognised as the preferred solution, while also indicates that effort should be undertaken to assess the potential impact of advanced technologies into the inventories of SNF. In order to provide decision makers with enough information, a number of activities are under way, including basic designs of a geological disposal facility for clay and granite host rocks. Both interim storage and definitive disposal are supported by an ambitious R&D programme, including the commissioning of a Hot Laboratory along with the ATC facility. This paper describes the Spanish spent fuel management system.

1. INTRODUCTION

The Spanish policy on radioactive waste and spent nuclear fuel (SNF) management and on decommissioning of nuclear installations is defined by the government through the Ministry of Industry, Energy and Tourism (MINETUR), and described in the cabinet approved general radioactive waste plan (GRWP). The sixth GRWP was approved in 2006 [1]. A new revision of this important document is in discussion to update the status of radioactive waste management. ENRESA is a state owned company whose responsibility is the radioactive waste management and nuclear installations decommissioning. It is also in charge of managing the Fund, to cover the financial costs of the Spanish system on this subject. The responsibilities of ENRESA are clearly established in [2, 3]. The Consejo de Seguridad Nuclear (CSN), independent from the government, is the sole responsible body for nuclear safety and radiological protection. So, it is also in charge of those issues regarding waste management. Finally, in order to complete the stakeholders on SNF, the NPP operators, that operate on-site SNF storage, whose terms are agreed with ENRESA, and pay the corresponding fees for the fund.

Spain has seven nuclear power reactors in operation in five sites, with a total power of 7.39 GW(e) (approx. 22% of total electricity generation in Spain), and two reactors permanently shutdown: Vandellòs I NPP is in latency period prior to its final dismantling. José Cabrera NPP is in the process of being dismantled. Finally, one other NPP, Sta. Mª de Garoña NPP, is stopped for reasons not related to nuclear safety. The operator is asking for an

extension of operating time to the regulator and the government. The current status is pending a decision of restarting its activity or being permanently shutdown.

Up to the eighties, some fuel from several units was sent abroad for reprocessing. As a result of this, some high and medium long-lived radioactive waste packages should return to Spain in the next years. However, the policy changed to an open cycle in the mid-eighties, so all the spent fuel from light water reactors has been stored on site since that decision. The total amount of spent fuel currently stored is around 4700 tons of uranium (tU), and the total foreseen amount of spent fuel with the hypothesis of the 6th GRWP – i.e. 40 year operation of NPP – reaches 6700 tU of spent fuel.

2. GENERAL ASPECTS OF THE SPANISH SPENT FUEL MANAGEMENT POLICY

According to the 6th GRWP, the priority for spent fuel management is the construction of a centralized storage facility for all the Spanish SNF and other high and medium long lived level waste that should return from old reprocessing contracts or arise from Spanish nuclear plants decommissioning ('special wastes' according to Spanish regulation).

For planning and financing purposes direct disposal of SNF in deep geological formations is considered as the basic solution. However, the door is open should other options arise in the future, such as advanced cycles (including advanced separation and transmutation techniques), that may significantly alter the basic scenarios addressed in the GRWP.

The whole SNF management strategy is supported by a strong R&D programme that deals with interim storage, disposal and advanced cycles projects, to allow decision makers have all the information needed to take decisions in the appropriate moment.

It is recognized that social and technological development during a number of decades will be required for implementing any final option, thus interim storage is seen as a necessity. For efficiency reasons, a centralized solution is preferred instead of keeping the SNF scattered in several sites. As a consequence, the construction of the centralized storage facility (the ATC, for its Spanish acronym: 'Almacén Temporal Centralizado') is the focus of the policy in this field. In addition, for several reasons, some ad-hoc actions have been implemented in various NPP sites to allow for an optimal SNF management until the ATC is commissioned.

3. AD HOC ACTIONS AT NPP SITES

In the 1990's, an extensive reracking programme was implemented in most of the Spanish NPP sites. Currently, all the NPP's pools have already been reracked. Nevertheless, additional activities have been needed to increase the SNF storage capacities in the last years. That is why, at reactor storage facilities were built at the following NPPs: Trillo, José Cabrera and Ascó. Another facility is planned for Sta. Mª de Garoña NPP. All of them host dry casks from different vendors and technologies to complement their pool capacities. The choice of the system was mainly motivated to solve the needs of the NPP in each case. In all cases, ENRESA has supplied the systems (sometimes also licensed by ENRESA, sometimes by the vendor itself) in agreement with the NPP operator. It is operator's responsibility to design, license, build and operate the at-reactor facility as a modification of the NPP. A summary of the current status is shown below:

Trillo NPP has an interim storage facility in operation since 2001. The spent fuel assemblies are stored in dual purpose metal casks ENSA-DPT (Fig. 1), licenced for storage and transportation and supplied by ENSA. The license has recently been reviewed to increase the

maximum burn-up from the previous 45 GWd/tU to 49 GWd/tU, in order to respond to higher burn-up fuel assemblies, foreseen in this reactor. ENSA-DPT cask has a capacity for 21 undamaged fuel assemblies PWR 16 × 16. The facility currently holds 25 casks out of 80, that is the capacity of the facility for 40 y of operation.

Jose Cabrera NPP was shut down in April 2006 and its management (not the ownership) was turned over to ENRESA to perform the decommissioning plan from February 2010. Prior to do so, according to regulation issued by the Spanish Ministry of Industry, the operator had to defuel the nuclear fuel pool, located inside the reactor containment building. ENRESA supplied HI-STORM systems to load the fuel assemblies. The license in Spain is hold by ENRESA. The particularity of this case is that the total inventory of 377 spent fuel assemblies (100.5 tU), including damaged fuel assemblies and non-fuel hardware (control rod assemblies, thimble plug devices, etc.) had to be loaded into the cask. In September 2009, all the inventory was succesfully transferred into 12 HI-STORMs (Fig. 1). Due to the management change to allow for decommissioning activities, ENRESA is currently operating the dry storage facility. Four HI-SAFE systems to store highly activated pieces from reactor internals decommissioning were added to complete the 16 positions of the pad.

Ascò 1 and 2 NPPs expected to complete the capacity of their pools in the fuel buildings by 2013 and 2014 respectively. Even if the ATC Centralized facility was in the process of being launched, a solution had to be implemented for these NPPs. A storage system, similar to the one selected for Jose Cabrera NPP, was chosen to provide complementary room for their pools. Currently, 7 HI-STORMs are stored in the pads of the the facility (Fig. 1).

TRILLO JOSÉ CABRERA ASCÓ

FIG. 1. A general view of several of the at-reactor facilities.

Finally, a similar situation is happening to Sta. Mª de Garoña NPP. In this case, if the NPP had kept on operating, their pool capacity would have been over by 2015. An action plan was launched several years ago and an at reactor dry storage facility has been designed and licensed. In this case, ENSA ENUN-52B was chosen as the cask to be stored there. The casks are being manufactured and the construction of the facility is expected to begin after summer 2015.

4. THE CENTRALIZED STORAGE FACILITY (ATC)

4.1. The Project

The project consists of three main elements:

- The storage facility itself;
- A technological research centre;
- A business park.

This threefold approach will assist local and regional development and will also give an opportunity to enhance country research in the nuclear spent fuel and high-level waste behaviour fields. As a consequence, it will give support to decision makers to take further management steps.

4.2. Status of the project

As explained at the introduction, the 6th GRWP, approved in 2006, established the strategy of spent fuel management in Spain based on a centralized facility. Several steps were taken to launch a site selection process to host that facility, based on the COWAM recommendations [4], providing information to all the stakeholders and asking for their participation in all the process. In 2006, the parliament urged the government to set an inter-ministerial commission to define the criteria and supervise the site selection process on a transparent, democratic and participative basis. From that same year an information campaign directed to all the municipalities in the country was carried out. The Spanish Secretary of State of Energy launched in December 2009 [5] a public call for candidate municipalities to host the centralised interim storage facility for nuclear spent fuel and high level radioactive waste. As a result of this process, 14 municipalities voluteered to host the facility. 5 of them were rejected for administrative reasons and, from the 9 left, one of them was invited to abandon as almost 100% of its land was environmentally protected. The 8 candidates remaining are shown in Fig. 2.

FIG. 2. Eight candidate sites.

Finally, in December 2011, the Cabinet of Ministries decided Villar de Cañas (Cuenca), among the 8 candidates, as the municipality to host the ATC site. This decision was formalized by a resolution of the Secretary of State of Energy in January 2012 [6]. This resolution also stated the project as essential public service and mandated ENRESA to do it.

From that moment, ENRESA has been taken the necessary steps according to the Spanish regulation:

- Characterizing the site to provide the regulator (CSN) and the MINETUR with the documentation needed to issue the preliminary permit according to the RINR [7]. This also can allow for preliminary civil works on the site (not safety related);
- Designing the facility in detail and providing the regulator (CSN) and the MINETUR with the documentation needed to assure the safety of the facility and to issue the construction permit of the facility according to the RINR [7]. The preliminary safety assessment report (PSAR) along with other documents was presented in January 2014 to comply with this;
- Preparing the environmental impact assessment to provide the Ministry of Agriculture and Environment with the documentation to issue the environmental impact statement. It is also needed for the preliminary permit referred to above;
- Working on the land use change from rural land to industrial use, according to local and regional regulation;
- Establishing a preliminary logistics planning;
- Providing information to stakeholders. An ENRESA office with an information center was established in Villar de Cañas from the beginning.

4.3. The ATC storage facility

The ATC Storage facility has been designed to meet the technical criteria of IS-29 [8], that closely follows similar regulation from the US, the 10 CFR 72 [9]. The main safety target is to protect the people (workers and public) and the environment from ionizing radiation. In order to comply with this target, measures shall be taken to:

- Limit and control radiation exposures and radioactive effluent release;
- Limit the probability of events that can make losing control of radioactive materials;
- Mitigate the consequences of such events;
- Minimize the waste generation;
- Specifically, it has been designed to keep the following functions:
  - Maintain sub-criticality;
  - Maintain confinement of radioactive material;
  - Ensure radiation rates and doses for workers and public do not exceed acceptable levels and remain ALARA;
  - Maintain retrievability;
  - Provide for heat removal to meet the above mentioned safety functions.

The selected technology for spent fuel and high level waste storage is based in vaults, were the fuel assemblies are transferred into capsules, then encapsulated and the capsules so welded sealed are finally stored in dry wells, providing double confinement (canisters and wells), providing the best conditions for storage during decades and cooling by passive means; i.e. natural convection. Fig. 3 shows a conceptual design of the facility.

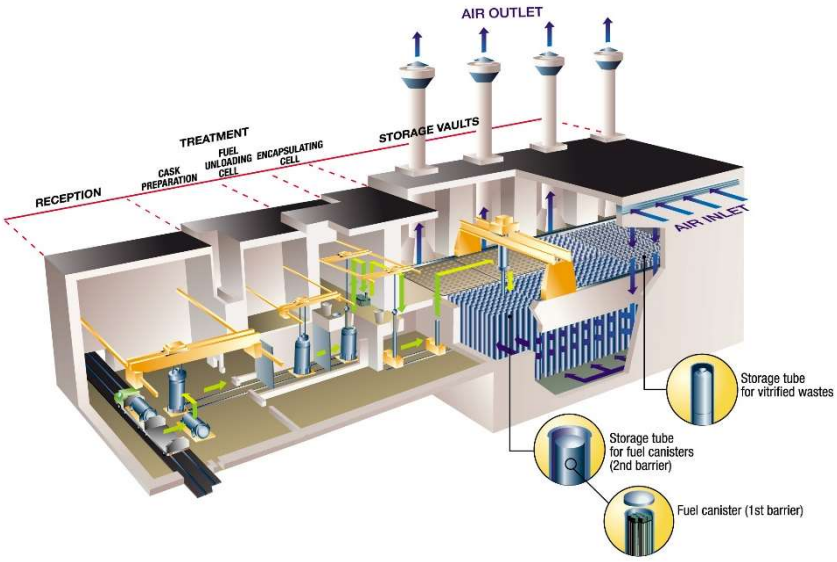

FIG. 3. Conceptual design of the ATC facility.

The facility will also include other storage buildings for "special wastes" and for SNF loaded casks to give flexibility to the ATC storage main process. Fig. 4 shows a layout of the nuclear island currently considered, where the storage buildings are coloured in red, being the rest of installations auxiliary to the whole facility.

FIG. 4. Layout of the nuclear island.

4.4. The technological research centre — the laboratory for spent fuel and radioactive waste

ENRESA R&D previous programmes put much effort in deep disposal issues, organized and justified from performance assessment exercises associated to different conceptual designs in granite, clay and, in a lesser volume, in salt. The R&D programme has evolved following the GRWP trends. The works in support to the deep disposal option has been reduced, although some effort is devoted to maintain teams and competences, but still providing information to decision makers and being aware of the international developments on this issue. Currently, the research work is more focussed in supporting activities for interim storage at an industrial stage, considering prolonged storage timeframes. The technological research centre foreseen as a part of the ATC complex will give momentum to the research work in the spent fuel field for interim storage and deep geological disposal, too.

The technological research centre will consist of four laboratories: three conventional laboratories and one hot laboratory.

- Laboratory of spent fuel and radioactive waste. For its characteristics, its location is expected to be inside the nuclear island of the facility (inside the "double fence"). This lab is designed to do research on fuel rods and other highly activated parts coming from fuel structural assemblies, fuel hardware and other activated parts from the reactor core. It will have concrete cells, shielded cells and glove boxes and equipped with state-of-the-art scientific equipment. As this facility is currently non-existing in Spain, it will serve as an opportunity for Spanish science to work on highly activated materials and develop locally the needed know-how on this subject. The fact of being associated to a storage facility is of particular importance as the main process will also be able to extract fuel rods during the lifetime of the facility to send them to this laboratory for research in support of the Life Management Plan of the whole facility;
- Laboratory of industrial prototyping and robotics, where works in support of worker training and prototype testing can be performed, as most of the main process of the facility is remotely operated;
- Materials laboratory, for testing materials, welding techniques, etc. in relation to equipment expected to be used in the ATC itself, but also for the deep geological repository testing;
- Environmental laboratory, in support of the radiological environmental monitoring programme of the site during exploitation.

5. CHALLENGES AND DISCUSSIONS

The ATC, although based in known technologies, is not free of technical and administrative challenges that must be faced before succeeding in operating the facility.

Firstly, the tight schedule for designing, licensing, building and commissioning the facility imposes stresses to the project. It is proposed to build and operate the loaded cask facility prior to the main buildings to provide flexibility to the operation of the facility.

Moreover, current regulation itself is not clear enough in some cases. Most of the regulation for spent fuel storage systems is based on NPPs and is not directly applicable, as a whole, to essentially passive systems as the ATC. In addition to it, the design has to consider the application of the updated regulation due to 9/11 attacks and post Fukushima.

Secondly, the site cannot rely on a water current as the rivers nearby are not abundant, so there is a policy of "zero liquid effluent" that imposes specific restrictions in the design.

Concerning transport, it is important to mention the issue on high burnup fuel (>45 GWd/tU, according to regulation) as there is not a generalized way to address it. The regulation itself indicates that it must be solved on a 'case by case' basis.

Finally, the operation of fuel inside dry hot cells in air and the encapsulation of thin shielded capsules imposes several restrictions in the operation: an important work is being agreed with NPP operators to characterize and classify the fuel assemblies to prevent fuel handling issues and the operation of fuel with cladding damage inside the cell, not to have fuel oxidation and spread of contamination inside the cell. Although the HVAC system will be prepared for those issues, the operation of the cell could be compromised unless clean enough. Of particular importance is to characterize, and reduce if feasible, the level of cruds of fuel assemblies to reduce its deposition in hot cell surfaces and, as a consequence, to control the doses on maintenance operations and to improve the equipment conditions. The process, in general terms, is performed fully remotely, with the corresponding difficulties in terms of maintenance, environmental qualification of equipment, etc.

6. CONCLUSION

The operation of the ATC is the main goal of the Spanish spent fuel management strategy that will provide a temporary solution (for decades) to store all the spent fuel from the Spanish power reactors and the high level vitrified waste and special wastes generated in the reprocessing of Spanish fuel abroad or in decommissioning of power reactors. This period is judged sufficient to allow for technical and social development that would make possible the definition of strategies for longer-term management options.

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SPENT NUCLEAR FUEL MANAGEMNT SYSTEM IN THE SYSTEM IN THE RUSSIAN FEDERATION

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Abstract

SNF Management Concept of the State Atomic Energy Corporation ROSATOM is in place in the Russian Federation. The Concept specifies goals and strategic vector of SNF management. The Russian Federation policy in the field of SNF management is based on the principle of SNF procession in order to ensure ecologically acceptable handling of fission products and return of regenerated nuclear materials in the nuclear fuel cycle. Establishment of reliable system of long-term monitored SNF storage, development of SNF processing technologies, balanced commitment of SNF regeneration products in the nuclear fuel cycle, final disposal of radioactive waste from processing are strategic approaches in SNF management.

1. INTRODUCTION

The basis of the policy of the Russian Federation in the field of spent nuclear fuel (SNF) management is the principle of its reprocessing to ensure the environmentally acceptable treatment of fission products and return to the nuclear fuel cycle nuclear recycable materials. The strategic directions in the SNF management are to create a reliable system of long term safe storage, development of innovative reprocessing technologies, a balanced involvement of recyclable materials in nuclear fuel cycle, final isolation radioactive waste.

According to this provision and considering the peculiarities of management approaches for different types of SNF, Fig. 1 demonstrates the general pattern of SNF management in Russia.

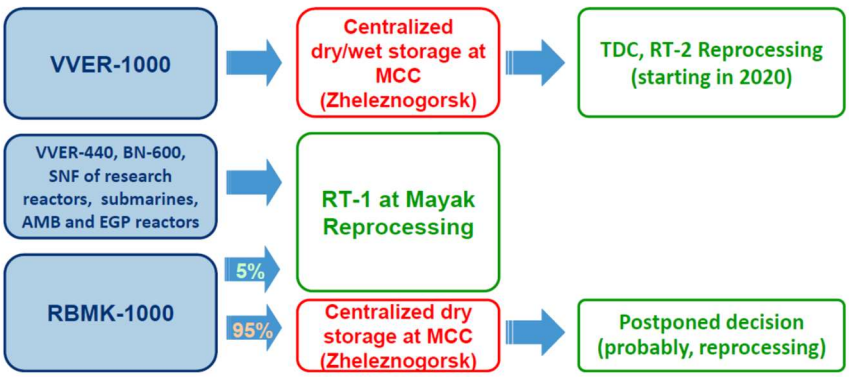

FIG. 1. Technological patterns of SNF management: temporal storage and reprocessing.

2. MINING CHEMICAL COMBINE (MCC-ZHELEZNOGORSK)

SNF from VVERs-1000 is transported for centralized wet (in future dry) storage at the mining and chemical combine, and in the future, after 2020, it will be reprocessed at the pilot

demonstration center for SNF reprocessing and later on at the large scale reprocessing plant RT-2. RT-1 plant now is the only commercial SNF reprocessing plant having been operated in Russia since 1977. The designed capacity is 400 tons per year. More than 5500 tons of SNF have been reprocessed. This plant reprocesses spent nuclear fuel from VVER-440, BN-600, research reactors, icebreakers and submarines. The infrastructure for diversifying the type of SNF suitable for reprocessing is in the course of development (damaged spent fuel from RBMKs, SNF from AMB, EGP and all the types of research reactors). Reprocessing technology is based on the Purex process (recovery of regenerated uranium, plutonium and neptunium as the reprocessing products). The reprocessing process (modified purex) includes dissolution of spent fuel with a subsequent separation of plutonium, uranium and neptunium from fissile products and minor actinides. RT-1 deals with the range of spent fuel compositions producing both highly enriched reprocessed uranium and low enriched reprocessed uranium that are further used to manufacture fresh nuclear fuel. The end products from SNF reprocessing are:

- Melt of uranyl nitrate hexahydrate (enriched up for the RBMK-1000 fuel fabrication). Re-enrichment is achieved by blending the purified uranyl nitrate solutions resulting from reprocessing of different spent fuel types with different uranium enrichment levels (BN, RRs, propulsion reactors fuels and VVER fuel);
- Uranium oxides with ^{235}U enrichment up to 17 %;
- Neptunium dioxide (for production of ^{238}Pu isotope);
- Plutonium dioxide (for fast reactor fuel).

2.1. Waste handling facilities

The first in the world semi industrial facility for partitioning of high level wastes was put in operation at RT-1 in August 1996. SNF reprocessing is accompanied with a production of RadWastes which are subjected to treatment. Current practice for ILW and HLW management from SNF reprocessing at RT-1 plant involves HLW vitrification in EP-500 ceramic melter with design capacity 500 liter of concentrated HLW per hour. Alum phosphate matrix of the radioactive glass is produced using direct evaporation-calcination-vitrification technology. Vitrified wastes are placed in steel canister and are stored in a dry vault type storage facility.

2.2. RT–1 upgrades

At present time, several projects are being implemented at RT-1: Construction of an AMB SNF cutting and packaging complex that will enable further reprocessing of such fuel; Modernization of the receiving unit for leaky RBMK-1000 SNF; Improvement of radioactive waste management systems.

During previous years extensive efforts were carried out to upgrade the RT-1 plant at PA Mayak. The main areas — extending the list of reprocessed fuel types and enhancing environmental safety. Routinely reprocessing of ‘damaged’ SNF from RBMK-1000 and AMB reactor units, as well as of SNF from all types of research reactors will be started in the recent years. Recently, PA Mayak started to reprocess SNF from research centers (Scientific Research Institute of Atomic Reactors, Institute for Physics and Power Engineering named after A. I. Leypunsky, Kurchatov Institute) that had been previously stored at the sites of their generation. It was preceded by major efforts on the development of unloading and transportation technologies.

3. INFRASTRUCTURE AT MCC

The infrastructure of SNF treatment currently being created at the MCC includes the following facilities (as shown on the Roadmap Fig. 2):

- Centralized water cooled (wet) SNF storage facility;
- Centralized air cooled (dry) SNF storage facility;
- Pilot Demonstration Center for innovation based SNF treatment;
- MOX fuel fabrication for fast neutron BN-800 type reactors.

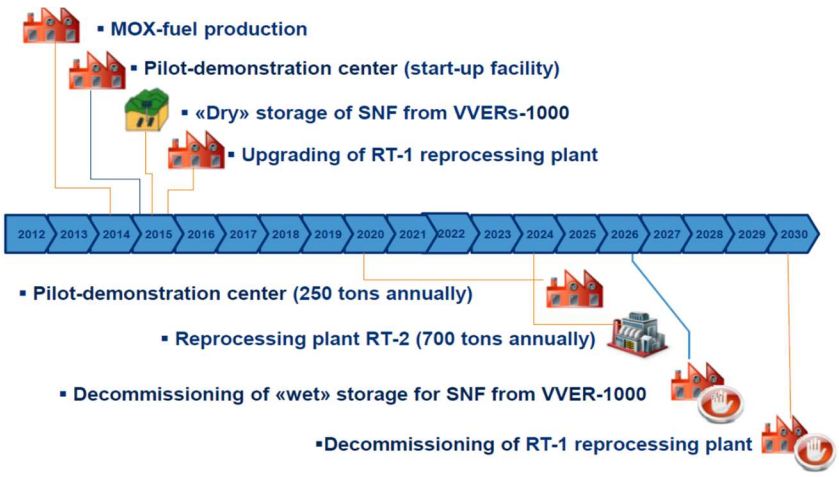

FIG. 2. Roadmap: development of SNF management infrastructure in Russia.

3.1. Centralized wet storage facility

SNF wet centralized Storage Facility (Fig. 3) was put into operation in 1985 as the first stage of the RT-2 plant for reprocessing of SNF from VVER-1000 reactors; it has been successfully operating for almost 30 years taking VVER-1000 SNF for storage. Over the period of 2008–2014 some upgrades were made at the MCC wet storage facility. The renovation was mainly focused on raising the safety levels for storage of SFA and bringing this to the conformity with current norms and regulations in order to ensure further safe operation of the facility and uninterrupted receipt of SNF from NPPs. This work was started three years before the accident at Fukushima Daiichi NPP took place and by the time of that accident the following upgrades had been already finished:

- The seismic stability of the building was considerably improved, the foundation and building structures reinforced and the roof lightened;
- Some cranes were replaced to increase the lifting capacity and to meet the requirements of current regulations;
- The efficiency and reliability of the cooling system was improved.

The effective measures to address beyond-design-basis accidents at the wet storage facility were developed; i.e. water irrigation of SFAs in damaged bays, stable cooling of undamaged basin bays, and reliable operation of regular ventilation. Currently for the purpose of handling the beyond-design-basis accidents concerned with dewatered storage bays, a project of water irrigation is being implemented that is based on the passive principle of protection and the

system of auxiliary emergency tanks which, in cases of a power loss or a seal failure in bays, will ensure stable cooling of the SNF stored.

FIG. 3. Centralized water-cooled storage facility.

3.2. Dry SNF storage facility

Eleven power units of RBMK-1000 type are currently being operated in Russia. Annual SNF generation rate accounts for 400 tones/year. In 2012, dry storage facility at MCC (Fig. 4) was commissioned and this year there was the first shipment of SNF from Leningrad NPP. Purpose built spent fuel assemblies (SFAs) cutting complex, container fleet and conveyors were designed and constructed to support these operations. To date, the process flow has been successfully implemented and the regular SNF shipments have been carried out. In 2014, shipment of SNF from Kursk NPP has been launched under the same process flow. Similar efforts at Smolensk NPP are scheduled to be completed in 2016.

In 2015, construction of the second unit of SNF dry storage facility at MCC is scheduled to be completed. The hot-cell for ahndling RBMK-100 fuel assemblies is shown in Fig. 5.

FIG. 4. General exterior view of the first facility complex for dry SNF storage.

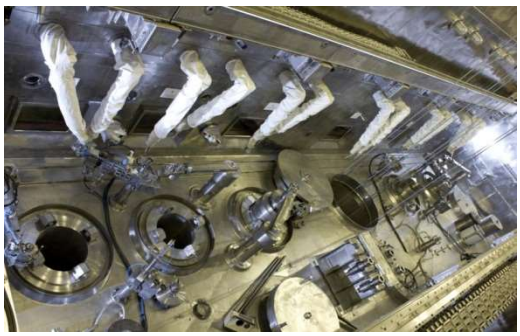

FIG. 5. Hot cell for RBMK – 1000 SNF casing.

The dry storing technology used at the MCC storage facility is based on the principle of passive safety protection and in case of a loss of power supply all conditions of safe SNF storing will be retained thanks to the natural air-cooling by convection. All engineering operations while transferring SNF to storage as well as the storing process itself are fully automated to exclude the influence of human factors on SNF storage safety.

OAO (VNIPIET Main Institute) calculated the ultimate seismic stability for the building constructions and equipment of wet and dry SNF storage facilities. The calculations showed that at maximum seismic load of 7 points on the MSK-64 magnitude scale — the expected maximum for the site of the facility location — the building structures of the wet storage facility maintain integrity up to 8-point load (MSK-64 scale), and the building structures of the dry storage facility maintain integrity up to 9.6 point load according to the same scale.

Part of RBMK SNF is unsuitable for dry storage. Relevant SNF amounts have been identified; the reprocessing technology has been tested at RT-1 PA Mayak.

4. PILOT DEMONSTRATION CENTER

WWER-1000 SNF is routinely shipped off-site to a centralized wet storage facility KhOT-1 at MCC the capacity of which has been increased. Dry storage for VVER-1000 SFAs is planned to be put into operation in 2015.

4.1. Pilot demonstration center for VVER-1000 spent nuclear fuel

For VVER-1000 SNF reprocessing the pilot demonstration center (PDC) in mining and chemical combine (Krasnoyarsk area) was designed, developed and now is under construction. The key goal of the PDC innovation technologies development is to achieve ecologically acceptance and economically efficiency of the reprocessing technologies. The PDC will be constructed in two stages. The first stage namely the research hot cells with capacity about 5 ton of SNF per year and the entire infrastructure required are planned to be commissioned by the end of 2015. The goal is scientific research in the area of new technologies for SNF reprocessing from thermal neutron and fast neutron reactors as well as the nuclear fuel cycle closure.

The PDC will address the following challenges:

- Improvement of new technologies for reprocessing of SNF from fast and thermal neutron reactors;

- Development of HLW partitioning technologies for further radioisotope recovering for commercial purposes and reducing radio toxicity of ultimate disposal waste.

The PDC second stage to the extend to full capacity operation up to 250 tons of SNF per year is planned to be commissioned in 2020 and will represent a complete complex of operations from the SFA destruction to final product (including U-Pu-Np mixed oxide and Urep) along with conditioned radioactive waste vitrified in borosilicate glass for further ultimate disposal.

5. FAST REACTORS

Russia implements a comprehensive programme of transition to a closed fuel cycle, developing two technologies of fast reactors with sodium and heavy metal coolant. In 2014 fast reactor with sodium coolant BN-800 at Beloyarsk NPP and the MOX (uranium–plutonium oxide) fuel fabrication plant at the mining &chemical combine (Krasnoyarsk area, Siberia region) were put into operation.

Simultaneously, the problem of spent fuel accumulation could be solved by utilization of reprocessed uranium and plutonium in thermal reactors. For this purpose the concept of REMIX (Regenerated MIXture of U, Pu oxides) has been developed in the Russian Federation. REMIX fuel is fabricated from an non-separated mixture of uranium and plutonium obtained from SNF reprocessing, that is further adjusted with enriched natural uranium in order to give the required content of fissile nuclides in the recycled fuel. Calculations have shown that the composition of REMIX fuel (1.0-1.5 % Pu + 2.5% ^{235}U) makes possible multiple recycling of the total amount of uranium and plutonium reprocessed from spent nuclear fuel with 100% loading in the WWER-1000 core. State Corporation 'Rosatom' has started a programme for fabrication of three experimental FAs of REMIX fuel with trial loading of REMIX FAs in unit 3 of Balakovo NPP (VVER-1000) planned for summer of 2016.

INVITED SPEECH 03: DEEP GEOLOGICAL DISPOSAL OF SPENT FUEL IN SWEDEN

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Abstract

In March 2011, the Swedish Nuclear Fuel and Waste Management Co. (SKB) applied for licenses to build and operate a final repository for spent nuclear fuel at Forsmark and an encapsulation facility at Oskarshamn, after more than 30 years RD&D. The spent fuel will be encapsulated in corrosion resistant copper canisters and disposed in short boreholes from tunnels at a depth of approximately 500 meter in crystalline rock. The canisters are surrounded by compacted bentonite clay. The primary safety function of the KBS-3 system is total isolation for a very long time. If the isolation is breached retardation of the radionuclides in the canister, buffer and rock provides a secondary safety function. The RD&D has gone through different phases, including concept development, safety assessment development, siting activities, technical development and public information and consultation activities. Important research has been performed on all aspects of relevance for the design of the repository system and for the understanding of the scientific bases for long term safety predictions. This work has been coordinated by SKB, while much of the actual work has been performed by universities, research institutes and consulting organizations in Sweden and abroad. SKB has developed its own research facilities, two underground research laboratories, Stripa and Äspö, an encapsulation laboratory and a bentonite laboratory. In the URLs basic research on rock behaviour and on transport of water and contaminants has been performed. Äspö is also used to test the components of the KBS-3 system, e.g. the bentonite buffer, to perform full scale experiments, as well as to develop and test excavation and handling equipment. Active international cooperation has been important for the success. The laboratories have also played important role in developing public acceptance. Public information, consultation and involvement is important for SKB's activities. As a result there is a very positive public opinion in the two municipalities where we are going to build the repository and the encapsulation facility. The public is also deeply involved in the ongoing licensing process. The licensing process has proven to be protracted. In parallel SKB is doing preparatory work to be able to start construction soon after receiving a licence and develops handling equipment, investigation methods and control systems that will be necessary for the construction and operation of the facility. Much of this work is performed in close cooperation with Posiva in Finland.

SPENT NUCLEAR FUEL MANAGEMENT IN SWITZERLAND: PERSEPECTIVE FOR FINAL DISPOSAL

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Abstract

Spent nuclear fuel management in Switzerland must be considered from the perspective of the development and implementation of a safe deep geological repository for both high level waste and spent fuel assemblies. This paper provides a brief overview of the most relevant issues relating to the management and disposal of the spent fuel arising from the five operational Swiss nuclear power plants. The principal strategy for spent fuel disposal in Switzerland is to enclose it in canisters that are embedded in bentonite, surrounded by the Opalinus Clay host rock. A prerequisite for the licensing of the geological repository is the development of the scientific basis to ensure, with specified safety margins, the handling of the spent fuel and its encapsulation in disposal canisters. In this context, the characteristics of the spent nuclear fuel, such as radionuclide inventory, source term, fission gas release and decay heat, have to be assessed. Furthermore, specific issues such as the handling of spent fuel in the repository surface facility and repackaging of the fuel into the final disposal canisters must be properly addressed. With respect to the handling operations, several safety-relevant issues have to be considered. One of the most important considerations for the near-field of the geological repository is the heat transfer between the canisters and the surrounding bentonite and host rock, which sets a bounding value for heat production per canister. Because of the very high values for burnups and decay heat of the Swiss spent fuels, the loading of the canisters must be optimized according to the spent fuel properties at the time of emplacement. Some relevant aspects of this optimised loading process are discussed in the paper. This paper also examines issues relevant to the long-term safety of the repository; these require comprehensive analysis and development of appropriate programmes and strategies. More specifically, a brief overview of the burnup credit programme for the assessment of criticality safety and the experience from the Full-Scale Canister Emplacement Experiment in the Mont Terri Rock Laboratory for geophysical monitoring of the changes in the material properties of the bentonite backfill with simulated decay heat are presented

1. INTRODUCTION

In Switzerland, five nuclear reactors are currently in operation (Table 1). A potential future operation scenario assumes a 60 year lifetime for all the reactors, except for Mühleberg, which will be shut down in 2019.

TABLE. 1. COMMERCIAL NUCLEAR REACTORS CURRENTLY OPERATING IN SWITZERLAND

Name	Reactor type	Output / MW(e)	Estimated operation[a]
Beznau I	PWR (Westinghouse)	365	1969-2029
Beznau II	PWR (Westinghouse)	365	1971-2031
Mühleberg	BWR (General Electric)	373	1971-2019
Gösgen	PWR (Kraftwerk Union)	1010	1979-2039
Leibstadt	BWR (General Electric)	1220	1984-2044

[a] 60 years operational lifetime with the exception of Mühleberg

A total of approximately 12 300 spent fuel assemblies are expected to be discharged from the operation of the Swiss reactors. Currently, the spent nuclear fuel (SNF) is stored in the NPP fuel pools and in transport/storage casks (TSC) at the interim dry storage sites (ZWILAG and ZWIBEZ). It is worth noting at this stage that Switzerland reprocessed SNF up to 2006. More specifically, 771 t(HM) have been sent for reprocessing in France and the UK and a total of 634 vitrified residue canisters (180-l flasks) will be returned to Switzerland as high level waste (HLW) that is also considered for geological disposal. The HLW is also stored in TSC at ZWILAG. Following the shut-down of the NPPs, all SNF and HLW will be transported in TSC to the repository surface facility, where it will be repackaged into final disposal canisters (FDC). The FDCs then have to be transported to the repository caverns, embedded in bentonite and surrounded by the opalinus clay host rock. NAGRA, the Swiss National Cooperative for the Disposal of Radioactive Waste, plans to submit a general license application for a high level waste site by 2022. The SNF/HLW repository is expected to become operational by 2060 with emplacement until 2075.

Section 2 summaries the NAGRA estimates for SNF from the operation of the Swiss reactors and its corresponding characteristics. A description of the final disposal canister is provided in Section 3 and issues related to the safe handling of SNF and the loading of the canisters are presented in Section 4. In Section 5, some repository safety aspects for SNF are presented. More specifically, considerations on burnup credit and long-term criticality safety are provided and an overview of the full-scale Canister emplacement experiment at the Mont Terri Rock Laboratory is presented.

2. SPENT FUEL CHARACTERISTICSA AND MODEL INVENTORY

SNF that has not been sent for reprocessing is foreseen for geological disposal. The number of spent fuel assemblies (FAs) expected to be disposed of in the repository is given in Table 2, according to NPP and fuel type (UO_2/MOX). The number of FAs is given in the last column, with the assumption of 60 years of NPP operation (47 years operation for Mühleberg).

TABLE. 2. NUMBER OF FAS PER NPP UNTIL END OF LIFE (EOL) AND FUEL TYPE TO BE CONSIDERED FOR THE REPOSITORY

NPP	NPP type	Fuel type	FAs EOL
Beznau I	PWR	UO2	1 681
Beznau II	PWR	MOX	232
Gösgen	PWR	UO2	1 633
Gösgen	PWR	MOX	148
Leibstadt	BWR	UO2	7 422
Mühleberg	BWR	UO2	1 187
Tot UO_2		UO2	11 923
Tot MOX		MOX	380
Tot PWR		–	3 694
Tot BWR		–	8 609
Total all			12 303

For each of the Swiss reactors, different designs of fuel assemblies have been used and, among these, the fuel enrichment and burnup also vary. Furthermore, many of these spent fuel assembly's exhibit high burnup.

The radionuclide inventory of the FAs (needed for long-term safety analysis), the source term (needed for fuel handling issues) and the decay heat (needed for loading optimization) are assessed by dedicated depletion calculations [1].

2.1. Spent fuel radioactivity evolution

A graphical representation of the evolution of the activities of various waste categories for up to 1×10^6 years after the emplacement of waste in the final repository is given in Fig. 1. The time evolution was determined using the model inventory for radioactive materials (MIRAM 14) [2]. MIRAM 14 aims to quantify and comprehensively characterize all Swiss radioactive waste that has already been produced and will arise in the future. The database contains information on key safety relevant data for each waste type, including waste volumes, radionuclide inventory, dose rates, material composition and surface area to mass ratios for metals and alloys. The MIRAM 14 data provides the basis for long term safety analysis as well as the planning of the deep geological repositories and related facilities. The MIRAM waste type categories are: BA (operational waste from the NPPs and other facilities); BE (spent fuel assemblies); RA (reactor waste); SA (decommissioning waste from the NPPs, ZWILAG, research facilities); and WA (waste from reprocessing of spent fuel assemblies). The activity is clearly dominated by the SNF.

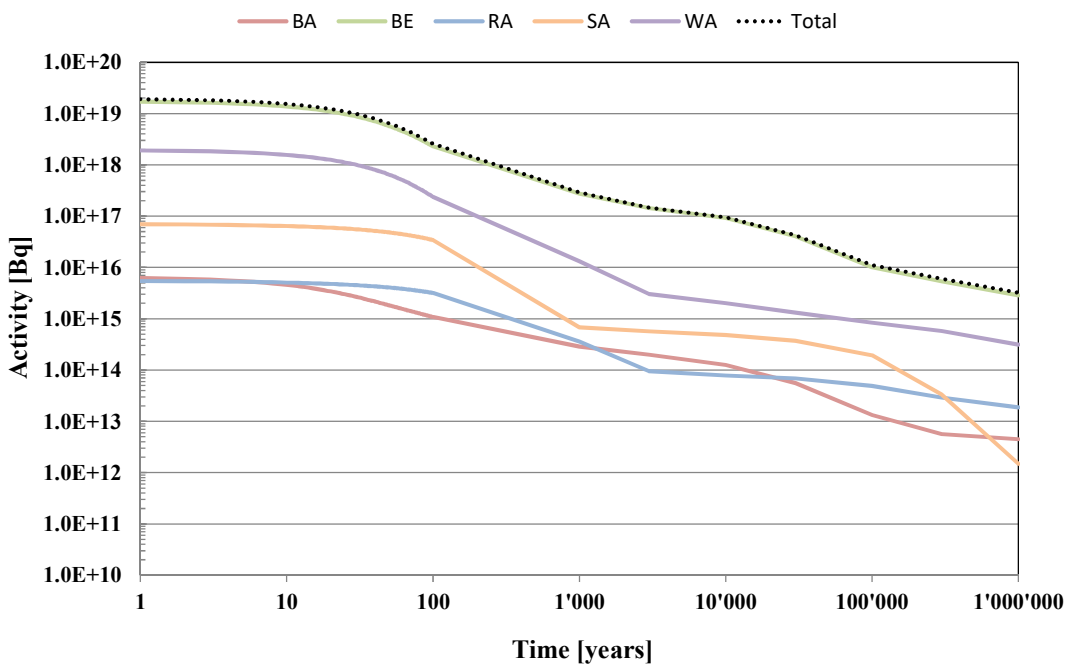

FIG. 1. Activity evolution for main waste categories after emplacement in the geological repository [3].

2.2. Fission gas release

The quantification of fission gas release (FGR) from fuel rods is relevant for short-term as well as long term safety analyses: In the short term, it is of relevance for the safe handling and encapsulation of FAs in the final disposal canisters.

> "*In the context of long-term safety, FGR from spent fuel is known to correlate with the instant release fraction (IRF), which represents the fraction of the inventory of safety relevant radionuclides that may be rapidly released from the fuel at the time of canister breaching and contact with water. Thus, campaigns to calculate the fission gas release for all FAs of Swiss reactors have been carried out in order to obtain reliable average FGR data. Combined with measurements of radionuclide release from spent fuel carried out by NAGRA in collaboration with PSI, SKB and Studsvik for fuel with burnups from 45 to 65 GWd/t(HM) [4], these data have provided the basis for a radionuclide release model for spent fuel [5]. Since FGR is a function of fuel power history and generally increases with burnup, it is of interest to determine the distribution and average FGR at reactor discharge end-of-life (EOL). Such studies have been performed for the Beznau, Gösgen and Leibstadt reactors, based on calculations with FGR codes for all fuel assemblies irradiated in the reactors. The calculations consisted of full core analysis and showed that the release of fission gas from UO_2 fuel is strongly correlated with the linear heat rating, which is dependent on fuel temperature and power density*" [5,6].

Furthermore, the high temperatures near the fuel centre (~1100°C) and high diffusion coefficients in UO_2 lead to increasing fission gas release as burnup progresses. These results, in combination with the burnup distribution of the Swiss FAs, allow the estimation of the FGR for the FAs of each reactor.

As an example, Fig. 2 gives the FGR fractions as a function of burnup and irradiation cycle calculated for the PWR Beznau reactor (KKB). Note that only the FGR values at the end-of-life of the FA in the reactor are relevant for this study. The maximum fraction in KKB is lower than 6%, the maximum being reached in the fourth cycle on a total of five cycles considered [7]. The average values for the 5 NPPs, which are used for the release model of NAGRA, are set as follows: 4.5% for KKL [8] and KKM (BWR fuel); 1.8%/3.4% for UO_2/MOX KKB; and 14%/16% for UO_2/MOX KKG (PWR fuel). In the case of KKG fuel, which is characterized by high power density and high burnup, the FGR reaches a maximum value of 24% for UO_2 and ~30% for MOX.

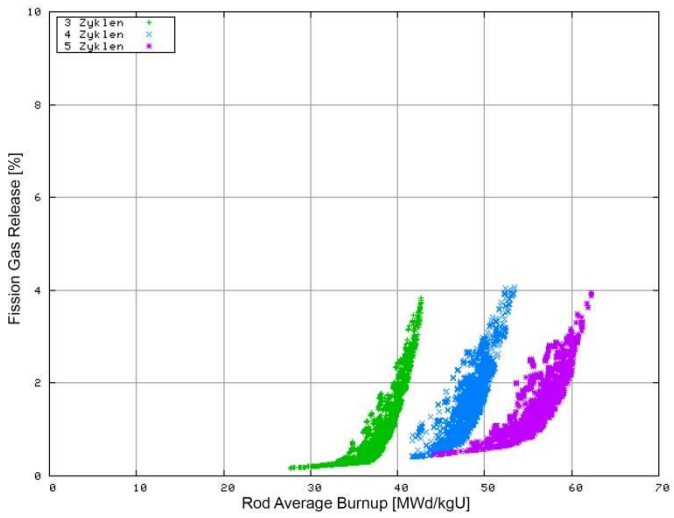

FIG. 2. Fission gas release fraction for KKB UO_2 fuel rods as a function of burnup [7][1].

3. SNF CANISTER FOR GEOLOGICAL DISPOSAL

For the geological disposal of SNF, NAGRA foresees disposal canisters of two different sizes, according to fuel type (PWR and BWR). Furthermore, NAGRA considers three different concepts depending on the fuel loading: 1) UO_2-FA from BWR (BE-S-1), with maximum 9 UO_2-BWR-FA per canister; 2) MOX and UO_2-FA mixed loading from PWR (BE-D-2), with 1 MOX-PWR-FA and maximum 3 UO_2-PWR-FA per canister; 3) UO_2-FA from PWR (BE-D-3), with maximum 4 UO2-FA per canister. Fig. 3 illustrates the current design for canister BE-S-1, which consists of carbon steel. An alternative design of a copper-coated cast iron canister is also under consideration. The canister material must be selected as being proven for welded fabrication and with properties potentially capable of achieving the minimum 1000 year lifetime required by the Swiss Federal Nuclear Safety Inspectorate (ENSI) [9].

FIG. 3. Spent fuel canister design for BWR FAs.

3.1. Residual decay heat evolution in disposal canisters

The variation with time of the average residual power of each of the three FDC types (BE-S-1, BE-D-2 and BE-D-3) has been calculated starting from the reference year 2060 (beginning of emplacement in the repository) up to 107 years. The results are shown in Fig. 4 for all three

[1] In the label on top-left of the figure, 'Zyklen' can be read as cycle.

types and correspond to the average residual power in watts per canister. The residual power evolution with time shown in Fig. 4 is given on a double logarithmic scale and it also shows that the power output conforms, on average to the safety requirement of 1.5 kW/canister (maximum limit) at the time of emplacement, which is relevant for near field safety considerations [10].

3.2. Logistical optimization of fa loading into final disposal canisters

The loading of the canisters must be carried out according to the spent fuel properties at the time of emplacement; in particular the heat limit of 1.5 kW/canister must be complied with. Because of the high value of residual heat coming from the Swiss SNF, some of the disposal canisters may not be completely filled. An optimization technique must be used in order to minimize the number of additional canisters necessary for disposing of all SNF, by using fuel data at the level of individual FA as well as design and logistical data on the surface encapsulation facility (e.g. docking stations available, loading capacity per year). A stochastic optimization procedure based on the simulated annealing algorithm [11] was used by NAGRA (and is currently under revision to achieve with current logistics) to assess the number of canisters required, according to the FA residual heat and by using fuel data at the level of individual FA.

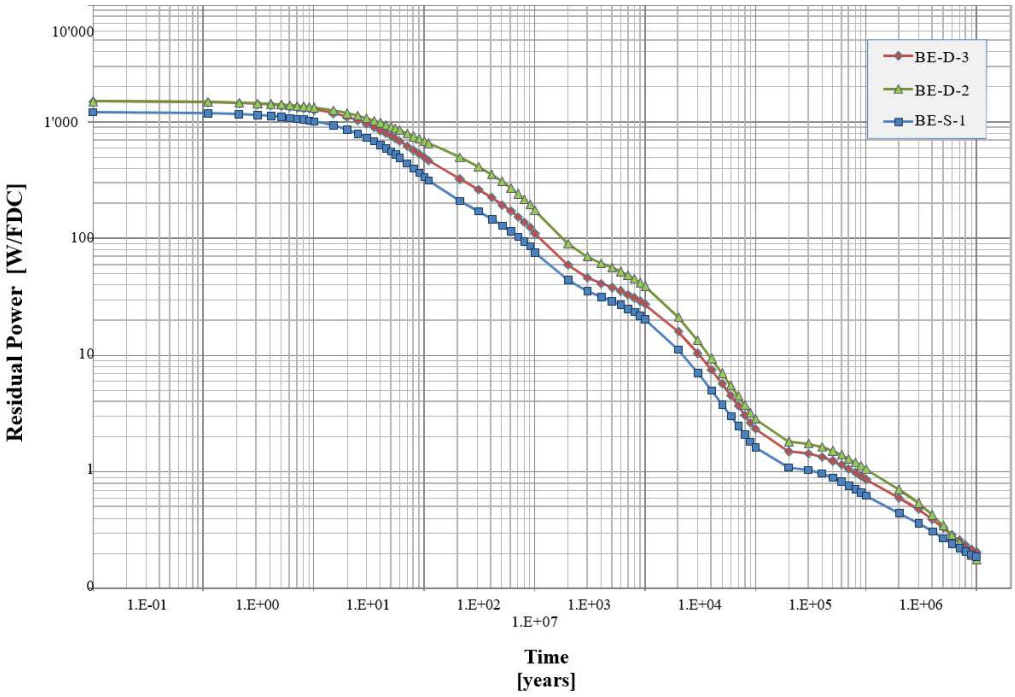

FIG. 4. Residual power evolution per average FDC for the three types (starting point is 2060, beginning of emplacement).

4. SNF HANDLING IN THE SURFACE FACILITY

A prerequisite for the licensing of the geological repository is the development of the scientific basis to ensure, with specified safety margins, the handling of the spent fuel and its encapsulation in final disposal canisters. These operations are planned in the surface encapsulation facility. The SNF unloading/loading operations will be carried out in hot-cells. Four docking stations dedicated to transport casks and four dedicated to disposal canisters are planned to operate simultaneously. A simplified illustration of the FA handling and canister encapsulation is shown in Fig. 5. Safety issues relevant to the opening of the transport/storage

cask, handling of spent fuel in the repository surface facility and repackaging of the fuel into the canisters for geological disposal have to be addressed.

5. SAFETY RELEVANT ISSUES AND R&D ACTIVITY

To ensure safe disposal of the SNF, one requirement is to meet the criticality safety criteria in accordance with the regulatory standard. The effective neutron multiplication factor must not exceed 0.95 in the most reactive conditions (when the canister is filled with water), including different types of uncertainties that define the upper criticality limit (UCL). Studies related to criticality safety assessment for long term criticality scenarios have led to the conclusion that a methodology for criticality safety evaluations must take in account the burnup of the fuel in order to meet the UCL. In this context, NAGRA and LRS/PSI6 are cooperating to develop an advanced methodology based on state-of-the art reactor physics codes including burnup credit aimed at criticality safety evaluation of deep geological repositories. The change in isotopic composition of the fuel with decay time, the variation of burnup along the axial profile and the analysis of uncertainties that contribute to lowering the UCL are currently under investigation.

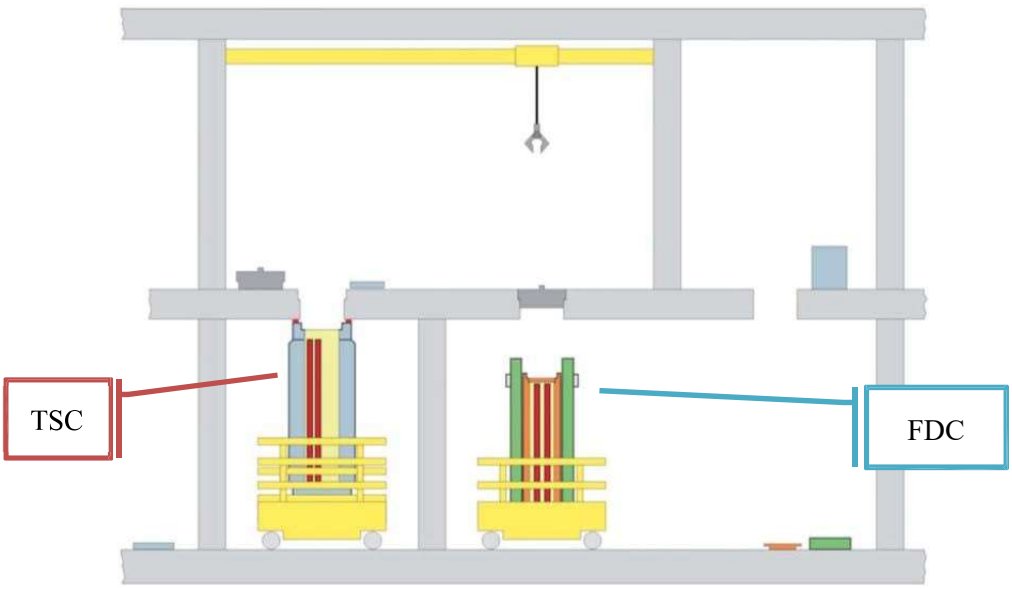

FIG. 5. Simplified illustration of SNF unloading/loading in the NAGRA encapsulation facility; 1 'cask' docking station (left), 1 'canister' docking station (right) [12, 13].

Furthermore, within the framework of NAGRA RD&D activities, it is worth mentioning the Full scale emplacement (FE) experiment at the Mont Terri Rock Laboratory [14]. This is a full scale multiple heater test in a clay rich formation ('opalinus clay'), which simulates the construction, waste emplacement, backfilling and early post closure evolution of a spent fuel repository tunnel as realistically as possible (Fig. 6 gives a three dimensional view of the experiment). The main aim of the experiment is to investigate the induced thermo-hydro-mechanical coupled effects on the host rock at this scale and to verify the engineering feasibility of a repository tunnel and emplacement procedures in underground conditions, as well as the optimization of the bentonite buffer material production.

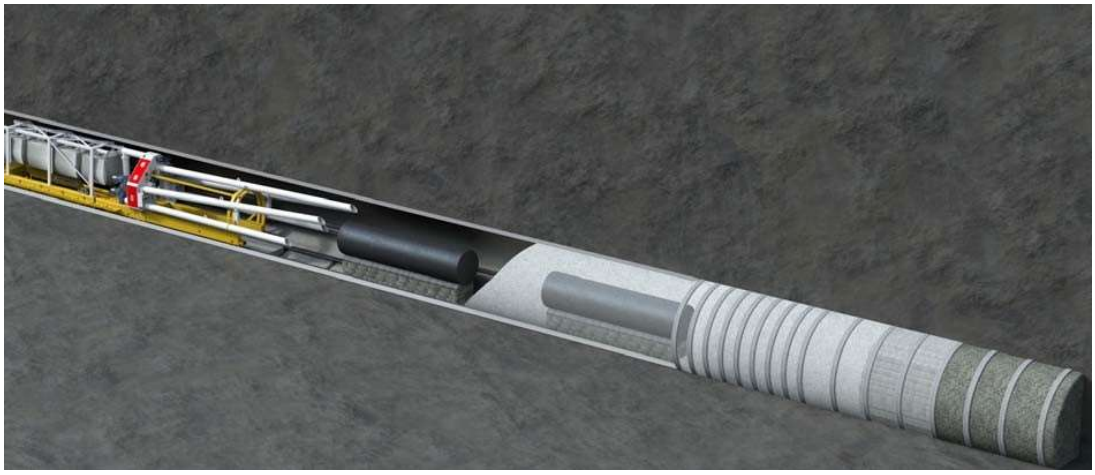

FIG. 6. Three-dimensional view of the FE experiment: repository tunnel, heaters, bentonite buffer and machine used for backfilling.

6. CONCLUSION

The Swiss programme on SNF management is presented in this paper, with the focus on some of the relevant nuclear aspects. A summary of the main points is as follows:

- Both wet and dry interim storage sites are used in Switzerland;
- Deep geological disposal of SNF and high level waste (HLW) is required by law;
- A general license application by NAGRA to construct a geological repository to dispose of all;
- SNF and HLW is expected to be submitted by 2022:
- Studies on SNF characterization are ongoing, focusing on both short-term and long-term safety aspects;
- Investigations on the optimum final disposal canister design are ongoing;
- Ongoing R&D activities for long term and operational safety (see BU credit, FE, etc.).

Based on the studies currently being carried out and on the future developments foreseen for the coming years, a consistent approach to the definitive disposal of SNF will be developed, with the aim of addressing the questions and uncertainties related to this complex topic.

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MANAGING SPENT NUCLEAR FUEL FROM GENERATION TO DISPOSAL: INTEGRATION TO THE BACK END OF THE FUEL CYCLE

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Abstract

The current management system in the United States for commercial spent nuclear fuel does not emphasize integration among storage, transportation, and disposal. The large containers in use for dry-storage remain at high temperatures for decades and, thereby, delay removal from decommissioned reactors. The large containers also have no easy path to disposal unless (1) disposal is delayed (up to 150 years for some geologic media); (2) the contents are repackaged into smaller, cooler packages; or (3) the high temperatures are used as de facto site-selection criteria for a repository. Implementing consolidated interim storage could address many issues that exist because of this lack of integration. A consolidated interim storage facility that includes appropriate capabilities acts as a universal coupler that allows existing disparate parts to integrate as a system. Previous agencies and commissions have noted this theme before as a way to provide flexibility in the waste management system. This rationale is elaborated upon here.

This paper has been removed as large parts of the text have been reproduced elsewhere. The reader is referred to the following reference:

RECHARD, R.P., PRICE, L.L., KALININA, E., BONANO, E.J., "Integrating Management of Spent Nuclear Fuel in the United States by Consolidating Storage", IHLRWM 2015, Charleston, 12–16 April 2015, ANS, La Grange Park (2015) 749–756.

STATUS AND CHALLENGES IN AN INTEGRATED APPROACH
(Session 2)

Chairpersons

B. DROSTE

Germany

M. CHIGUER

France

KEYNOTE SPEECH 02: HOW SPENT NUCLEAR FUEL MANAGEMENT OPTIONS MIGHT AFFECT GEOLOGIC DISPOSAL

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Abstract

The recognition that the endpoint of spent fuel management practices will be deep geologic disposal of radioactive wastes leads to questions about how alternative options for spent fuel management might affect performance of a geologic repository. Do some options for spent fuel management simplify the siting and design of a geologic repository? Do some geologic disposal concepts favour specific spent fuel management practices? Do some spent fuel management options favour specific geologic disposal concepts? Are some waste forms inherently preferable than others for geologic disposal? Given the historical difficulty in many nations associated with siting and licensing geologic repositories for permanent disposal of spent nuclear fuel and high-level radioactive waste, are there activities that the spent fuel management community could or should undertake today to facilitate future disposal operations? A review of existing geologic disposal concepts that have been proposed in multiple nations (e.g., mined repositories in granitic and crystalline rock, mined repositories in salt, mined repositories in clay-rich rocks, and, although less well-developed than mined repository concepts, deep borehole disposal for small-volume wastes) strongly suggests that there are multiple ways to achieve safe long-term isolation of a wide range of spent fuel and high-level waste forms. Long-term repository safety is, in general, not dependent on specific treatments of either spent nuclear fuel or high-level radioactive waste other than packaging. However, multiple aspects of the form in which the waste will be disposed are relevant to repository design and performance, including waste volume, radionuclide content, thermal power, waste package size, and waste form and package lifetime in a range of geologic environments. Consideration of how these factors impact repository performance suggests that choices made now regarding spent nuclear fuel management may affect future flexibility in repository siting and design. Sandia National Laboratories is a multi-programme laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000. This abstract is Sandia publication 2014-20157A.

HISTORY AND CURRENT STATUS OF SPENT FUEL MANAGEMENT AT DUKOVANY NPP

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Abstract

The first Czech Nuclear Power Plant, NPP Dukovany, has almost reached 30 years of safe operation. Significant amount of spent fuel has been produced, though a number of fuel cycle improvements helped to increase fuel performance and decrease its quantity. During the operation years, spent fuel management had to respond to many changes, both political and technical. We had to leave an option to send the spent fuel back to the country of origin and to find our own way how to deal with spent fuel. After a provisional measure of storing the fuel in neighbouring country we adopted spent fuel storage system based on dual purpose metal casks stored in the plant. New challenge is a future strategy — long term storing, final disposal or reprocessing for re-using.

1. INTRODUCTION

NPP Dukovany is a nuclear facility operating 4 PWR units of Russian VVER 440/213 type. The operation of the units started in the period of years 1985-87. This year we will celebrate 30 years of successful, safe operation. The reactors VVER 440 work with 349 relatively small fuel assemblies with a mass 126 kg of uranium, with hexagonal shape and a shroud. Up to now all fuel has been provided by Russian company TVEL.

The original spent fuel management corresponded to the political situation of the time. The original agreement with Soviet Union supposed to ship all spent fuel back to the country of origin. The first step was a transfer of the fuel to a wet storage facility operated yet in the area of NPP Jaslovské Bohunice in Slovakia, at that time part of one federal state Czechoslovakia. Soon afterwards the political situation had changed. Soviet Union ceased and new Russian Federation refused to accept the spent fuel from Czechoslovakia. The next change was division of Czechoslovakia on two countries. The independent Slovakia also refused to receive and store SF from the Czech Republic and asked us to take all stored fuel back.

2. HISTORY OF THE MANAGEMENT OF SF AT DUKOVANY NPP

It was necessary to solve the situation quickly. At first the storage capacity of SF ponds had to be increased since the original project considered only short storage time. A replacement of original racks by compact, high density racks was performed in all storage pools. The capacity of each increased twice, from 350 positions to 699, including 17 cells for hermetical cases. The available cooling time increased twice as well.

The second step was to build the interim SF storage within the Czech Republic. A concept of dry storage was chosen. The particular storage technology was metal dual purpose cask Castor 440/84, provided by GNS, Germany. Dry interim storage facility was built in the area of the plant, however, its capacity was limited on 60 casks by a political decision. At that time it was supposed to build a big central underground storage for fuel not only from Dukovany NPP, but also from Temelin NPP being built at that time. Technology of DPC was suitable for this strategy, despite of higher cost.

Dry interim storage facility started to operate on 5 December 1995 when the first cask was placed there. Then return of the fuel from Slovakia started, where in total 1170 FAs were stored. At first 15 transports by C-30 cask for 30 FAs were performed (the same cask was

used for shipment to Slovakia). The fuel was unloaded from the cask back to the storage pond and subsequently 84 FAs were loaded into the Castor. Later it was managed to perform loading of Castor cask directly in the wet storage in Bohunice NPP. The rest of the fuel was transported by 9 Castors. The last cask was only partially full. It was re-watered, immersed into the pool and filled in to the full capacity.

In the meantime the state concept had been changed again. The idea of the central underground storage was abandoned and it was decided to store the spent fuel solely in the area of the NPPs. The limiting government decision was revoked. Nevertheless it was clear that the limited storage hall will be full after 10 years. Facing the new conditions it was necessary to extend the capacity of the storage in Dukovany. So the new storage building was built, connected with the old building into one nuclear installation (as shown in Fig. 1). The capacity of the second hall is more than twice higher — 133 casks. At that time it was considered that such capacity will be sufficient for the whole operation period of Dukovany NPP. The first cask was placed into the new hall on 22 December 2006. It was a modernized type of Castor 440/84M designed for the fuel with initial enrichment up to 3.87% and with higher burnup.

FIG. 1. Dukovany NPP dry interim storage facility and new extension.

3. CURRENT SITUATION AT DUKOVANY NPP

What is the situation now? The fuel cycle is being improved continually. The original project counted with the fuel assemblies with homogenous enrichment up to 3.6% U235. An average amount of discharged assemblies per unit and year used to be more than 110 along with annual plant electricity production less than 13 TWh. Now we are using the fuel of advanced construction with radial enrichment profiling, burnable absorbers, with average enrichment 4.38% operated in 5-year cycle. The average spent fuel output is 72 assemblies per unit and year along with annual plant electricity production higher than 15 TWh. The fuel is being cooled in the pool 7-8 years and then loaded into the Castor cask.

The old storage hall is full with 60 casks, 29 casks have been loaded into the new hall, one cask is being loaded now at the reactor hall. The NPP is working hard on the LTO project (Fig. 2) to prolong the operation beyond the design lifetime which was 30 years. Options of

operation up to 40, 50 or 60 years are being analyzed. In case of operation over 45 years the capacity of our dry storage isn´t sufficient again.

FIG. 2 Storage hall of new dry interim storage facility.

A change of the cask type is also inevitable. From 2016, the fuel assemblies with higher enrichment 4.38% and higher burnup 55 MWd/kgU will be loaded. Upgraded Castor cask 84M license has been issued with set of conditions for storage of this type of fuel. This will cover a period from 2016 to 2021. Later on the other modernized types of fuel will have to be loaded. The current type of cask is not applicable for this fuel. Our company is calling a tender for cask supply beyond 2021.

4. NATIONAL STATUS

The state strategy of spent fuel management is still indefinite — ´wait and see´. The concept counts to build a final deep geological disposal for high level waste and spent fuel in 2065. An option of reprocessing of fuel is still open, however, unreal under current political and economic conditions. Until final transport of casks to somewhere we have to deal with a problem of long-term cask storage. In the time of likely repository opening many casks will be beyond their design lifetime, which has been set to 40 (older type) resp. 60 years. As an ageing management program, so far we perform regular checks of the casks according to supplier´s guide. To get knowledge, we are going to order analyses of the cask material and spent fuel deterioration for longer storage time. To proceed with spent fuel management the best way, the sharing of experiences with other countries and international cooperation is highly desirable, because we are facing the international problem with big impact on worldwide public acceptance of nuclear energy.

ENLARGEMENT OF THE OLKILUOTO SPENT FUEL INTERIM STORAGE - SPENT NUCLEAR FUEL MANAGEMENT IN FINLAND

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Abstract

Regulator's experiences having oversight for the enlargement of a spent nuclear fuel interim storage are shared in this presentation. An overview of the current situation of spent fuel management and near future plans in Finland are given. In Olkiluoto, the three reactor units have a shared interim storage for spent fuel. Considering spent fuel from these three units and the fact that the final disposal of spent fuel begins after 2020, extra space in the spent fuel storage was estimated to be needed around 2015. To conduct the enlargement, the operator was required to submit to STUK the documentation comparable to a construction license application of a nuclear facility. When conducting modifications in an existing nuclear facility, the updated safety requirements have to be followed. The major challenge in designing the enlargement was to modify the facility to to withstand a large airplane crash. Synchronization of construction phases with implementation of the system modifications proved out to be more challenging than originally considered. This emphasizes the importance of good, thorough and detailed planning of each part of the project and overall coordination between project packages. The spent fuel is cooled in the interim storages before encapsulating it for the underground disposal. The construction license application of the nuclear waste facility was submitted at the end of 2012 and the operation is estimated to begin after 2022.

1. SPENT FUEL MANAGEMENT IN FINLAND

The Finnish nuclear legislation sets the framework to the spent nuclear fuel management in Finland [1]. It defines that the nuclear energy producers are responsible for the spent nuclear management and excludes the possibility to export or import spent nuclear fuel. These conditions have formed practices for the spent fuel management in Finland: the spent fuel from the reactors is stored for few decades in the spent fuel interim storages and then transported to an encapsulation and disposal facility.

In Finland, there are four operating reactors, one under construction and two licensees having a Decision-in-Principle to proceed with the planning and licensing of a new reactor. The first nuclear power plant (NPP) units are VVER-440 PWR reactors in Loviisa. They began operation in 1977 and 1980. These units are owned by Fortum Power and Heat Oyj (Fortum). The other two operating reactor units are ASEA-Atom BWR type reactors in Olkiluoto. These Olkiluoto 1 and 2 units began operation in 1978 and 1980. The reactor unit under construction is an Areva EPR reactor in Olkiluoto. All the Olkiluoto NPP units are owned by Teollisuuden Voima Oyj (TVO). TVO also has a Decision-in-Principle to construct a fourth reactor unit in Olkiluoto.

The other holder of a Decision-in-Principle to proceed with the planning and licensing of a new reactor is Fennovoima. Fennovoima is a newcomer in the Finnish nuclear energy sector. Fennovoima's nuclear site will be on the coast of the Bothnian Bay in Hanhikivi, Pyhäjoki.

The spent fuel from Loviisa NPP units was originally transported to the Soviet Union and later to Russia for reprocessing until 1996. Since then, after the change of Nuclear Energy Act, which prevented the export of spent nuclear fuel, Loviisa stored the spent fuel of the two units in a spent fuel interim storage. Fortum has increased the Loviisa NPP spent fuel storage capasity gradually by replacing the original storage racks by dense storage racks.

In Olkiluoto, all the spent fuel have been stored in the spent fuel interim storage. The option for enlarging the storage was foreseen in the original design. More storage capacity was needed at the interim storage around 2014 because Olkiluoto 1 and 2 units have some decades of operating time left, the OL3 unit is expected to begin the operation in the future and the disposal of the spent fuel will begin around 2022. These aspects set the boundary conditions for the amount of capacity needed at the interim storage. TVO, the licensee in Olkiluoto NPP site, decided to double the number of spent fuel storage pools. This enlargement project will be discussed in chapter 2.

After the spent fuel of Fortum's and TVO's reactors is cooled few decades in the interim storages, it will be transported to the encapsulation plant and then transferred to disposal facility. The encapsulation plant and disposal facility form together a nuclear waste facility to be operated by Posiva Oy which in turn is co-owned by Fortum and TVO. The nuclear waste facility will be situated at Olkiluoto site. Posiva submitted an application for the construction license of the facility in the end of 2012. The operation of the nuclear waste facility is planned to begin around 2022. The nuclear waste facility and its licensing is discussed in chapter 4.

Fennovoima has not yet specified the spent fuel management in detail. The Decision-in-Principle of Fennovoima is valid until 30th June 2015 and the company has to apply for a construction license before that. It is prerequisited in the Decision-in-Principle that the type of the spent nuclear fuel storage and an overall plan for nuclear waste disposal have to be included in the construction license application. The options for Fennovoima's spent fuel disposal are either to co-operate with existing power companies and Posiva or to design its own disposal facility.

2. OVERSIGHT OF THE SPENT FUEL INTERIM STORAGE ENLARGEMENT

TVO had a need to enlarge its spent fuel storage facility due to construction of a new NPP unit and increase of the operating life of the existing NPP units. The possibility of enlarging the spent fuel storage was considerd in the original design of the storage and the enlargement was included in the operational license of Olkiluoto NPP 1 and 2 units. The licensing of the enlargement project was conducted as a major plant modification. Prior to the construction of this modification, the operator was required to submit to STUK an application which was as detailed documentation as needed for applying for the construction license of a new nuclear facility.

The Olkiluoto spent fuel interim storage is a pool type storage. Originally there were three storage pools and the operator desided to enlarge the capacity to six pools. The original spent fuel storage was commissioned in 1988. When conducting changes in an existing nuclear facility, the updated safety requirements have to be followed. The major challenge in designing the enlargement was to modify the original part and design the new part to withstand a large airplane crash (APC) [2]. The operator designed stuctures outside the spent fuel storage and some features inside the storage that protect the fuel pools from direct impact in case of APC.

The designing of protective structures was an optimisation task between safety issues that were partly opposite to each other. The protective structures for the spent fuel pools had to be dimensioned light enough to enable the handling but strong enough to withstand the impacts followed by an APC to the storage building. The risk of these structures to fall in the pools and consequences to the stored spent fuel had to be minimized.

When designing protective structures the licensee had presented several design options that it stated to meet the regulatory requirements and had then chosen the most suitable one considering all the aspects. The acceptability of the solutions presented by the licensee was not always obvious. Computational and experimental analyses were needed to evaluate the acceptability of the chosen solutions.

At the moment, the enlargement project is almost completed. The construction activites and commissioning the system modifications are already done. The decision to approve the increased storage capacity will be given by STUK during 2015.

3. SAFETY OF THE OPERATING SPENT FUEL STORAGE DURING ENLARGING CONSTRUCTIONS

The enlargement of an operating spent fuel interim storage facility is a challenging task because the operating spent fuel storage has to fulfil all the safety requirements despite construction activities at the facility. The same principle has to be followed also when implementing system modifications at the spent fuel storage. To comply with these principles it is essential to plan in detail the construction, system modifications, their phases and synchronization. To avoid the disturbances at operating part of the Olkiluoto spent fuel storage facility, the construction work of the new structures outside the existing storage building was finished before the integration of the buildings and systems.

Despite all the detailed planning carried out in the project an operational occurrence compromised a safety feature of the spent fuel storage facility. The safety feature compromised was the underpressure maintained in the radiation protection monitoring area of the spent fuel storage. The underpressure of the radiation monitorig area is an additional safety feature aiming to minimize the spread of radioactive substances out side the facility by airflow.

The initiating event was an unexpected stopping of a supply air ventilation fan of the storage. The exhaust air ventilation fan continued functioning normally. The system modification to increase the ventilation capacity by increasing the exhaust ventilation fan capacity had been conducted earlier to meet the needs of the enlarged spent fuel storage. When the supply air ventilation stopped, the exhaust air ventilation fans kept functioning normally which resulted an increase of underpressure inside the original storage building. A temporary protective element wall had been built at the end of original pool hall to separate it from construction activities of the enlargement. The temporary element wall was the outer wall and also the boundary wall of the original storage pool hall. For this reason the temporary wall was dimensioned to withstand the underpressure inside the building and to meet the leak tightness requirements.

This excessive underpressure caused by the ventilation malfunction exceeded the strength of the temporary wall and resulted in an inward bulging of it. The deformation resulted the loss of air tightness and then the loss of underpressure of the pool hall. There was not any risk of serious consequences of this occurrence such as radioactive releases because spent fuel was not handeled at the time. Normal conditions were restored by starting the redundant ventilation system and repairing the temporary wall. Also some modifications to ventilation system were carried out later on to avoid the recurrence of the same operational occurence.
The case described above showed the challenge of synchronizing construction phases with the process system modifications of the storage. Even though all the operational occurrences can not be anticipated, careful planning of the construction and system modification phases can

minimize the consequences of operational occurrences. For example in this case, the modifications of exhaust air fans could have been postponed to later phase of the project. Also the design feature of the ventilation system that stops the ventilation fans if the air flow of one end decreases enough would have saved the air tightness of the pool hall. This design feature has been implemented at the spent fuel storage later on.

4. THE MODIFICATIONS FOR SPENT FUEL STORAGES DUE TO THE FUKUSHIMA ACCIDENT

Due to the stress-tests made for the nuclear power operators after the Fukushima accident, STUK required the operators to investigate how the nuclear power plants are prepared to and will survive with exceptional natural phenomena and disturbances in external power supply.

At Olkiluoto site, the licensee identified the following necessary modifications to improve the safety of spent fuel storage:

- A connection to the spent fuel storage building to feed cooling water from an external source;
- Spent fuel pool water level and temperature monitoring systems that are independent of power supply and that can withstand the design earthquakes;
- Improve storm water drainage;
- Prevent the seawater from flooding the storage building;
- Transportable fire water pumps at the site;
- Improved availability of raw water at the site.

The cooling water connection in the outside wall of the storage building was designed within the enlargement project already before Fukushima accident. These improvements are mostly aldready implemented at the Olkiluoto spent fuel storage.

At Loviisa site, the improvements for the spent fuel interim storage due to stress tests were:

- Seismic resistance analyses of fuel pools;
- Connections to feed cooling water from external sources;
- A steam out flow route for the situations when cooling the spent fuel by boiling the pool water;
- Improved availability of water level and temperature monitoring.

Some of these improvements mentioned above are already conducted as system modifications. For the rest of the modifications the basic designing is done and the detailed planning is about to begin.

5. SPENT NUCLEAR FUEL DISPOSAL FACILITY

In Finland the chosen strategy of spent fuel mangement is the once through cycle and the disposal of spent fuel in the bedrock [3]. As mentioned earlier in this paper in chapter 1 Posiva submitted the construction license application in the end of 2012 of the nuclear waste facility which consist of an encapsulation plant and a disposal facility.

The encapsulation plant will be constructed above the underground disposal facility. In the encapsulation plant the spent fuel bundles are packed in cast-iron-copper canisters that are closed leak and air tightly by welding. Canisters are transferred to the disposal facility by a

canister lift and disposed of in disposal holes. The disposal holes and the tunnels are filled with clay material that protect the canister from water and bedrock movements.

The contact authority of a nuclear facility licensing is the Ministry of Employment and the Economy (MEE). The ministry requests a statement on safety of the facility from STUK. STUK's safety assessment is based on the technical documentation the applicant provided to STUK as part of the license application. The contents of the technical documentation is specified in Nuclear Energy Decree [4]. During the review of Posiva's construction license application STUK requested Posiva for additional information or clarification in topics that were not described in sufficient detail in the application. In the beginning of this year STUK finalized the safety assessment. The main conclusion was that planned nuclear waste facility can be built to be safe. There is a sufficient reliability that there are no detrimental radiation effects to the public or environment. In the statement to the Government [5] STUK raised areas that need further development before specific construction step or before submittal of operating license application.

6. CONCLUSIONS

The enlargement project of the spent fuel interim storage has emphasized the significance of detailed planning of the modifications of an operating spent fuel interim storage. Also the new safety requirements set the challenges to the designing of the enlargement of existing spent fuel storage. The detailed planning of the synchronizing the construction and system modification activities proved to be important to minimize the consequences of unexpected operational occurrences

There are not any major challenges to be solved in the spent nuclear fuel management in Finland. The spent nuclear fuel storage operations and the proceeding of the nuclear waste disposal facility licensing have followed the chosen spent nuclear fuel management strategy set by the Government in 1983. Also the target time line set in the strategy has been followed so far.

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LONG TERM STORAGE OF SPENT NUCLEAR FUEL AND HLW IN DUAL PURPOSE CASKS TOWARDS DISPOSAL — CHALLENGES AND PERSPECTIVES

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Abstract

The safe and secure interim storage of spent nuclear fuel and high-level waste from nuclear power reactors is one of the major long-term issues to be solved in the nuclear business. Today no country has a repository available for final disposal or in some countries reprocessing capacities for spent fuel are not sufficiently available as planned. In the meantime the safe interim storage for the long-term — may be 80 years or even longer — is the only available option from today's perspective and dry cask storage is increasingly favoured throughout the world for various reasons. This paper describes and discusses the major challenges of spent fuel management in Germany after the phase-out for nuclear electricity generation was decided in 2011 and a new repository siting procedure was implemented in 2013. Consequences from those decisions which were legally founded by amendments of the German Atomic Energy Act result in the need to transfer all remaining spent fuel from limited reactor operation (last reactor shutdown until the end of 2022) into casks for subsequent dry interim storage on-site. Storage licenses are generally issued site-specific considering specific dual purpose casks (DPC) and their inventories. These storage licenses are generally limited to 40 years so far. But the need for extending that interim storage period in the future has become obvious. Even though, this seems to be not an issue to be solved today questions about additional safety demonstrations will arise as soon as licenses need to be extended. Certainly, these questions will ask for reliable data about the long-term performance and safety of structures, systems, and components, e.g. the long-term performance of cask components and materials like bolted closure systems including metal gaskets, or fuel rod behaviour concerning cladding materials under stress and temperature conditions. In case of dual purpose casks for storage and transportation this includes aspects on how to demonstrate transportability during or after several decades of interim storage. Long-term investigations often require plenty of time and therefore need to be initiated timely. Preliminary R&D results and experiences from more than 20 years of safe interim storage in Germany are discussed. In addition, the national perspective towards potential data and R&D needs to demonstrate safety for extended interim storage periods is related to international actions as the US Extended Storage Collaboration Program (ESCP) and several IAEA projects focussing on that issue.

1. INTRODUCTION

The safe and secure management of spent nuclear fuel from commercial nuclear power plants and research reactors and high-level waste from reprocessing is a major global issue. Most countries are planning disposal in deep geological formations for the final solution but only a few have proceeded to open a repository within the next decade like Sweden and Finland. Other countries have to face significant delays of their initial disposal programmes, e.g. Germany and the US. Generally growing gaps between discharging the nuclear fuel from operating or shut-down reactors and the availability of final disposal lead to increasing importance of interim storage solutions and their long-term performance. In the future initially licensed interim storage periods will have to be extended for significantly longer storage periods. Most concepts consider dry storage in various cask or canister systems after an initial wet storage cooling period as the most favourable option.

2. SPENT FUEL MANAGEMENT IN GERMANY

The initial spent fuel management policy in Germany considered dry spent fuel and HLW storage in dual purpose transport and storage casks at two central interim storage facilities Gorleben and Ahaus. Since 1994 two options were mentioned by the German Atomic Energy Act (AtG): Direct disposal and reprocessing of spent fuel. After the two central interim storage facilities started their operation contamination issues during spent fuel shipments to France for reprocessing finally led to a transport ban followed by eliminating the reprocessing option in 2005 and ending the existing contracts with France and the UK. In addition all spent

fuel from nuclear power reactors was decided to be stored at the nuclear power plant sites and therefore 12 on-site interim storage facilities were licensed, built and taken into operation. Additionally, AVR research reactor fuel is stored in an interim storage at the Jülich research centre, THTR fuel at Ahaus, and VVER fuel from the shut-down VVER reactors of former Eastern Germany (GDR) at Greifswald. This means 16 interim storages with more than 1,000 loaded casks are operated currently.

The dry interim storage concept for spent fuel and HLW was established around the 1980s and has never been changed since that time. Most recently the German Nuclear Waste Management Commission (ESK) has published revised "Guidelines for Dry Cask Spent Fuel and Heat-generating Waste" [1]. The document describes major safety goals and related design requirements as well as the technical concept of accident safe dual purpose casks for transportation and storage. The storage period is generally limited to 40 years. This period results from the former spent fuel management concept consisting of centralized interim storage, a waste conditioning plant and a geological repository in salt being available until 2035 at the latest.

The dual purpose casks consist of a monolithic thick-walled cask body made of ductile cast iron or forged steel and a bolted double barrier lid system equipped with metal gaskets. The lid system is permanently monitored for proper leak-tightness. However the lid closure systems are qualified for the entire storage period of up to 40 years so far, a qualified repair concept in case of a hypothetical lid failure is established as well. The cask interior is vacuum dried and helium filled providing inert conditions throughout storage minimizing potential degradation effects of fuel assemblies and fuel baskets. Generally, all major safety goals are fulfilled by the casks, though they are stored in a storage building providing monitoring, handling and maintenance equipment as well as weather protection and additional shielding. A specific additional requirement concerns a valid Type B(U) cask approval prior to loading and to guarantee transportability of all stored packages at any time during the entire storage period.

The storage of nuclear fuel, including spent fuel and radioactive waste with significant contents of fissile material requires (if the proportion of certain uranium and plutonium isotopes exceed the limits specified § 2, para. 3 AtG) a license under § 6 AtG. The licensing authority in this instance is the Federal Office for Radiation Protection (BfS), while supervision is performed by the competent authority of the respective 'Land' (Federal State). Storage licenses issued by BfS contain all relevant safety evaluations to satisfy the protection goals safe enclosure, shielding, subcriticality, and heat dissipation under operational and accidental conditions of the specific storage facility and define conditions and requirements for safe and secure operation. As mentioned ESK guidelines [1] define basic design and approval requirements. "ESK Guidelines for Periodic Safety Inspections and Technical Ageing Management for Interim Storage Facilities for Spent Nuclear Fuel and Heat-generating Radioactive Waste" [2] offer guidance concerning actions, documentation, and evaluation of storage operation experience including appropriate ageing management systems.

The nuclear policy in Germany changed dramatically and quickly after the Fukushima disaster in March 2011. The German parliament decided to phase out nuclear energy production by shutting down 8 NPPs immediately and the remaining 9 until the end of 2022. This became legally binding with an amendment of the Atomic Energy Act in August 2011. In addition the repository siting and exploration process in Germany, so far performed for the Gorleben salt dome, was changed and a new 'Repository Site Selection Act' was entering into force in July 2013. By this law the repository siting process was completely reset including

the identification of criteria and procedures for search and selection of potential repository sites. The 'Commission on the storage of highly radioactive materials' formed by 33 members (1 Chair, 8 scientists, 8 representatives from social stakeholder organizations, 8 members from the German parliament, 8 'Länder' ministers) includes comprehensive public participation and is required to present a final report by the end of 2015 or maybe mid of 2016. Thereafter potential host rock formations for further exploration have to be identified by 2023 and the final site selection has to be decided by 2031. If this ambitious schedule comes true it is followed by license application, evaluation, approval and finally repository construction. Taking into account at least 20 years more to get a repository ready to operate this is going to happen not before almost the year 2050. Concerning current interim storage licenses expire between 2032 and 2047 followed by repository operation for at least 30 years future extended interim storage periods of several decades have to be taken into account right now.

3. TECHNICAL ISSUES CONCERNING SPENT FUEL STORAGE IN DUAL PURPOSE CASKS BEYOND THE INITIAL LICENSE PERIOD

Since the 1990s dual purpose casks are used and stored at several interim storage facilities throughout the country. Safety evaluation was performed for a maximum license period of 40 years including potential accidental scenarios on site and degradation effects. Due to robust design features and qualified and approved manufacturing, loading and storage procedures relevant stress factors have been minimized. This includes potential corrosion from the interior and from the outside. Therefor careful vacuum drying procedures are essential to ensure inert conditions in the cask cavity and also at the metal gasket closure systems. From the outside protection measures are established to avoid corrosion effects by moisture and air pollutants. For that reason inspection and maintenance measures during storage operation could be minimized and all relevant safety functions are guaranteed passively. So far more than 1,000 casks have been loaded and stored for partially more than 20 years without any major problems. About 600 additional casks are expected to store all spent fuel from remaining nuclear power plant operation.

As mentioned before extended interim storage of spent fuel and HLW beyond 40 years is an upcoming issue in Germany. Although the initial time limit is not a technical limit any license extension or renewal will require additional safety demonstrations taking into account potential degradation effects, ageing mechanisms, assessment methods and the current technical and scientific state of the art. That means systematic analyses are required to identify relevant degradation mechanisms and stress factors (thermal, mechanical, radiological, environmental). Furthermore investigations will be necessary to provide information and data about quantitative changes in material properties and system function. Examples are the long-term performance of metal gaskets as one of the most sensitive cask components to guarantee the safe confinement of the radioactive materials and their reliable extrapolation over time and temperature. BAM has already started a research programme in that area and publishes results continuously [3]. Currently the programme is going to be extended by support from additional external funding. Other issues BAM started to investigate are the degradation of polymers used for neutron shielding purposes due to thermal and radiological stress factors and related time dependent changes of material structure and gas release [4], and the ageing behaviour of elastomeric O-ring seals used to enable leakage rate measurements of the metal barrier gaskets [5]. Furthermore, the long-term performance of the cask interior including cladding integrity is a potential issue. Fuel elements cannot be inspected inside the casks during storage without additional elaborate measures and it has to be agreed timely (and preferably before all spent fuel is loaded into

casks) what data and information need to be gathered for long-term integrity demonstrations for extended storage periods.

In a single case an interim storage license at the Jülich research centre (152 casks containing AVR research reactor fuel) did already expire after only 20 years in 2013. Also in this case the time limit was fixed for political and not for technical reasons. With the requirement to ensure transportability of all casks at any time an inspection programme was initiated to demonstrate sufficient cask properties prior to public transportation. Basis is the valid Type B (U) cask type approval but this is not sufficient [6, 7]. In addition every single cask has to demonstrate proper compliance with all defined and required properties by the cask type approval. For that purpose especially the bolted lid systems have been investigated in detail demonstrating sufficient bolt condition and pretension and also sufficient leakage rates of the primary lid systems of a representative selection of casks. This became necessary even though proper seal system function is permanently monitored during storage but without providing specific leakage rates at all times. Although all inspections were performed successfully no shipment took place so far because no subsequent storage option could be licensed whether on site or at another facility like Ahaus. Thus until today ongoing storage beyond the initially licensed period is just tolerated by the responsible state authority for a limited time and a destination for the spent fuel has to be established as soon as possible.

4. STRATEGIC ASPECTS CONCERNING EXTENDED INTERIM STORAGE PERIODS

Basically dry interim cask storage in Germany was established at a time when spent fuel management was just a minor issue and the expectation was to have limited interim storage until a deep geological repository would be available timely. But as mentioned the national nuclear policy changed several times ending up with the final nuclear phase out decision and a complete reset of the repository siting procedure leading to the likely need for significantly extended interim storage periods. Today, for various and sometimes similar reasons many other countries also face the need for extended spent fuel storage towards reprocessing or disposal. This encouraged many organisations such as IAEA and OECD/NEA to draw increasing attention to this issue. Several programmes were established including research programmes by national and international organisations with the main purpose to identify and address relevant technical and scientific issues and to share and discuss gathered experience. A major initiative was taken by the United States with the Extended Storage Collaboration Program (ESCP) under the lead of the Electric Power Research Institute (EPRI). Among many others BAM is also taking part and currently chairs the ESCP International Subcommittee.

Besides the technical aspects of extended interim storage also regulatory conditions have to be considered. In Germany storage licenses are issued specifically under §6 of the Atomic Energy Act (AtG) and independently from co-located nuclear power plants as well as from regulations and approval procedures for radioactive materials transport. Whereas transport regulations are ruled by internationally agreed IAEA standards storage licenses are issued nationally concerning site specific conditions. Both package design approvals for cask transportation and storage licenses are assessed and issued independently from each other by separate applicants and competent authorities but with the general request of the permanent transportability of all casks during storage to be provided by valid package design approvals for all cask types. As transport regulations are focussed on usual transport regimes including loading and unloading of casks with the opportunity to regularly inspect all cask components as requested this is in parts not applicable to casks being permanently loaded and stored. In

addition package design approvals are very limited in time (3 to max. 10 years) and need to be renewed regularly also during interim storage periods. With the perception that all casks need to be shipped subsequently to interim storage periods IAEA started an initiative to consider this aspect and to improve the regulations in the future. This process is still ongoing and substantial results have already been developed for further consideration in future revisions of the regulations.

Concerning the long-term aspects of current and extended interim storage periods every dual purpose cask is operated at just one single place at a storage facility under the approved operation regime. Every cask also needs subsequent transportation to another facility whether for further interim storage or preparation of the spent fuel or HLW for final disposal at a later time. This means necessary actions to demonstrate and manage proper cask conditions during storage and for later transportation need to be harmonized and merged to just one maintenance and ageing management programme as part of the interim storage operation regime including periodic safety reviews. In the meantime this issue has been noticed as becoming essential towards a consistent spent fuel and HLW management concept in the sense of a holistic approach taking into consideration all areas of operation and last but not least also future disposal. The major target course is to establish effective and efficient programmes to demonstrate long-term safe and reliable operation for current and extended interim storage periods by means of on-site inspections, ageing management programmes and periodic safety reviews added by basic research programmes whenever needed. Proactively or timely gathered data and knowledge on relevant long-term degradation mechanisms of materials, systems and components including the cask interior will provide various benefits for required future safety assessments along with license extensions, radiation dose minimization, cost effectiveness, and for development of appropriate interfaces concerning final disposal concepts.

5. CONCLUSIONS

The national spent fuel and HLW management approach for interim storage towards final disposal has seen several changes during the past more than 30 years. Currently the established interim storage concept mainly based on approved dual purpose casks for storage and transportation has demonstrated safe and secure operation at several centralized and on-site storages for up to more than 20 years. Even though storage licenses of most sites expire between 2032 and 2047 changes in the national repository siting process will lead to extended interim storage periods in the future. To provide necessary improved knowledge and data to assess safety for extended periods of operation of casks and components appropriate programmes need to established timely under consideration of transportation as well as storage conditions. BAM has already started investigations of metal gaskets and polymers but more comprehensive programmes will be needed in the future. Merging regulatory requirements from the transportation and the storage area to a consistent approach for loaded casks in long-term interim storage followed by transportation is a major challenge which in the meantime is addressed by IAEA and national authorities to improve the regulations.

Besides technical issues for the licensing of extended interim storage periods providing sufficient safety assessments also public and political perception will play a dominant role. Current interim storages have been accepted and licensed with a limitation of interim storage periods to a maximum of 40 years. Extension of that period at the same sites will need many efforts at persuasion and a broad political and public consensus. But this becomes also relevant in case casks are intended to be shipped to another (central) site whether a new or already existing one. Finally extended interim storage will be part of the entire national

disposal strategy and a holistic approach for managing the spent fuel and HLW taking into account all areas of storage, transportation and disposal should be the best way towards a consent based and long-lasting reliable future approach.

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DESIGN AND CONSTRUCTION WORK EXPERIENCE OF INTERIM STORAGE FACILITY FOR SPENT FUELS

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Abstract

Recyclable-Fuel Storage Company (RFS) has completed construction works of the storage building and fabrication works of several metal casks for spent nuclear fuels of light water reactors as the Japanese first off-site interim storage operator. RFS are now undertaking the safety review under new regulation standards based on the lesson learned by Fukushima-Daiichi accident and preparing for the start of operation. This paper introduces the outline of the design and construction work experience for this facility. This interim storage facility is intended to temporarily store the spent nuclear fuels of light water reactors till they are reprocessed. The building has a storage capacity of up to 288 dry metal casks. The storage building has natural air cooling systems for removing heat from the filled casks. The centre part of the storage area has high-height exhausting part for effective ventilation. On the other hand, the other part of the storage area has low-height ceiling because metal casks are moved by the air pallet transporter instead of overhead crane. Metal casks are designed so that the basic safety feature of the casks such as containment, shielding, sub-criticality and heat removal are safely maintained during storage periods of 50 years. Negative pressure is kept for inner part of casks and positive pressure is kept for the part between two lids. The casks are also filled with helium gas so as to convey heat effectively and to maintain the fuel integrity. These casks are used for dual purpose i.e. both storage and transport. For transport, tertiary lid and impact limiters are also set on the casks. During storage, visual inspection, pressure monitoring between the double lids, temperature checking on the casks and monitoring of radiation dose of inner storage building, etc. are carried out. The establishment permit for spent fuel storage and approval of design and construction methods were issued in 2010. After the approval, RFS started the construction of the spent fuel storage facilities. Construction works of storage building including installation of the electrical and mechanical equipment were completed in August 2013. Fabrication of several metal casks was also completed with various inspections as transport and storage casks. Currently, RFS has been undertaking NRA review under new regulation standards. As preparation work for the start of the operation, RFS carried out the training of the handling casks using mock-up cask, air pallet transporter, handling equipment and transport vehicle.

1. OUTLINE OF THE STORAGE FACILITY

RFS was jointly invested and established on Nov. 2005 by Tokyo Electric Power Company and the Japan Atomic Power Company to serve as a company specializing solely in spent fuel storage. Recyclable-Fuel Storage Centre (RFSC) is located in Mutsu City of Aomori prefecture (Fig. 1). The interim storage facility is intended to temporarily store the spent nuclear fuels till they are reprocessed. The spent nuclear fuels are planned to be stored for 50 years.

1.1. Storage building

RFS plans to eventually store about 5000 ton-U of spent fuels. The first building of RFSC has a storage capacity of up to 288 dry metal casks (3000 t(U)) (Fig. 2). The storage building is approximately 62 m in width, 131 m in depth and 28 m in height. The storage building has natural air cooling systems for removing heat from the filled casks. The centre part of the storage area has high-height exhausting part for effective ventilation. On the other hand, the other part of the storage area has low-height ceiling because metal casks are moved to storage area by the air pallet transporter instead of overhead traveling crane. (Fig. 3).The building has 1-story structure above the ground level, which was made of steel reinforced concrete.

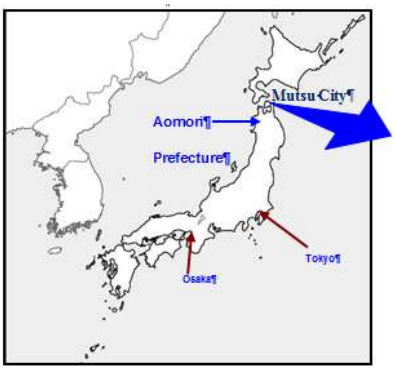

FIG. 1. Location of Mutsu City.

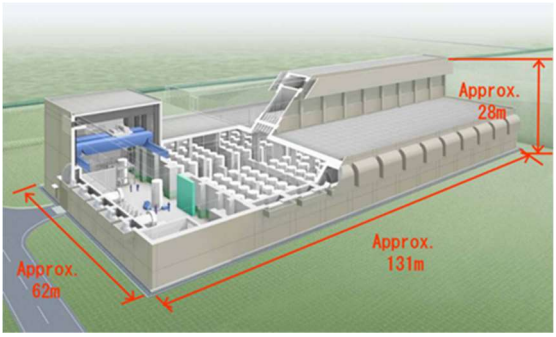

FIG. 2. Image of a storage building.

The storage building was separated into receiving area, storage area and auxiliary area (Fig. 3).

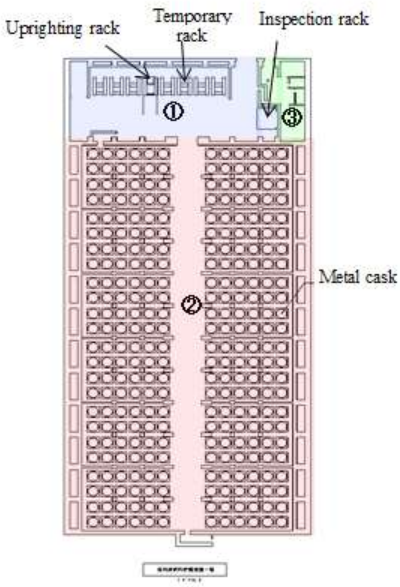

FIG. 3. Each area of storage building.

General description of each area:

(1) Receiving area (casks are moved by using cask transport vehicle and overhead traveling crane):
   - Temporary racks: there are seven racks where casks received are temporarily placed;
   - Uprighting rack: there is one rack that is dedicated to upright horizontally placed casks;
   - Inspection rack: there is one rack where casks are inspected, etc.

(2) Storage area (casks are moved by using air pallet transporter):
   The area where metal casks are stored vertically. The area has a space where 288 casks can be stored.

(3) Auxiliary area.

1.2. Metal cask

The metal cask (Fig. 4) for BWR spent fuels can store 69 fuel bundles (approx. 10 t(U)). This cask is designed to be used for both storage and transport, and the cask is approximately 2.5 m in diameter and 5.4 m in length. Metal cask is designed so that the basic safety features of the cask such as containment, shielding, sub-criticality and heat removal are safely maintained during storage periods of 50 years.

1.2.1. *Containment*

The metal cask has a multiple containment structure with primary lid and secondary lid. To seal the lids, metal gaskets are used from the viewpoint of maintaining the containment function over a long period. Negative pressure is kept for inner space of the cask and positive pressure is kept for the space between two lids. If metal gasket leak happened, it can be detected as the pressure of the space between the lids decreases. Moreover, the cask has a structure that can mount a tertiary lid, which uses an elastomer gasket, at the time of transport.

1.2.2. *Shielding*

Cask body, primary lid and secondary lid have a function of gamma-shielding and resign is used as neutron shielding material.

1.2.3. *Subcriticality*

The spent fuels are loaded into the space sectioned by basket plates, and boron is added in the basket plates as a neutron absorbent so as to maintain sub-criticality.

1.2.4. *Heat removal*

Fins are set on the cask and the cask is filled with helium gas so as to convey the heat effectively to the surface of the cask and to be cooled by natural air.

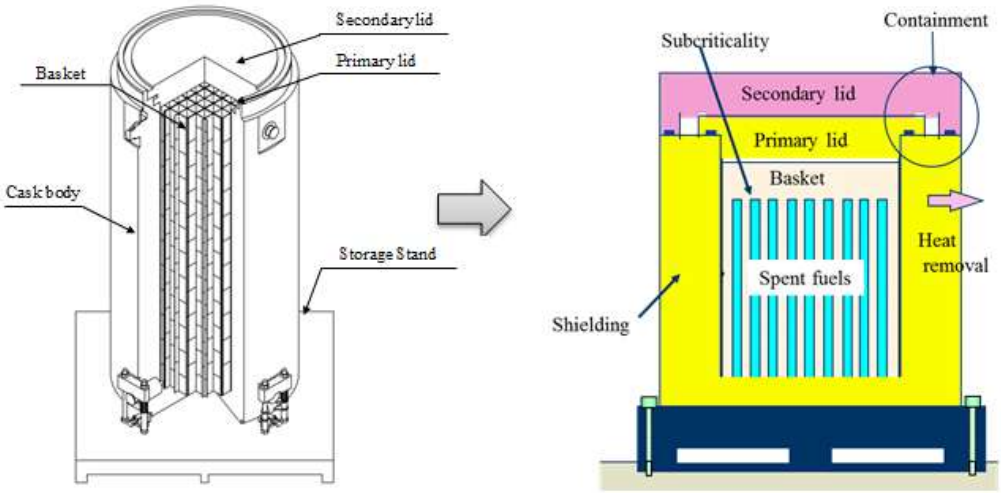

FIG.4. Image of the metal cask.

2. CONSTRUCTION OF THE FACILITY

RFS submitted application for establishment permit to operate the RFSC to the Minister of Economy, Trade and Industry in March 2007.The establishment permit for spent fuel storage for RFSC was issued in May 2010. Then RFS submitted application for approval of design and construction methods for spent fuel storage facilities to the Minister of Economy, Trade and Industry in June 2010. RFS started construction work of the storage building and fabrication work of metal casks in August 2010 after the issue of the approval of the design and construction methods. Although construction work of the storage building was temporarily suspended under the influence of the Great East Japan Earthquake of March 2011 for about 1 year, construction work was completed in August 2013 (see Fig. 5). Installation of the all electrical and mechanical equipment was also completed. Fabrication of several metal casks was also completed with various inspections as transport and storage casks (see Fig. 6).

FIG. 5. Overview of the storage building.

FIG. 6. Metal cask.

3. OPERATIONAL CONCEPT OF THE FACILITY (FIG. 7)

RFS takes charge of storage work and electric power companies take charge of fuel loading work and transport work before and after the storage between the facilities. In nuclear power plants, spent nuclear fuels which are cooled in the spent fuel pools, are loaded into the metal casks Fig. 7. Filled casks are transported to the port near the RFSC by the dedicated INF-3 ship and then the casks are carried to the storage building by dedicated transport vehicles through the exclusive road. Filled casks are handed over from electric power companies to RFS with the records of fuels loading and pre-shipment inspection, etc. after receiving inspection at the storage facility. Metal casks are carried to storage area by air pallet transporter and fixed on the storage position. During storage, visual inspection, monitoring pressure in the space between lids of casks, monitoring temperature of air supplied into and exhausted from the storage building, temperature checking on the casks and monitoring radiation dose of inner storage building, etc. are carried out by RFS. For transport after storage, records during storage are handed over to the electric power companies and pre-shipment inspection is carried out by electric power companies.

The mock-up cask (Fig. 8) is used for operational training. The mock-up cask is simulated the real metal cask and the size and the weight of the mock-up cask are almost same as those of the real metal cask.

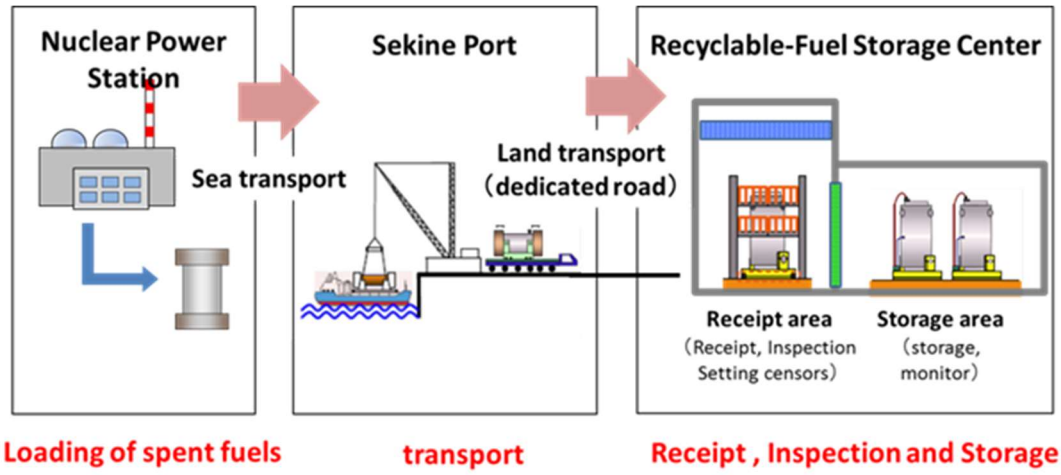

FIG. 7. Image of operational concept.

FIG. 8. Mock-up cask.

4. CURRENT STATUS

In December 2013, the new regulation standards based on the lesson learned by Fukushima-Daiichi accident were enforced for all facilities of nuclear fuel cycle. RFS conducted the further assessments for earthquakes, tsunami, volcanos and tornado etc. and RFS submitted the application for renewal of establishment permit to operate the RFSC to the Nuclear Regulation Authority (NRA) in January 2014. RFS has been undertaking NRA review under new regulation standards as of April 2015.

As preparation work for the start of the operation, RFS carried out the training of the handling cask using mock-up cask, air pallet transporter, handling equipment and transport vehicle.

STATUS AND PROSPECT OF SNF REPROCESSING AT MAYAK PLANT

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Abstract

Currently, Russian reprocessing of used nuclear fuel (UNF) is carried out at the radiochemical plant RT-1 (FSUE "PA "Mayak"), which was built and put into operation in the USSR in 1977. Today RT-1 has three independent process lines for treatment of VVER-440, RBMK-1000, BN-350 and BN-600 used fuels. Used fuels from research and naval propulsion reactors are also treated in one of the three process lines. RT-1 operator with the support of Rosatom is actively working to repatriate nuclear fuel from reactors of Russian origin, thus improving safety of research reactors and contributing to an ongoing IAEA programme aimed at reducing the risk of nuclear proliferation and terrorist threats. Due to the uniqueness of the technical capabilities, the reprocessing plant can manage a wide range of used nuclear fuel (UNF). At present RT-1 is undergoing modifications connected with a new programme to expand the range of UNF that can be reprocessed there. For the execution of the programme, production technology is being created to enable the recycling of UNF (including the defective one) from AMB, VVER-1000, BN-800/1200, EGP-6 and research reactors with the current RT-1 capabilities.

1. INTRODUCTION

Since the start of reprocessing at Mayak, the methods of the safe conditioning of process waste have changed. Since 1987, the process waste from the plant has been vitrified using EP-500 melter system with direct electric heating. The main objective of this facility is to immobilize high-level waste (HLW) and intermediate-level waste (ILW) in a sodium-alum phosphate glass. After expiration of their service life, four furnaces previously used at Mayak have been put out of operation. The fifth furnace EP-500/5 is currently under construction. The start of HLW reprocessing with this melter is planned for 2015. Years of experience in the field of managing used nuclear fuel (UNF) have shown our customers high reliability and stability of Mayak as a partner and its readiness to develop cooperation in the field of nuclear technologies. Mayak is now seeking new reprocessing markets, including treatment of defective fuel, in countries, that have accumulated several thousand tons of irradiated fuel. We are ready to receive UNF from VVER-440 reactors operated outside the Russian Federation in Bulgaria, Hungary, Slovakia, Czech Republic, Ukraine, Armenia and other countries.

2. BACKGROUND/CURRENT STATUS

The basis of the current plant for reprocessing of UNF was the facility for separation of weapons-grade plutonium from irradiated uranium slugs. The decision to build a facility at the Mayak PA for regeneration of used nuclear fuel was made in June, 1967 by the Council of Ministers of the USSR. The first train delivered used nuclear fuel to the plant in 1970. The operation of RT-1 plant was started in March, 1977. As of the beginning of 2015 the plant had reprocessed more than 5,650 t of used nuclear fuel in terms of uranium.

Today RT-1 plant reprocesses used fuel from VVER-440 and BN-600 nuclear power reactors, naval propulsion reactors, and various fuels from research reactors. The total annual volume of reprocessed fuel amounts to 160 t in terms of uranium.

The plant is mainly engaged in reprocessing UNF from VVER-440 reactors. In addition to reprocessing VVER-440 fuel from the Russian nuclear power plants Kola and Novovoronezh, the plant treats used nuclear fuel from the nuclear power plants Kozloduy (Bulgaria) and

Rovno (Ukraine) as well. In 2014 defective fuel from the nuclear power plant Paks (Hungary) was transported to Mayak for reprocessing.

As a result of reprocessing of used nuclear fuel at the plant, 150 t of UNH with 1% enrichment in terms of uranium, and several tons of U_3O_8 with the enrichment greater than 5% in terms of uranium are produced annually. Up to 1.5 t of plutonium dioxide (in terms of metal) is also extracted annually and transferred to the storage facility.

3. PROSPECTS/DEVELOPMENTS

The current priority for RT-1 plant is to increase the amount of fuel reprocessed, including expansion of its range.

At the beginning of 2015, one of the three production lines at the plant was put out of operation to facilitate modifications, which enables the reprocessing of used nuclear fuel from VVER-1000, as well as various types of defective UNF.

In recent years large-scale reprocessing of several types of used nuclear fuel has been mastered at the plant using standard equipment.

In 2011, the pilot-scale reprocessing of RBMK-1000 fuel was undertaken, and in December 2014 the plant started the large-scale reprocessing of damaged used nuclear fuel of RBMK-1000 reactors.

In 2014, an experimental batch of uranium-beryllium UNF was successfully reprocessed. Now the plant is ready to start the large-scale reprocessing of this UNF type.

In 2014, a pilot reprocessing of MOX fuel from BN-600 reactor was undertaken. All processes, including dissolution, were carried out in the regular operation mode. The plant is now available for MOX fuel reprocessing. In recent years, several other types of commercial UNF have been reprocessed at the plant using standard equipment.

In 2015, a heavy-duty cutting machine AR-1000 will be manufactured. After its installation and commissioning, it is planned to commence the commercial reprocessing of both standard and defective VVER-1000 fuels in 2017.

A new building complex including UNF cutting and canister-loading section is under construction at RT-1 plant to enable treatment of long fuel assemblies from Beloyarsk NPP AMB reactor. It is expected to be in position to reprocess any kind of UNF from AMB reactors as early as in 2020. In the future, the complex will also be used for the treatment of used fuel assemblies from EGP-6 (Bilibino NPP) and KS-150 (shipped from A-1 NPP in Bohunice, Slovakia) reactors. Another area of activities is preparation of pyrophoric fuel for reprocessing. Pilot-scale reprocessing of uranium metal fuel was carried out in January 2014. Activities on modification of grinding and dissolving unit equipment are in progress providing possibility to control gas medium composition while handling pyrophoric fuel.

Reprocessing of pyrophoric mixed uranium-plutonium nitride fuel also requires upgrading of the gas purification system. After its modification in 2017 the plant will be available for the large-scale reprocessing of nitride fuel. In addition, RT-1 plant makes preparations for the large-scale reprocessing of sparingly-soluble fuels based on uranium and zirconium alloys, as well as plutonium dioxide and other uranium and plutonium compounds, used in research

reactors, ice-breaker propulsion reactors and fast reactors. Electrochemical treatment in nitric acid was chosen as a basic method of fuel dissolution.

According to the schedule, the plant will start experimental reprocessing of uranium-zirconium and other types of composite fuels, as well as plutonium materials in 2018 after the manufacture and installation of the equipment. In 1 or 2 years introduction of new technologies will convert RT-1 plant to multi-purpose reprocessing facility. There are three main areas of activities planned:

- The primary goal is to expand the range of services related to reprocessing of used fuel from VVER reactors from Russian and foreign nuclear power plants. We are ready to receive UNF from VVER-440 from our long-standing partners in Bulgaria, Hungary, Slovakia, Czech Republic, Ukraine, Armenia and other countries. As early as in 2017 RT-1 plant will be capable of reprocessing fuel from VVER-1000 reactors, including defective and non-conforming fuel;
- Another important area of activities is providing services related to reprocessing of any type of research reactor fuel;
- A distinctive feature of RT-1 plant will be the capability to reprocess any canister-loaded defective UNF from Russian and foreign reactors pre-prepared for the safe transport at the site of its storage.

By extending the range of used fuel, which can be reprocessed, RT-1 reprocessing capacity can reach 400 t/year, with the potential for further increase.

UNH is the traditional U-product of the plant. Currently, options are being considered to convert the process to produce uranium product in form of U_3O_8 with the enrichment 1 – 3 %. The main objective of plutonium manufacturing is to provide MOX fuel for fast and thermal reactors.

An extra task of the radiochemical production is manufacturing of semi-finished products for the isotope facility at the Mayak site. At present seven different elements are extracted at RT-1 for isotope fabrication.

Modernization of RT-1 plant is impracticable without technology optimization. The main goals of technology modification include:

- Considering the prospective increase in the plant production capacity, the bottleneck of the technology is the filtration unit. The change for membrane clarification of the feed solution has been planned to improve both productivity and quality of the filtration;
- The extraction process currently employed at RT-1 provides two refining cycles in the plutonium process line to produce two separate neptunium and plutonium products. The newly designed extraction technology is implemented through a single-cycle plutonium affinage followed by neptunium and technetium transfer to high-level and intermediate-level raffinate streams from uranium and plutonium process sections, respectively;
- The implementation of this technological solution will significantly reduce the operating costs of plutonium process section and decrease twofold the volume of intermediate-level liquid waste at this stage.

Another activity aimed at the liquid waste reduction is to implement technology of used extractant regeneration by steam distillation. The introduction of this technology will result in both complete elimination of liquid radioactive waste stream of this category and recycling of a valuable and expensive reagent. The implementation of these technologies is scheduled to be completed by 2017.

One of the priorities of RT-1 plant is the creation of new process lines for the reprocessing of radioactive waste. Liquid radioactive waste is planned to be treated at three complexes, depending on the category of waste. For the reprocessing of low-level radioactive waste a purification pilot plant has been established and successfully tested. Several technological solutions implemented at the pilot plant include ultrafiltration, reverse osmosis and sorption.

Currently, the facility is operated in a testing mode. By 2020 it is planned to establish and put into operation an industrial complex for liquid LLW purification. The construction of a complex for ILW cement solidification is in progress at the plant. The due date for construction completion is in 2015.

After commissioning of the cementation complex, discharges into special reservoir Karachai will be terminated.

For the treatment of high-level waste it is planned to finish the construction of electric verification furnace EP-500/5 in 2015, which will be the last furnace of this type.

Future processing of high-level waste is planned to be performed at a new verification complex. The complex will comprise two process lines. The first line will be equipped with a removable electric furnace of low capacity for handling routine waste resulting from NPP UNF reprocessing and for generating borosilicate glass. The second line will be equipped with a removable medium-capacity electric furnace generating aluminophosphate glass. The main objective of this process line is to solidify accumulated HLW, as well as waste resulted from reprocessing of non-standard fuel from Russian reactors.

The commissioning of these facilities will eliminate special reservoirs operated by Mayak from the scheme of liquid radwaste management. Conditioning and compacting of solid radioactive waste will be implemented at a solid radwaste management complex, which has been under construction since 2008. The main process stages of the complex are sorting and fragmentation, decontamination, incineration and compaction of all types of solid LLW and ILW.

4. CONCLUSIONS

(1) As a result of above-mentioned measures, by 2018 RT-1 plant will become a multi-purpose radiochemical facility capable of reprocessing the widest range of used nuclear fuel including defective and non-conforming fuel;
(2) RT-1 plant is ready to increase the reprocessing volume of VVER-440 used nuclear fuel from Russian and foreign nuclear power plants;
(3) In 2017 RT-1 plant will be ready for reprocessing of used nuclear fuel from VVER-1000 reactors.

SAFE SOLUTIONS FOR TRANSPORT AND DRY STORAGE OF DEFECTIVE FUEL RODS

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Abstract

Defective fuel management is a major challenge for nuclear operators when time comes up for finding a long-term solution for all their used fuel assemblies, defective included. Managing the defective fuel assemblies indeed requires implementation of specific technical solutions. Reprocessing the defective fuel assemblies is the most decisive way to get rid of them: there is then no need to assess the behaviour of this specific type of fuel assemblies in the long-term. Defective fuel assemblies are being today safely transported to and reprocessed in a treatment plant. When reprocessing is not the preferred option, the nuclear operator must choose between two dry storage solutions for the defective fuel assemblies: interim or long-term. This paper describes the existing solutions for transportation and dry storage of defective fuel assemblies. Transportation of defective fuel assemblies is today being performed. Various cask designs exist for that purpose and specific operations are implemented for preparing the defective fuel assemblies for transport. The advantage of such solutions is that they can support the operator in getting rid of the defective fuel assemblies once and for all. The paper presents the operational experience, the new licensing methodology relative to transport of defective fuel rods. Interim dry storage of defective fuel assemblies has been implemented for decades in the USA. Specific operations are performed for preparing the defective fuel assemblies which are then stored along with intact spent fuel assemblies in the dry storage cask. The advantage of such a solution is its short-term cost effectiveness. The paper presents the existing dry interim storage solutions for defective fuel assemblies, the associated operational experience. A long-term dry storage solution for defective fuel assemblies is being developed, compliant with direct disposal safety requirements. The solution is based on the dry encapsulation of each defective fuel rod into a fuel rod capsule, followed by its transfer into a capsule canister (like a fuel assembly skeleton), which is replacing a fuel assembly position of the transport and storage cask, and the associated specific licensing approach: this is the best available technology regarding long-term storage safety requirements. The paper presents this unique capsule technology, the transport and storage casks that can be used for transportation of defective fuel rods to the storage facility and especially focusses on the associated robust safety demonstrations that support the transport and storage licensing approach currently used in Europe.

1. INTRODUCTION

Today the organizations in charge of long-term management of all spent fuel assemblies are looking for a solution for managing the defective fuel assemblies along with intact spent fuel assemblies. In the same time, the nuclear industry all around the world is being confronted with the end of life of some reactors, and the need to evacuate from their pools all - defective as well as intact - used fuel assemblies. On the other hand, for power reactors in operation, the evacuation of the defective fuel assemblies is a way to optimize their spent fuel pool management.

Whereas the solutions for managing intact spent fuel assemblies are well-known, a specific technical solution must be implemented for managing the defective spent fuel assemblies as well.

Reprocessing the defective fuel assemblies is the most decisive way to get rid of them as there is then no need to assess the behaviour of this specific type of fuel assemblies in the long-term (Fig. 1). Reprocessing is a solution for avoiding need to assess behaviour of this specific type of fuel in the dry storage facility or final disposal. Defective used fuel assemblies have been and are being today safely reprocessed with specific administrative or technical measures for some of them.

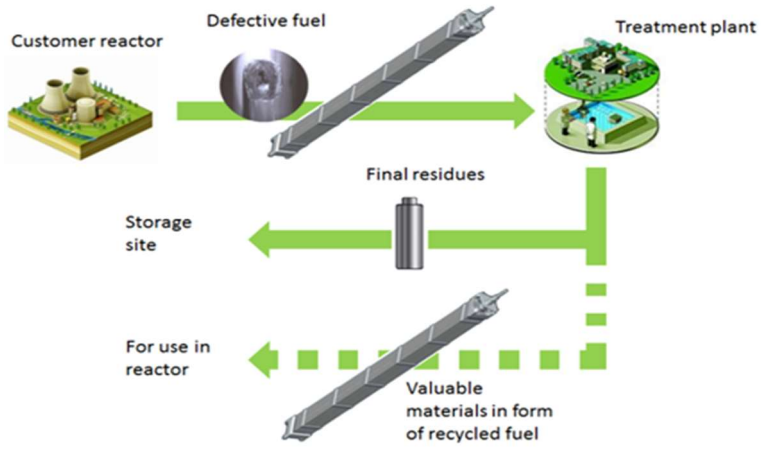

FIG. 1. Reprocessing of defective fuel (flow chart).

When reprocessing is not the preferred option, the nuclear operator must choose between two dry storage solutions for the defective fuel assemblies: interim or long-term. In order to meet this need, safe and sustainable solutions for transport and mid- and long-term dry storage of defective fuel assemblies have been developed and are implemented:

- Transportation of defective fuel assemblies is today safely being performed;
- Interim dry storage of defective fuel assemblies has been implemented for years in USA;
- A most efficient technology for long-term dry storage of defective fuel assemblies is being developed, based on robust safety demonstrations for getting the license of the transport and storage cask.

The purpose of this paper is to describe the existing solutions for transportation and dry storage of all types of defective fuel assemblies.

2. DEFINITIONS

According to Ref. [1], a spent fuel assembly is categorized as either defective or not, based on its ability to perform its designated functions without requiring the fuel assembly to be handled in a non-standard manner. The main safety functions to be performed by a spent nuclear fuel assembly are related to its ability to maintain confinement of the radioactive inventory, to remain subcritical and to maintain structural integrity.

Regarding confinement, defective fuel assemblies can be classified into three main categories:

Leaking fuel rod (Fig. 2): the cladding contains small holes or hairline cracks. No nuclear material can exit the cladding.

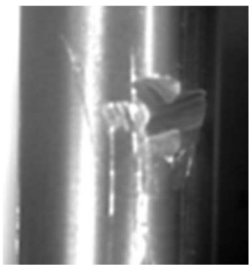

FIG. 2. Leaking fuel rod.

Damaged fuel rod, Fig. 3, the cladding contains bigger holes or cracks, the fuel rod is broken or the end plug is missing. Loss of nuclear material cannot be excluded.

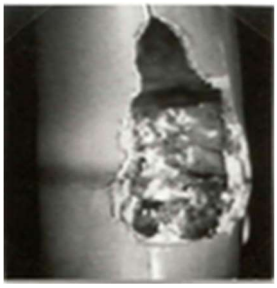

FIG. 3. Damaged fuel rod.

Fuel debris, Fig. 4, residual materials containing nuclear fuel e.g. nuclear pellets, fragments of pellets, or fuel powder.

FIG. 4. Fuel debris.

Various types of conditioning exist for defective fuel assemblies or rods:

- Quivers / Fuel rod canisters for storing several defective fuel rods together (open or close, all fuel rods in defined positions or inserted in tubes with end cap common to all tubes) — can be used for wet storage and also for transport;
- Bottles for wet storage and/or transport of one or several defective fuel assembly(ies);
- Fuel Rod Capsules for wet or dry storage or transport of fuel rods (one fuel rod capsule for one fuel rod) with screwed or welded closure (Figs 6–10);
- Capsule canisters for assembling fuel rod capsules — dimensions are similar to fuel assembly dimensions for easy handling (Fig. 12).

3. SOLUTIONS FOR TRANSPORTATION OF DEFECTIVE FUEL ASSEMBLIES

3.1. Solutions

For transporting the defective used fuel assemblies/rods to centralized storage site or reprocessing plant, AREVA proposes a comprehensive range of transportation solutions.

The MP197HB transport cask (Fig. 5) is licensed in USA for transporting all types of defective fuel assemblies (regarding containment issue).

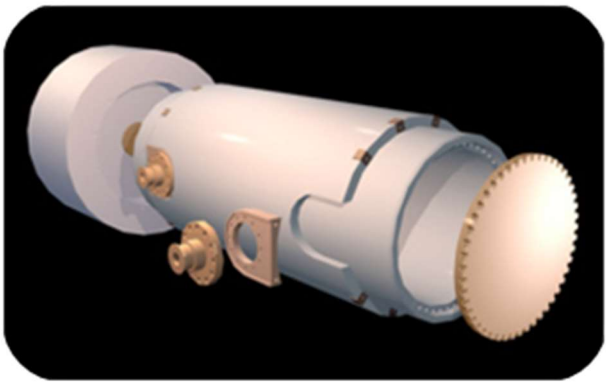

FIG. 5. MP197HB transport cask.

The following casks are licensed for transportation of leaking fuel assemblies:

- TN®12 and TN®13, licensed in France, for EDF needs;
- TN®17/2, licensed in France and Sweden, for SKB needs;
- TN®117, licensed in France and Italy, for SOGIN needs.

For transportation of damaged fuel rods and fuel debris, European regulation requires the fuel rods or cartridges with debris to be inserted into screwed fuel rod capsules (Fig. 6), assembled in capsule canisters to be loaded in the transport cask after license extension has been obtained.

Depending on Safety Authority requirements, the screwed fuel rod capsule can be either particle-tight only, or particle-tight and gas-tight.

FIG. 6. Screwed particle-tight and gas-tight fuel rod capsule for transport of damaged fuel rods and fuel debris.

The advantage of such transport solutions is that they support the operator in getting rid of the defective fuel assemblies once and for all: they are evacuated from the power reactor site towards off-site centralized storage or reprocessing plant.

3.2. Safety assessment

- Confinement

The function of the cask is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transport. In accident conditions, even considering the rupture of all of the fuel cladding, the internal pressure never exceeds the maximum design pressure used to justify the structural strength of the packaging, so that there is no impact on the containment.

- Radiolysis

The hydrogen explosion risk safety case is a French Safety Authority requirement.

The phenomenon of radiolysis due to the potential presence of residual water in the leaking fuel rod and the subsequent risk of hydrogen explosion have been evaluated on the basis of extrapolation from measurements of the hydrogen rate in the cavity of TN®12 packages, loaded with leaking and intact fuel assemblies. These gaseous samples and their analyses were performed after shipment before unloading at the AREVA La Hague Reprocessing Plant.

This evaluation leads to limit the transport duration from 1 year to several months or to reduce the number of potential leaking fuel rods in order to maintain the maximum potential hydrogen rate in the cavity below the Lower Limit of Inflammability (LLI).

- Criticality

The basket has been designed to assure an ample criticality safety margin. The intact fuel assemblies and the defective fuel assemblies are assumed to stay within their basket compartments based on basket geometry during normal operations.

In case of a severe accident, the safety demonstrations consider that all the content (constituted by the basket and intact fuel assemblies and defective fuel assemblies) is ruined and mixed in the centre of the cask with a limited amount of water (due to the fact that the cask is considered as a double water leak tight barriers package). This quantity of water in the cavity in accident condition is very conservative and results from the potential ingress of water consequently the penetrating test at 15 m under water surface.

3.3. Operational experience

For EDF needs, AREVA has performed many shipments of leaking fuel assemblies. In addition, AREVA has a strong international experience in transportation of leaking fuel assemblies. Shipments were performed from Germany, Switzerland, Belgium, Japan… to the La Hague plant in France.

Screwed particle-tight and gas-tight capsules have been provided for conditioning defective fuel rods in two plants in Germany, one plant in Netherlands and one plant in Switzerland. They were then put together in a capsule canister. The TN®17/2 cask (Fig. 7) was used for transporting these encapsulated defective fuel rods from Germany to the AREVA La Hague plant for treatment.

FIG. 7. TN®17/2 transport cask.

The TN®12/2 cask has been used for shipments of damaged fuel rods from a French plant and a Swiss plant to La Hague.

Screwed particle-tight but not gas-tight capsules have been provided by AREVA at a German plant and a Finnish plant for wet storage purpose. This design is available for transport of defective fuel rods, if the French Safety Authority so requires.

4. SOLUTIONS FOR INTERIM DRY STORAGE OF DEFECTIVE FUEL ASSEMBLIES

4.1. Solutions

Considering interim dry storage of defective fuel assemblies, the following solutions are licensed by the US NRC:

- Leaking fuel assemblies (usually referred to as 'damaged' in the US) shall be stored in a canister's usual fuel assembly compartment with additional end caps that have multiple holes in order to permit unrestricted flooding and draining; there are designated basket locations for the storage of leaking fuel assemblies;
- Damaged fuel assemblies and fuel debris (all together referred to as 'failed' in the US) shall be stored in a failed fuel can that is placed in a canister's compartment which is slightly larger than a fuel assembly compartment; there are designated basket locations for the storage of 'failed' fuel assemblies.

AREVA developed and has been implementing for several decades, solutions which are compliant with the NRC requirements. The advantage of such solutions is their short-term cost effectiveness.

For storage of leaking (called 'damaged' in the US) fuel assemblies, AREVA provides the following NUHOMS® dry shielded canisters:

- 32PTH, 37PTH, 24PT4 for PWR fuel assemblies;
- 61BT/BTH, 69BTH for BWR fuel assemblies; and
- 37PTH and 61BT/BTH canisters are presented in Fig. 8.

FIG. 8. 37PTH and 61BT/BTH, dry shielded canisters licensed for interim dry storage of leaking fuel assemblies.

For storage of "failed" (so-called in the US) fuel assemblies, AREVA provides the NUHOMS® 24PTHF canister (Fig. 9):

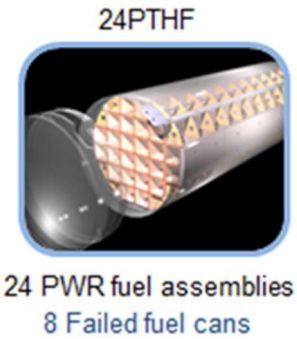

FIG. 9. 24PTHF, dry shielded canister licensed for interim dry storage of failed fuel.

4.2. Safety assessment

- Confinement

For intact fuel, the fuel cladding is the first barrier for confinement of radioactive materials. For leaking fuel assemblies, top and bottom caps are provided to contain fuel debris such as broken rods, loose pellets and/or pieces of cladding in the fuel compartment. For damaged fuel, failed fuel cans are designed to contain fuel material for avoiding dispersion in the canister cavity.

The canister is designed to provide confinement of all radionuclides under normal and accident conditions postulated to occur during transportation and storage. The NUHOMS® canister is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.

- Radiolysis

The top and bottom caps fitted on the compartments dedicated to leaking fuel and the failed fuel cans for damaged fuel have been designed to have no influence on the drying performance of the cavity (i.e. no water trap nor water surface retention). Moreover the loading operation of the cask with defective fuel remains the same as for the loading of intact fuel (same draining, drying and leak tight tests).

After the drying, the expected absence of residual water ensures there is no impact on the safety evaluation.

- Criticality

The basket has been designed to assure an ample criticality safety margin under the conditions of fresh fuel in a canister flooded with fresh water.

The intact fuel assemblies and the defective fuel assemblies are assumed to stay within their basket compartments based on the canister and basket geometry.

The NUHOMS® canister is analyzed for additional considerations arising from defective fuel mechanical uncertainties after an accident. In case of a severe accident, rod breakage may be postulated to occur in rods with known pre-existing gross cladding failure. This may result in a more reactive configuration than that generated with intact fuel therefore a specification limiting the number of known rods with gross cladding damage per fuel assembly is established.

- Shielding

Shielding evaluation takes into account the following considerations:

  - Specific loading plan with the defective fuel in dedicated basket compartments;
  - Some rods coming from each of the potential defective fuel assemblies in the canister are assumed to collect against the canister shell at the bottom of the canister.

4.3. Operational experience

Hundreds of defective fuel assemblies are currently stored in AREVA interim dry storage casks in the USA.

5. SOLUTION FOR LONG-TERM DRY STORAGE OF DEFECTIVE FUEL ASSEMBLIES AND RODS

5.1. Solution

AREVA is developing a new, long-term oriented solution for the dry storage of defective fuel rods: the concept consists of welded fuel rod capsules (Fig. 10), assembled in a capsule canister (Fig. 12) that can be loaded in an AREVA transport and storage cask (Fig. 13) for transportation to and/or storage in a long term dry storage facility. This welded fuel rod capsule concept is designed for final disposal of defective fuel rods and based on an existing and experienced screwed fuel rod capsule concept that has been already in use for more than 10 years for wet storage and/or transport purposes.

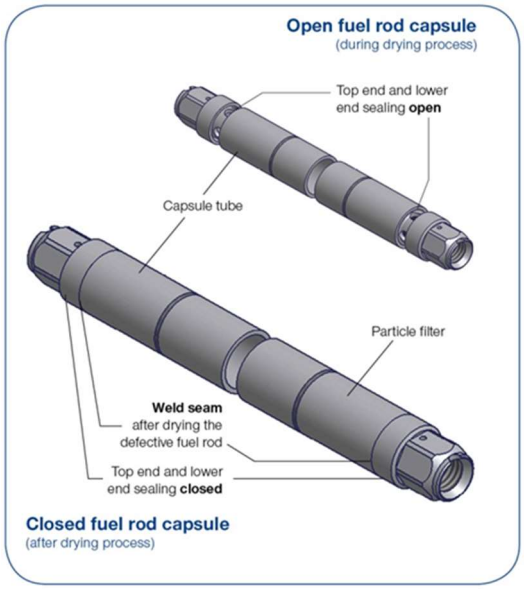

FIG. 10. Welded fuel rod capsule for long-term dry storage of a defective fuel rod.

This fuel rod capsule takes the function of a barrier to assure the safe enclosure of radioactive material and is therefore at least equivalent to an intact fuel rod cladding with tight welded end plugs. Two different types of fuel rod capsule are being developed that can be used for any kind of PWR and BWR fuel rods.

The principles of the process are the following:

- Preparation and pre-treatment of the defective fuel rod / fuel debris;
- Transfer of the defective fuel rod (or cartridge for debris) into the fuel rod capsule in the pool;
- Dewatering and drying of the defective fuel rod (or cartridge) within the fuel rod capsule;
- Gas-tight enclosure of the fuel rod capsule by welding;
- Non-destructive testing of the welds;
- Transfer of the fuel rod capsule into the capsule canister;
- Re-attachment of the top end piece of the capsule canister;
- Handling and transfer of the loaded capsule canister to the AREVA cask;
- Storage in the same AREVA cask of the loaded capsule canister along with intact spent fuel assemblies.

All these operations are managed in the spent fuel pool (Fig. 11) which allows to significantly reduce the dose rate compared to operations performed outside of pool. Should the nuclear operator so require, the operations can also be performed in a hot cell facility.

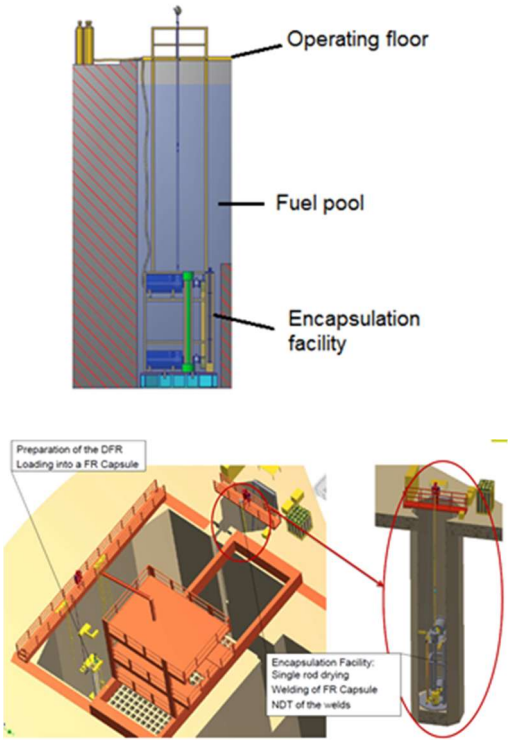

FIG. 11. Encapsulation facility positioned in the spent fuel pool.

This encapsulation technology uses a drying process which allows a physical and reliable demonstration of the remaining amount of residual water within the defective fuel rod.

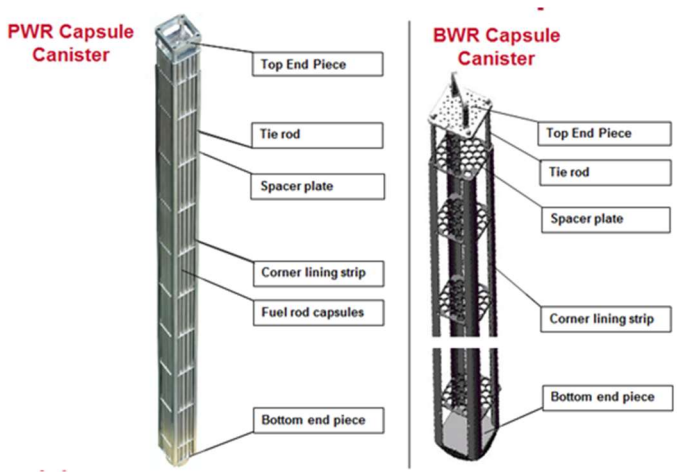

FIG. 12. PWR and BWR capsule canisters.

All the equipment and processes described here above are covered by AREVA patents.

Listed below are the main technical topics for a transport and storage cask licensing point of view:

- Qualification of the encapsulation drying process, and estimation of the residual water content;
- Mechanical behaviour of a capsule canister, and comparison with an intact spent fuel assembly;
- Qualification of the welding process of the fuel rod capsule and non-destructive testing of the weld;
- Loading capacity: a compromise between the maximal possible number of fuel rod capsules to be loaded in a capsule canister resp. cask and the easiest way to get the license;
- The cask license shall cover:
  - Different types and characteristics of defective fuel rods in capsule canisters;
  - Mix loading of intact used fuel assemblies and defective spent fuel rods in capsule canisters.

FIG. 13. TN®24E (left) and TN®24SH (right) casks that can be used for transport and dry storage of defective fuel rods loaded into fuel rod capsules.

5.2. Safety assessment

- Confinement

For intact fuel, the fuel cladding is the first barrier for confinement of radioactive materials. For defective fuel rods in capsule canister, the capsule provides the first barrier for confinement of fuel debris such as broken rods, loose pellets and / or pieces of cladding.

The function of the cask is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The cask is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.

- Radiolysis

The capsule canister has been designed to have no influence on the drying performance of the cavity (i.e. no water trap nor water surface retention). Moreover the loading operation of the cask with capsule canister remains the same as for the loading of intact fuel (same draining, drying and leaktight tests).

The very low amount of residual water, physically demonstrated by the drying process, ensures there is no impact on the safety evaluation.

- Criticality

The basket has been designed to assure an ample criticality safety margin under the conditions of fresh fuel in a cask. The intact fuel assemblies and the defective fuel rods in capsule canisters are assumed to stay within their basket compartments based on basket geometry.

In case of a severe accident, the safety demonstrations consider that all the content (constituted by the basket and intact fuel assemblies and the defective fuel rods in capsule canisters) is ruined and mixed in the centre of the cask with a limited amount of water equal to 1 l (due to the fact that the cask is considered as a double water leak tight barriers package). This quantity of water in the cavity in accident condition is very conservative and results from the potential ingress of water consequently the penetrating test at 15 m under water surface and the failure of all the capsules releasing water remaining in each defective fuel rod after drying operations (which is physically demonstrated to be very low).

- Shielding

The shielding evaluation of a cask loaded with capsule canister system will generate less dose rate compared to a cask loaded with intact fuel assemblies due to the number of capsules which is less than the number of rods in intact fuel assemblies.

To conclude, the technology developed by AREVA is the most efficient in terms of residual water in the defective fuel rod and allows to obtain a robust safety demonstration for the transport license of the AREVA transport cask and the storage license in the long term dry storage facility: it is the best available technology regarding safety requirements for dry storage.

5. CONCLUSIONS

Today nuclear utilities and organizations in charge of back-end issues have to assess their spent fuel assemblies management policy for all types of fuel assemblies, defective included. Even when implementing new build projects, the stakeholders are asked by the Safety Authorities to present a comprehensive solution for final management of intact and defective fuel assemblies. The management of defective fuel is therefore becoming a matter of concern for all nuclear operators. AREVA is continuously developing technologies that meet the customers' needs and the utmost safety requirements.

AREVA proposes qualified and proven solutions for transport and dry storage of defective fuel assemblies and rods:

- Robust and experienced solutions for transport of defective fuel assemblies;
- Cost-effective solutions for interim dry storage of defective fuel assemblies;
- The best available technology for long-term dry storage of defective fuel assemblies, deriving from existing and experienced fuel rod encapsulation technology.

All that said, reprocessing remains the most decisive way to get rid of defective fuel in a risk reduction approach as this specific type of fuel is replaced by final residues in the dry storage facility or geological disposal, residues packages being designed for long-term storage or disposal.

REFERENCE

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Damaged Spent Nuclear Fuel, IAEA NE-Series No. NF-T-3.6, IAEA, Vienna (2009).

RUSSIAN EXPERIENCE AND PROPOSALS ON MANAGEMENT OF NON-CONFORMING SNF OF RBMK REACTORS

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Abstract

The RBMK-1000 power reactors have been operated at Leningrad, Kursk and Smolensk NPPs in the European part of Russia. The yearly fuel discharge from the RBMK-1000 reactors makes up 3500 SFAs placed in reactor cooling pools and separate spent fuel wet storage facilities at NPPs. Since 2012 the RBMK-1000 SNF has been transited to safer dry storage in a centralized dry storage facility at the Mining and Chemical Combine in Kraskoyarsk region, Siberia. According to the approved technology leak-tight SFAs with sound grid skeletons (hereinafter, the conforming SFAs) are subject to dry storage at NPP storage facilities and transfer to the centralized storage facility. Transition of defective and leaky spent fuel (hereinafter, the non-conforming SFAs) to dry storage in NPP storage facilities and their transfer to the centralized storage facility are not provided. One of the ways to manage the non-conforming RBMK-1000 SFAs is reprocessing at Mayak PA, separation of uranium, plutonium, neptunium, and vitrification of HLW. In 2011, a pilot batch of leaky SNF was removed from the Leningrad NPP to try out and verify engineering solutions. In 2014, a batch of leak-tight non-conforming SNF was shipped from the Leningrad NPP to a reprocessing plant. The paper addresses Russian experience and proposals on management of non-conforming SNF of RBMK-1000 reactors.

1. INTRODUCTION

The RBMK-1000 reactor is a one-circuit channel-type graphite-moderated power reactor designed in 1960s. The RBMK-1000 power reactors have been operated at Leningrad, Kursk and Smolensk NPPs in the European part of Russia. A fuel assembly of RBMK-1000 (Fig. 1) consists of two fuel rods bundles, each is about 3.6 m long and 79 mm in diameter. They are secured one above another on a bearing rod with a pintle. The total length of the fuel assembly is about 10 m. Each bundle contains 18 fuel rods, which are installed in a skeleton of ten spacer grids (SG) and one end grid secured on the tube. Material of the fuel rod cladding is E-110 zirconium alloy, material of the fuel pellets is sintered uranium dioxide. Fuel assemblies are installed into vertical reactor channels. The fuel is reloaded without reactor shutdown. The fuel assemblies discharged from the RBMK-1000 reactors are stored in the reactor cooling pools. After three-year cooling period they are transported to a separately located wet storage at NPP using an on-site container. Each SFA is put into a canister in the NPP storage facility to minimize the radioactive contamination of the pool water.

Transition of the RBMK-1000 SNF to dry storage in the centralized storage facility is planned to improve safety. In late 2011, a centralized storage facility in Krasnoyarsk region (Siberia) and a complex for SNF transition to dry storage were commissioned in the facility of Leningrad NPP. The similar complex at Kursk NPP was commissioned in 2014, the complex

at Smolensk NPP is being constructed. The design-basis efficiency of each NPP complex is 3600 SFA per year.

According to the approved technology leak-tight SFAs with sound grid skeletons (conforming SFAs) are subject to transition to dry storage and transfer to the centralized storage facility. Technology for transition to dry storage includes leak tests and visual inspection of SFAs, cutting of SFA into bundles, placing of bundles into ampules, loading of ampules with bundles into a cask, drying of SNF in the cask, and preparation of the cask for shipment. SFAs are cut in a hot cell of the NPP complex. TUK-109 casks are used to transport SNF to the centralized storage facility. TUK-109 packaging contains an energy absorbing container (EAC), a metal concrete cask with two lids, a basket and 144 ampules for the fuel rod bundles. The long-term storage of SNF in the metal concrete casks (for 50 years) is possible. The cask delivered to the centralized storage facility is opened, the basket with SNF is transferred to the hot cell, where the ampules with the fuel rod bundles are reloaded into 30 seat canisters, and then the canisters are sealed with welding and put into storage cells. The transition of defective and leaky spent fuel (non-conforming SFAs) to dry storage in the NPP complexes, their transfer and receipt at the centralized storage facility are not provided.

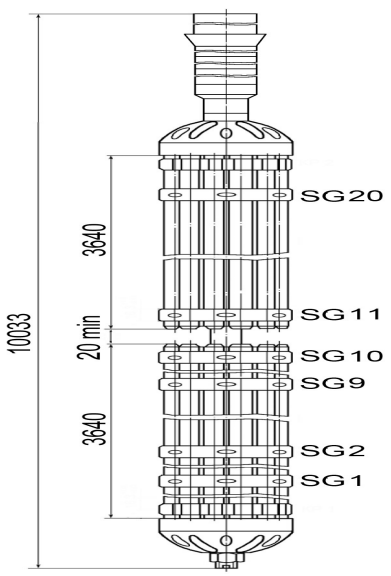
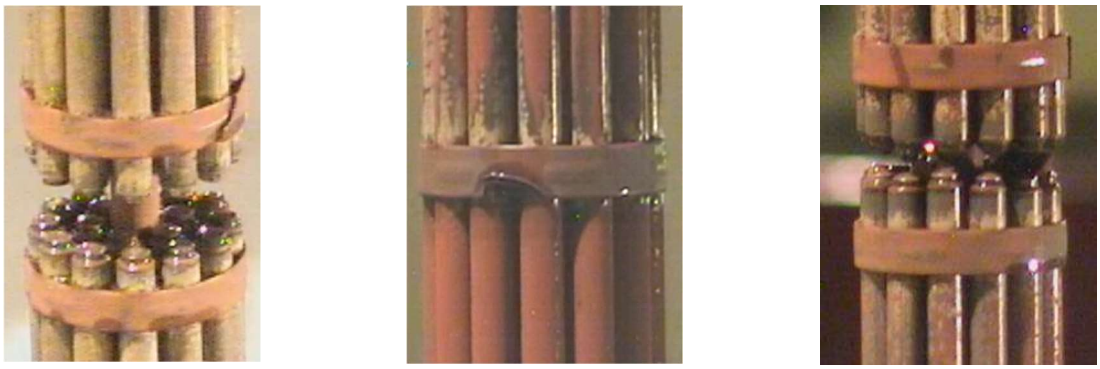


FIG. 1. RBMK-1000 fuel assembly.

2. NON-CONFORMING SFA

The conforming SFAs are selected for transition to dry storage in compliance with a special procedure. The non-conforming SFAs are leaky ones and those tight ones but with the following damages:

- More than two destroyed spacer grids in succession;
- Damaged or destroyed spacer grid 10 or spacer grid 11 located near the gap between the fuel rod bundles (Fig. 2a);
- Local increase of the SFA diameter more than 87 mm induced by the damaged spacer grid(s) (Fig. 2b);
- A gap between the fuel rod bundles is less than 9 mm (Fig. 2c).



a *b* *c*

FIG. 2. Damaged RBMK-1000 SFA: crack on the spacer grid (a); tear on the spacer grid (b); and a gap between the fuel rod bundles less than 9 mm (c).

Leaky SFAs are detected by measuring the specific activity of the water in their storage canisters. All damages to the leak-tight SFAs structure are detected visually.

About 3500 leaky SFAs are stored at three NPPs with the RBMK-1000 reactors. Total amount of damaged leak-tight SFAs is unknown because the SFAs are examined directly before cutting. The largest statistics on damaged SFAs is at the Leningrad NPP. 24% of leak-tight SFAs have been discarded because of their damaged structure during three years of operation of the Leningrad NPP complex. The shares of leak-tight non-conforming SFAs with different damages are shown in Fig. 3.

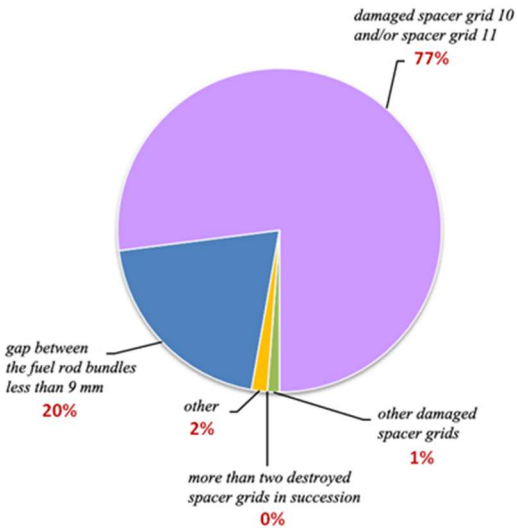


FIG. 3. Types of damage to non-conforming SFAs.

Beside the SFAs with minor damages mentioned above, there are severely damaged SFAs (bent, deformed, partially destroyed) and pilot SFAs with a peculiar structure. Due to changes in their form and cross-section, the severely damaged SFAs cannot be loaded into a basket of

the on-site container using a standard transfer cask and transferred to the NPP storage facility. Transition of the pilot SFAs with a peculiar structure to dry storage would require their cutting across the rods, and that cannot be done at the NPP complexes.

3. REMOVAL AND REPROCESSING OF THE PILOT BATCH OF LEAKY SNF

Before commencing the work on transition of the RBMK-1000 SNF to dry storage (2012) it was expected that the amount of non-conforming SFAs would make up 3-5% and the most SFAs would be leaky. One of the safe ways to manage the non-conforming RBMK-1000 SFAs was reprocessing with separation of uranium, plutonium, neptunium, and vitrification of HLW. In 2010-2011, the feasibility of delivery and reprocessing of leaky RBMK-1000 SNF was analyzed and a pilot batch of SNF was delivered and reprocessed at Mayak PA to confirm practicability. The pilot batch consisted of eight leaky SFAs.

Reprocessing of the RBMK-1000 SNF and the use of obtained regenerated uranium are considered unpractical due to the small content of Uranium-235. But the most of leaky SFAs were discharged from the reactor ahead of schedule and had little burn-up. The fuel and structural materials of the RBMK-1000 fuel assemblies are similar to those of VVER-440 which are routine reprocessed at the RT-1 plant, thus, no technical problems regarding the RBMK-1000 SNF radiochemical reprocessing were expected. So, the main task was to develop a technology for safe handling of leaky SNF that could be used both at the NPP and the RT-1 plant.

In 2010-2011, the RT-1 plant had no equipment for handling of TUK-109 casks; the complexes for SNF transition to dry storage at the NPPs were not commissioned and were not intended to cut leaky SFAs. Thus, it was planned to ship SNF in a TUK-11 cask with a modified Type 12 basket (Fig. 4) and to cut SFAs from the pilot batch in a hot cell of Unit 2 of the Leningrad NPP. Earlier this hot cell and the cask had been used to ship the fuel rod bundles to the Research Institute of Atomic Reactors, Dimitrovgrad, Russian Federation for examinations. Some modifications to the procedure for SNF loading in a cask and some new equipment were required to deliver the leaky SNF to the RT-1 plant.

The concept of leaky SNF handling was to use the leak-tight ampoules to transport the fuel rod bundles in a cask, to perform transport operations and temporary store the SNF at the NPP and at the RT-1 plant. The design of the leak-tight ampoule is shown in Fig. 4 e.

The use of leak-tight ampoule is induced by the regulations restricting the release of activity and the fuel into the cooling pool water during the temporary storage of leaky SNF. However, use of air-tight ampoules for leaky SNF removed from the wet storage brings about the problem of accumulation of radioactive hydrogen which limits the permissible storage period. SNF drying can solve this problem, but it cannot be used because of the technical difficulties related to the deployment of additional equipment in the hot cell, a new complicated design of ampoule and increased cost for the SNF preparation for shipment. Conservative calculations demonstrated that the concentration of radiolytic hydrogen in tight ampoules with leaky SNF during their storage for six months would not reach the explosive value. Experimental study of the hydrogen accumulation in a tight ampoule with leaky RBMK-1000 SNF demonstrated the significantly less rate of the hydrogen accumulation compared with the calculations. Thus, it was decided to seal the SNF-containing ampoules without drying or blowing with the inert gas.

a *b* *c* *d* *e*

FIG. 4. Transport Packaging (a) for shipment of fuel rod bundles of leaky RBMK-1000 SFAs and its components: inner plate (b), basket (c), insert (d), ampoule (e).

In process of SNF handling at the RT-1 plant it was planned not to withdraw the fuel rod bundles from the ampoules for further chopping and radiochemical reprocessing in order to prevent any spillage of the fuel in the hot cell and the removal of the spilled fuel from the ampoules. Consequently, the ampoule design shall allow chopping the fuel rod bundle and the ampoule together. On the other hand, the ampoule shall be strong enough to ensure the leak tightness under normal and accident conditions of shipment in a cask and during handling at the Leningrad NPP and RT-1 plant. This problem was solved by using 2 mm-thick tubes in the ampoule design and also by placing the tube-like inserts into the transport basket cells.

SNF-containing ampoules were loaded into a transport packaging using a transfer cask and two steel plates with holes and plugs (Fig. 5). The external plate was installed on the transport cask to protect the personnel from the radiation during loading of the SNF-containing ampoules. The internal plate was installed in the cask before to protect the personnel from the radiation during the removal of the external plate from the cask after the SNF loading and during the lid installation. The internal plate was transported in a cask and remotely unloaded at the RT-1 plant before withdrawal of the SNF-containing basket.

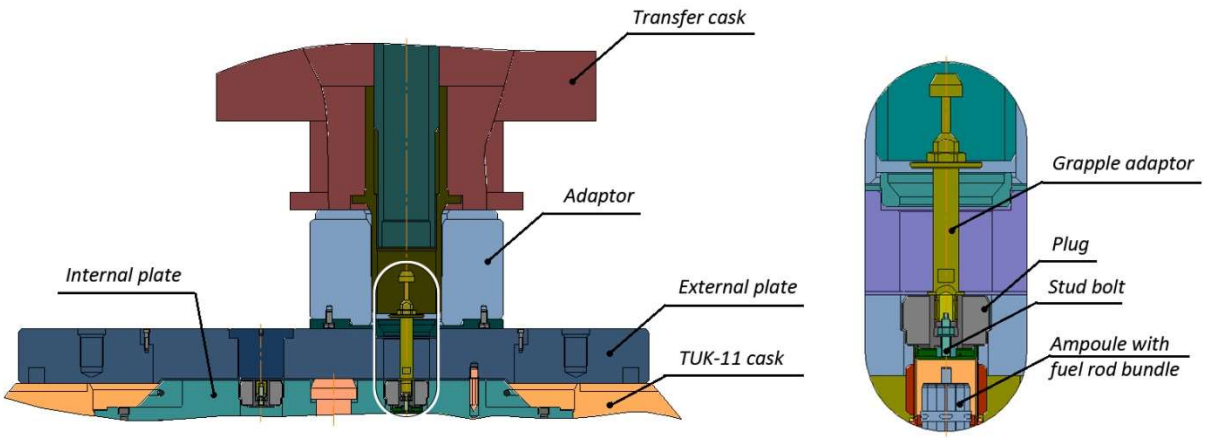


FIG.5. Loading of ampoules with fuel rod bundles in a cask.

Also the additional equipment was designed and fabricated. It included adaptors, grapples, tools and other equipment for loading the fuel rod bundles in the ampoules in the hot cell at the NPP, for loading the ampoules with the fuel rod bundles in a cask, for handling the casks and the ampoules at the RT-1 plant. Safety at all stages of SNF handling was justified, the safety assessment reports were prepared, the combined trials of the equipment were held, the certificate of approval for packaging design and shipment was obtained as well as the licenses authorizing the work.

Eight leaky SFAs formed a pilot batch. The cut, shipment and reprocessing of the SFA pilot batch took place in late 2011.

The analysis of capabilities of the NPP and the RT-1 plant, performed in 2010-2011, demonstrated that the developed procedure can be used to deliver for reprocessing only a small amount of leaky SFAs from Unit 2 of the Leningrad NPP. The procedure for transport and handling at of RT-1 plant and the equipment of NPP complexes shall be modified to remove the SNF for reprocessing on a regular basis.

4. REMOVAL AND REPROCESSING OF THE LEAK-TIGHT DAMAGED SNF BATCH

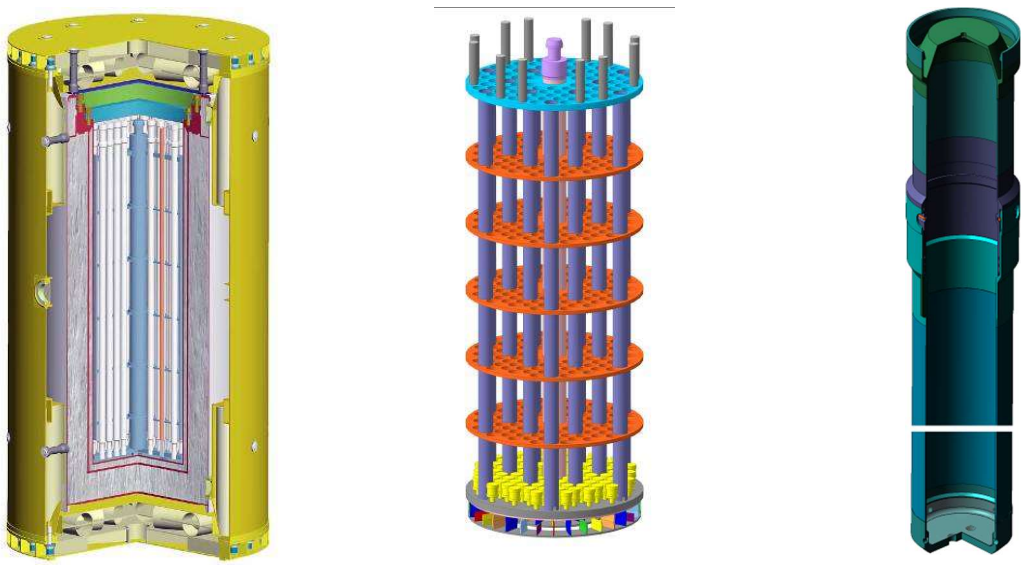
The concept was to use non-tight ampoules with the extended diameter and a TUK-109 cask to ship the fuel rod bundles of damaged SFAs (Fig. 6). TUK-109 cask was used for SNF standard loading at the Leningrad NPP but a lot of additional equipment had to be purchased by the RT-1 plant.

A special ampoule for the fuel rod bundles from the leak-tight damaged SFAs was designed. Ampoule design allows:

- Performance of transport and handling operations in the hot cell of the SNF storage at the Leningrad NPP;
- SNF drying in TUK-109 cask following the existing procedure;
- Penetration of water into the ampoules when they are stored in the pool at the RT-1 plant;

- Draining the ampoules when they are removed from the pool for further cutting at the RT-1 plant;
- Performance of transport and handling operations in the hot cell for SNF preparation for cutting at the RT-1 plant;
- Chopping the ampoules together with a fuel rod bundle in process of their reprocessing at the RT-1 plant.

Special baskets were designed and fabricated to arrange the ampoules in the TUK-109 cask. Due to the extended diameter of the ampoules, the basket capacity was reduced up to 102 ampoules comparing with a standard one. The safety was analyzed and the certificate of approval for the packaging design and shipment was obtained.



a *b* *c*

FIG. 6. Transport packaging (a) for shipment of fuel rod bundles of leak-tight non-conforming RBMK-1000 SFAs and its components: basket (b), ampoule (c).

Since the procedure for SNF unloading from the cask and for SNF transfer for chopping to the RT-1 plant (Fig. 7) was complicated in comparison to the standard one, the existing equipment had to be modified and the additional equipment had to be procured.

The hydraulic 1100 t crane and special yokes were procured to unload the cask with the EAC out of the railcar, to put it upright and to remove the EAC in a cask handling facility. A special Type TP9-1 railcar was procured to transport the cask upright without the EAC from the cask handling facility to the SNF storage facility and back.

The load capacity of cranes used to withdraw the SNF-containing basket out of the cask and place it in a pool was increased. The hoist trolleys used to transport the SNF-containing basket in and out of the pool were modified in order to increase the height of the basket transport way. The load capacity of the hoist trolley used to transfer the SNF-containing

basket in a pool was increased. An area for reloading of some ampoules from one basket to another was arranged in the pool in order to reduce the mass of the SNF-containing basket and to exclude any modifications of other hoisting equipment for transfer of SNF-containing basket to the hot cell for SNF preparation for chopping.

Also, the various auxiliary equipment for handling the cask, the basket and the ampoules at the RT-1 plant (adaptors, grapples, tools, etc.) was designed and fabricated.

The equipment for handling the ampoules of a new type was designed, fabricated and installed in the hot cell of the SNF storage facility at the Leningrad NPP.

The safety of non-conforming SFA handling at the Leningrad NPP and the RT-1 plant was justified, the safety assessment reports were prepared, the combined trials of the equipment of the hot cell of the SNF storage facility at the Leningrad NPP and the RT-1 plant equipment were performed, and the licenses for work were obtained.

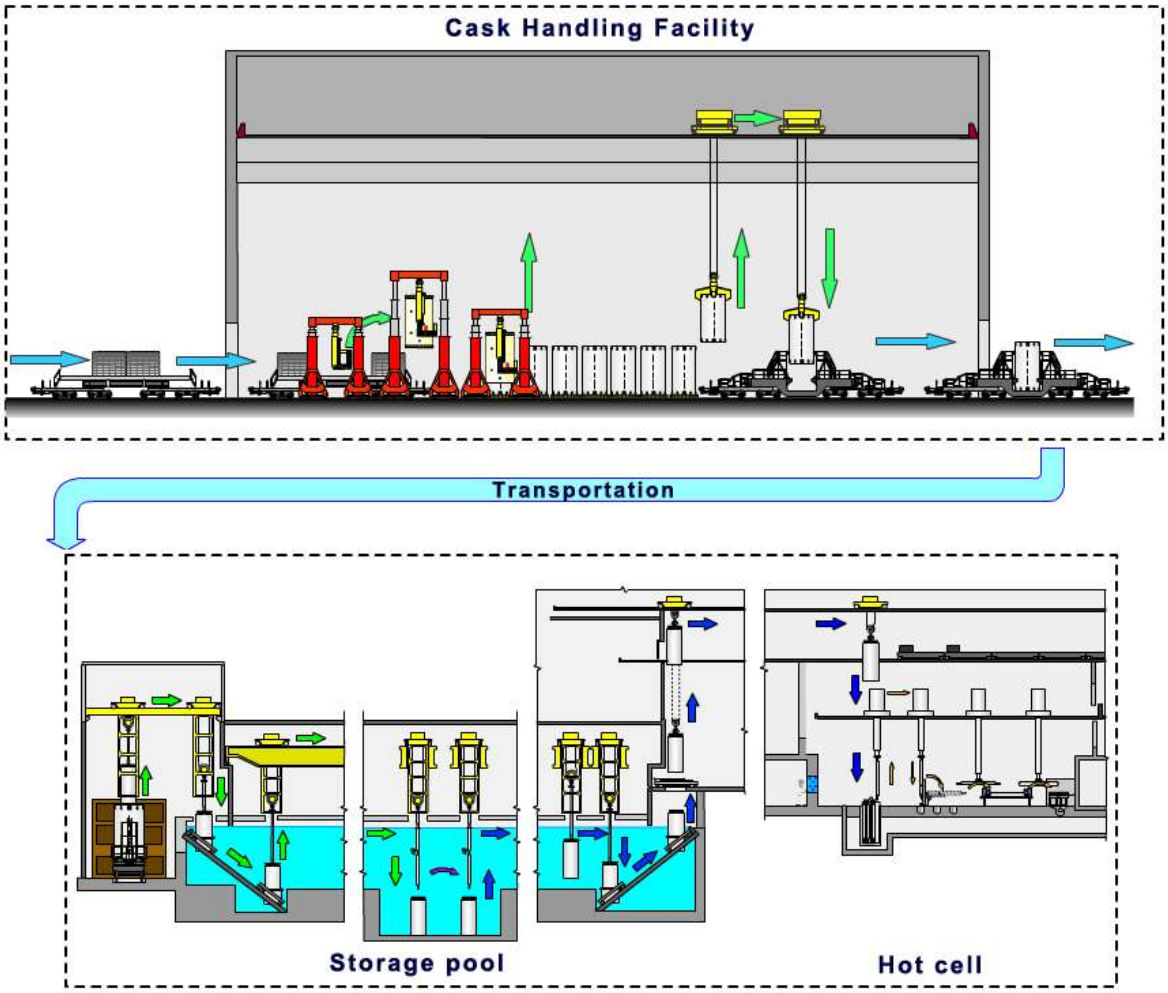


FIG. 7. Transport and process scheme of RBMK-1000 SNF handling at RT-1 plant.

204 leak tight SFAs with damaged spacer grids were selected for cutting and shipment to the RT-1 plant, and none of them should have destroyed spacer grid 11 or any damage of the

spacer grid 10. In October and November 2014, all selected non-conforming SFAs were cut and loaded into casks. In late November 2014 four casks loaded with non-conforming SNF were shipped to the RT-1 plant for reprocessing.

5. PROSPECTS

The share of non-conforming SFAs located in the storage facility at the Leningrad NPP was significantly larger than it was previously assumed. And the most of them were leak tight SFAs with damaged structures. One of the ways for handling leak tight non-conforming SFAs is to repair their skeletons in order to perform further SFA cutting, loading of the fuel rod bundles into standard ampoules, transit SNF to dry storage and arrange it in the centralized storage facility. The repair is to be performed in a hot cell of the NPP complex. In process of the SFA skeleton repair any tear on the spacer grid should be removed through the polishing, for instance, and the clamps should be installed on the SFAs to compensate the damage of the spacer grid. In process of developing the procedure for SFA repair the final cross-sectional dimensions of the fuel rod bundles shall be defined, the mechanical impact on the SFA grid skeleton and the fuel rods shall be assessed, the safe SFA cutting and the long-term storage of the fuel rod bundles fastened with the clamps shall be analyzed. Each SFA that has a gap between the fuel rod bundles less than 9 mm is to be disassembled through the removal of the fuel rod bundles from the bearing rod one by one.

In case the repair of the SFA grid skeleton is too complicated or failed, the ampoules with the extended diameter are to be used for loading with the fuel rod bundles, further transit to dry storage and transfer to the centralized storage facility or to the RT-1 plant for reprocessing. The use of ampoules with the extended diameter in the centralized storage will result in reduced amount of the fuel rod bundles in the canister and in increased amount of the cells necessary for their long-term storage. But the centralized storage facility has a spare capacity.

The leaky, severely damaged and pilot SFAs are to be reprocessed at the RT-1 plant. Leaky SFAs found in the NPP storage facilities and transferred to the storage facilities from the reactor units are to be prepared in the hot cells of NPP complexes. The technology shall include the use of tight ampoules and the collection of all fuel spillages as routine operation. In case the whole process cycle from the SNF sealing in the ampoules to its chopping will be too long, it is necessary to include the SNF drying in the ampoules in order to ensure the safe shipment and temporary storage. The activities on handling leaky SFAs will require the significant modification of the hot cells at the NPP complexes. This modification can be performed only after the removal of the most leak tight SFAs.

Special technology for handling severely damaged and pilot SFAs at the reactor units is to be developed to prepare these SFAs for the removal. It will include the SFA cutting in the reactor pool under the water layer. Since it is almost impossible to ensure the quick cutting and preparation of the severely damaged SFAs for the removal due to a variety of SFA damages, the handling procedure shall include the SNF drying and temporary storage. All activities on handling the severely damaged and pilot SFAs are to be started only after shutdown of the reactors for the decommissioning and after the removal of other SFAs from the reactor pools.

6. CONCLUSIONS

In 2011 and 2014, the non-conforming RBMK-1000 SFAs were shipped to Mayak PA and reprocessed at the RT-1 plant. The significant results are: concept of non-conforming RBMK-1000 SNF shipment and reprocessing in the expendable ampoules was verified; safe shipment of non-dried leaky SNF in the air-tight ampoules was proved; and procedure for SNF

transport and handling at the Leningrad NPP and the RT-1 plant have become mutually acceptable. In future the technologies for delivery of leaky, severely damaged and pilot SFAs for reprocessing and a technology for SFA grid skeleton repair in order to remove all non-conforming SNF from the NPPs are to be developed and introduced.

RETRIEVAL OF DAMAGED FUEL FROM WET AND DRY STORAGE USING INNOVATIVE REMOTE HANDLING TECHNIQUES

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Abstract

James Fisher Nuclear (JFN) has supplied remote handling solutions to the nuclear industry for over thirty years. This paper describes innovative remote handling techniques and equipment used for the retrieval of damaged fuel from both wet and dry storage, plus the removal of damaged fuel in a reactor being de-fuelled prior to decommissioning. In all these cases the fuel could not be retrieved by the installed fuel handling equipment and new solutions were required to remove the fuel safely and cost effectively. The retrieval of damaged fuel from the dry fuel stores at the Wylfa Magnox power station in the UK is described. The challenge for JFN was the provision of suitable equipment for the removal of the fuel through the existing fuel removal route and loading into the standard transport flasks for onward transport to the Sellafield reprocessing plant. Stringent requirements needed to be met including there should be minimum modifications to the plant and there should be no attachments left on the damaged fuel elements prior to placing in the transportation flasks. Another example of damaged dry fuel retrieval using specialised equipment is the retrieval of damaged fuel from two Magnox power stations during de-fuelling operations prior to decommissioning. A legacy fuel pond at Sellafield is an example of wet storage retrievals. The pond contains 1200 storage skips of spent fuel and has also accumulated sludges from the corrosion of fuel cladding and the fuel itself, fuel fragments, other debris and organic matter which have blown into the pond. Submersible Remote Handling Vehicles (ROV) have been deployed for the retrieval of damaged fuel from the floor of this pond and the sorting and segregation of fuel in the storage skips.

1. INTRODUCTION

At their peak in the mid-1970s, Britain's 26 Magnox reactors, spread across 11 sites, had an installed capacity of around 4200 MW(e) [1]. Today only one reactor, at the Wylfa site on the Isle of Anglesey, is still operational, whilst the remainder are at various stages of decommissioning or have been placed under a care and maintenance regime. These latter stages of the Magnox reactor lifecycle pose a number of significant engineering challenges, not least in the handling and recovery of stuck or damaged Magnox fuel, be that from within the core of the reactor itself or from pond or dry storage facilities.

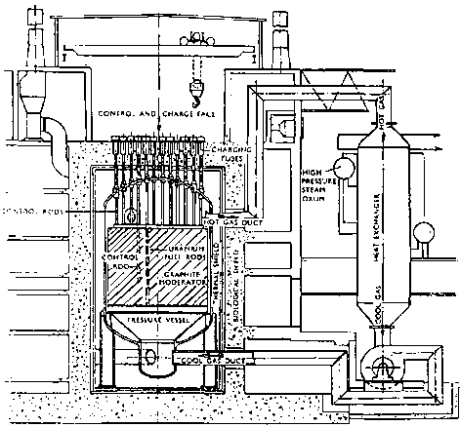

FIG. 4 Schematic of Calder Hall, the first Magnox power station [1].

Early Magnox reactor designs (see Fig. 1) had the reactor core inside a steel pressure vessel with a concrete shield surrounding it, and the steam generators outside. Later designs were purely intended for power generation and had a concrete pressure vessel with integral steam generators. The reactors use natural uranium metal fuel, have a graphite moderator and are cooled by carbon dioxide at a pressure of approximately 20 bar [1]. Unlike the fuel stringers used in the later and more efficient Advanced Gas Reactor designs, Magnox reactors are fuelled by thousands of individual fuel elements (Fig. 2) stacked nose to tail in vertical fuel channels formed by holes through the individual graphite blocks making up the core.

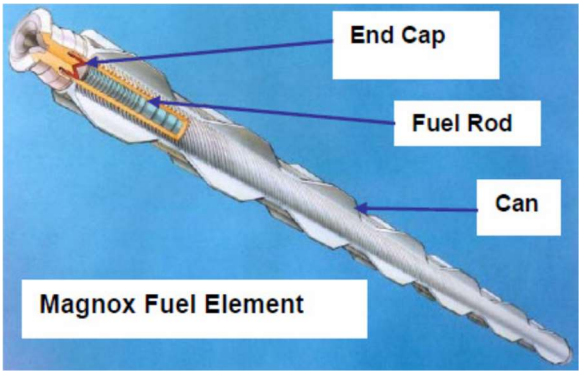

FIG. 5. A typical Magnox fuel element with lifting button [2].

Although the exact design of the Magnox fuel element varies dependent upon the station, each element comprises of a natural uranium bar (i.e. not enriched) sheathed in a magnesium alloy cladding (magnesium non-oxidising which is where the Magnox name is derived). Grooves milled into the external diameter of the cladding and sets of fins equally spaced over the length of the element provide an enhanced surface area for heat transfer. The fins also provide central axial location of the fuel elements in the reactor fuel channels.

The top of the element features a “button” or “spider” that is used as the lifting feature for the transfer of the fuel element to and from the reactor cores. The bottom of the element features a cone which locates into the feature on the neighbouring lifting feature to enable the elements to be stacked in the fuel channels of the reactor. As an example, a Wylfa design fuel element has a diameter across the fins of 95 mm, a length of approximately 1150 mm and an overall mass of 18 kg.

Once the irradiated fuel is removed from the reactor it is stored locally at the facility for a minimum period of 90 days, traditionally in an aqueous pond environment. The chemical reactivity of the Magnox fuel cladding with water means that the fuel cannot be stored indefinitely in this form, so it must be reprocessed.

In routine fuelling and defueling operations, Magnox fuel elements are handled using standard grabs designed to interface with the lift feature on the top of the fuel element. In certain circumstances, where the lift feature has become damaged or the element is jammed in its location, the standard handling equipment and techniques cannot always be employed to recover the element.

Faced with such challenges, specialist tooling is often required to facilitate the successful recovery of damaged or stuck elements, be that from within the core of the reactor itself, from

storage ponds or from dry storage cells. Utilising an iterative, pragmatic, trials based approach, JFN have designed, manufactured and operated machinery to successfully solve these fuel handling challenges in each of these environments.

Adopting a trials based approach offers a number of distinct advantages over more classical design methodologies, namely that key risks are mitigated and addressed as the design progresses; continual assurance is gained through the trials, operators become familiar with the equipment before it is used on site, and overall risk, cost and programme time is reduced.

Three case studies of novel fuel handling challenges are presented below, along with a description of the successful solutions developed by JFN.

2. CASE STUDY ONE: DAMAGED FUEL RECOVERY FROM REACTOR CORE AT CHAPELCROSS

First commissioned in 1959, Chapelcross in south west Scotland was one of the first generation of Magnox facilities, and was the sister station to Calder Hall, the UK's first commercial nuclear power station.

The design of the Chapelcross reactors mirrored that of its sister station, with the reactor core being enclosed in a steel pressure vessel, surrounded by a concrete shield. Contained within the pressure vessel is a complicated 24 sided, 6 metre high graphite structure made up of alternate layers of interlocking blocks and tiles. The structure, which weighs approximately 1150 tonnes, acts as the moderator and reflector for the reactor.

1696 vertical fuel channels penetrate the graphite core, and under normal operations, each of the fuel channels contains six fuel elements stacked nose to tail, making a total of 10 176 elements in the core, with a combined fuel weight of approximately 120 tonnes [1].

When the irradiated fuel elements are exhausted they are removed from the core using the discharge machine grab. The grab locates onto the lifting 'spider' on the top of each fuel element. After their 90 day pond storage they are packed into a standard Magnox fuel flask, and are exported to the Fuel Handling Plant at Sellafield for reprocessing.

As Chapelcross reached the end of its operational life, consideration was given to the full defuelling of each of the four reactors, and in particular how a number of elements with damaged lifting spiders could be successfully recovered, given that the standard discharge grab was not able to acquire this lift feature.

JFN was therefore engaged to design and develop a device capable of recovering these fuel elements from the core of the reactor. The device had to fit within the tight dimensional constraints of the existing discharge machine grab, and be lowered by the discharge grab to remotely locate onto the fuel element over the damaged lifting legs. The device then had to be able to exert a pull force of up to 500 kg on the element. When attached to the fuel element the device had to fit into a standard Magnox fuel skip, with the elements packed normally, to ensure that the maximum packing density could be obtained, and hence the discharge rate required to meet defuelling deadlines could be maintained.

Following early concept generation and optioneering activities, it was decided to embark upon the manufacture of a series of prototype devices to enable their relative merits to be easily assessed, and to inform the development of the design as the project progressed. This approach also ensured continual and progressive stakeholder buy in, and ultimately helped to

ensure that the final design was fit for purpose. The early prototypes of this device (see Fig. 3) involved a series of fixed and pivoting barbs arranged in different housings, incorporating variations to the barbs and their pivot orientations.

FIG. 3 Test of an early prototype device.

Several versions of these devices were designed, built and tested in order to determine the most effective design in terms of reliability and maximum load capacity. Through undertaking this prototype testing, issues were highlighted over the ability of the device to retain its grasp on the fuel elements towards the limit of the load capacity requirement. Problems were encountered with the pivoting barbs losing their grip on the outer surface of the fuel element as the jaws 'bit' into the element, sliding upwards as the surface layer of the magnesium cladding became damaged and distorted.

At this stage, the operation of the device was revisited and a new design conceived. This revised design incorporated a series of small jaws mounted inside an internal, downward tapering sleeve. Once positioned over the fuel element body, the jaws were designed to grip around the outside diameter of the fuel element in a similar manner to a collet.

The jaws had small teeth running circumferentially around their inside face to form grooves that aided the grip on the fuel element. The grooves needed to be large enough to dig into the outer diameter of the cladding, but still ensure that the device could be readily released once the element had been recovered.

Initial trials with this design proved that the ease of deployment over the fuel element, and the load carrying capacity of the grab were considerably greater than those of the early barb based prototype models. Full design substantiation was then undertaken to prove the grab was fit for purpose, including Finite Element Analysis of all the load bearing components to determine maximum stresses and deflection.

After minor modifications to optimise the performance of the device, by increasing the lead-in chamfers, it was found that the grab could be easily positioned over the fuel element without intervention, and was capable of repeatedly lifting in excess of 500 kg. The final device design (see Fig. 4) was therefore manufactured by JFN and a number supplied to Chapelcross. These devices have been subsequently used to successfully recover a number of fuel elements

with broken lifting spiders from the reactor. The elements, complete with the recovery device, were then exported in a standard Magnox fuel flask to Sellafield for reprocessing.

FIG. 4 The final device design.

The simplicity of the design and its successful operational deployment has led to further developments to accommodate different types of fuel elements and discharge arrangements. As well as those supplied to Chapelcross, JFN has also supplied the devices to Calder Hall and Wylfa power stations for similar fuel recovery projects.

3. CASE STUDY TWO: MAGNOX FUEL RECOVERY USING ROVS

The First Generation Magnox Storage Pond (FGMSP) at Sellafield is one of the site's four legacy storage pond facilities. The pond was constructed in the late 1950s to receive and store irradiated fuel from Magnox reactors in open topped containers called skips. In its 26-year operational life the FGMSP facility stored and prepared nearly 2.5 million fuel rods (approximately 27 000 tonnes of fuel) for reprocessing [3].

During the 1970s, Magnox reprocessing slowed causing fuel to be stored in the pond for longer periods than normal. This has resulted in the magnesium alloy casing corroding in the pond leading to the creation of radioactive sludges and poor underwater viewing.

The pond is located in a highly congested area, surrounded by buildings which originated in the early years of nuclear operations on the site. This limits the opportunities for new infrastructure, heavy lifting and temporary facilities around the perimeter of the pond.

Over the years the pond has accumulated significant quantities of waste materials, including sludge from the corrosion of fuel cladding, fuel itself, fuel fragments and other debris and organic matter which has blown into the pond. The skips used to store the fuel elements were originally stacked up to three high, and some of these have been dislodged or have tipped over, contributing to the accumulation of sludge, fuel elements and fuel debris on the pond floor. Currently the pond contains 1200 skips, an estimated 1500 cubic meters of radioactive sludge and 14 000 cubic meters of contaminated water.

In order to aid the clean-up of the facility and to provide enhanced risk and hazard reduction within the ponds, the export of fuel from the FGMSP is required as soon as the export capability becomes available. In order to meet the downstream reprocessing requirements,

there is a need to sort, segregate and consolidate the fuel within the storage skips, as well as recover fuel from the floor of the pond.

Owing to the high dose rates and corrosive environment associated with the ponds, any decommissioning work needs to be undertaken using remote handling techniques. Traditionally this has been done using long reach tooling and remote handling systems deployed from outside of the pond, however this approach has a number of inherent drawbacks and risks associated with it. JFN and Sellafield Ltd (SL) have therefore worked together to develop the use of Remotely Operated Vehicles (ROVs) to support the fuel recovery process.

The use of ROV technologies within the pond provides a number of advantages over alternative methods:

- The ROVs are controlled remotely at a distance from the side of the pond to minimise operators' exposure to radiation and to avoid potential radiological contamination;
- The ROVs are able to access all areas of the pond, which other remote handling techniques cannot;
- By modifying mature, commercial off the shelf (CoTS) technology, significant cost and programme savings can be made without compromising the quality, function or safety of the solution.

The submersible ROV market is a large, mature high technology market, which principally serves the offshore energy industry. ROVs are also used in other sectors including defence, security and film making, but their use in the nuclear industry has been relatively small, with visual surveys being their principal application.

Sellafield first deployed ROVs in the FGMSP in 1999, as a tool for visual inspection. Although this programme demonstrated that ROVs were capable of operating in the challenging pond environment and could gather useful information, it was not expanded to consider other applications. Therefore, in 2008 JFN (who had been operating ROVs since 1999) were engaged to develop the ROV capability and to extend the scope of operations beyond visual surveys to include more advanced characterisation techniques and to investigate the remote handling capabilities of ROVs to carry out actual fuel handling within the FGMSP.

Working collaboratively, JFN and SL identified potentially suitable ROVs and tooling which could be modified and developed to carry out these new activities. During this early phase of development close working relationships were established between JFN, SL, the ROV suppliers and others in the supply chain, including specialist tooling manufacturers and subject matter experts, which was to prove crucial to the successful outcome of the project.

The complexity and unique nature of the engineering challenge meant that in order to ensure a successful outcome, the project was built up over a series of incremental developments. The methodology of design, test, refine, test, train and rehearse meant that perceived risks with both the technology and techniques were mitigated as part of the ongoing development process.

In order to provide a remote handling capability within the pond, a bespoke skid was developed for the existing ROV, which had previously been modified and used to perform visual surveys of the pond. The skid interface was designed to be modular to allow a range of bespoke skids with specific tooling to be created and quickly exchanged on the ROV (Fig. 5).

FIG. 5 Hydraulic manipulator in use in the JFN test facility.

Mounted on this skid is a manipulator. The manipulator is based on a CoTS unit, which has been modified to meet the necessary project requirements and to ensure it is compatible with the pond environment. The manipulator has been designed to be failsafe i.e. to release its load under fault conditions or loss of power, to ensure that the ROV is always recoverable from the pond. Barbs on the gripper increased the gripping effectiveness, and key components of the manipulator are coated in a protective layer of Cerakote to enable the manipulator to withstand the alkaline conditions within the pond.

The manipulator skid was deployed from the ROV in JFN's test tank to enable extensive trials and training to be undertaken. As well as verifying the function of the manipulator, these trials also enabled operator skills to be built up and appropriate operational methodologies to be experimented with and finalised ahead of deployment within the pond itself. This approach improved the effectiveness and efficiency of the in pond operations, reducing the dose uptake and risk on plant. In addition the trials were also used to support PUWER[1] assessments and HAZOP[2] studies of the new equipment, procedures and processes, and enabled the equipment to be readily demonstrated to all stakeholders, giving them confidence in the performance and reliability of the ROVs and tooling in the pond.

Following these intensive trials, the ROV and manipulator skid was deployed in the FGMSP to pick up the loose fuel elements from the floor of the pond and place them in a skip. The ROV with its manipulator arm was successfully able to acquire and move fuel between skips. To date more than 4500 kg of fuel elements have been sorted and segregated using ROV technology, with many of these recovered from the pond floor. This work is on-going.

Although all ROV fuel recovery to date has been carried out using hydraulic manipulators, JFN has also developed two electric manipulators in response to an SL requirement to reduce

[1] Provision and Use of Work Equipment Regulations 1998
[2] Hazard and Operability Study

the risk of contaminating the ponds with hydraulic fluids. Not only does the use of an electric manipulator remove this risk but lifetime maintenance and operational costs are potentially lower than their hydraulic counterparts.

JFN has worked closely with a specialist company to re-engineer their standard manipulator products for use in nuclear ponds. Modifications included manufacture from stainless steel rather than aluminium to withstand the alkaline conditions of the FGMSP. The control system hardware of the manipulator was also thoroughly tested to ensure it could tolerate the high radiation environment. This was done by exposing the system to a ^{137}Cs radiation source in the JFN radiological calibration service facility, where the effect of the radiation on its performance was tested and evaluated. A new gripper was designed for general manipulation tasks and this also was fitted with a fail-safe mechanism. Two electric manipulators have been developed for the ROV: a 'forward facing manipulator' and an 'underslung manipulator'. These have been extensively trialled and demonstrated to SL at all stages of the development in the JFN test facility.

4. CASE STUDY THREE: STUCK FUEL RECOVERY FROM WYLFA DRY STORAGE CELLS

Wylfa Nuclear Power Station on the Isle of Anglesey was the last Magnox station to become operational, in 1971. Wylfa has two reactors with a combined total capacity in excess of 900 MW(e). Each reactor houses 6156 vertical fuel channels and each channel contains a stack of eight Magnox fuel elements, making a total of 49 248 elements per reactor [4]. Wylfa does not have fuel storage ponds but relies on dry storage cells with carbon dioxide cooling for spent fuel storage.

The fuel is removed from the reactors using the fuelling machine, which then transfers the irradiated fuel into the tubes in one of three Primary Dry Store Cells (DSC 1, 2 & 3), which are accessed via three plugs in the Pile Cap. When the fuel has cooled for its minimum period of 90 days, it is removed from the Dry Store Cells and is transferred via the Primary Discharge Route into flasks for transport to Sellafield and eventual reprocessing.

The Primary Dry Store Cells each contain 588 vertical tubes of 105 mm internal diameter. These tubes are suspended from a header plate and are configured at 11 different concentric pitches. 576 of the tubes are designated for the storage of irradiated fuel elements, and each tube can accommodate a maximum of 12 vertically stacked fuel elements, giving a total storage capacity of 6600 elements in each dry store cell. All tubes are cooled internally by natural carbon dioxide circulation, and externally via an air circulation system.

Each dry store cell has a dedicated transfer chute with an electrically powered hoist which is equipped with a standard discharge grab. This solenoid operated grab is fitted with three jaws which locate onto and close around the lifting button on the top of the fuel element. The hoist is then used to transfer the element from the dry store cell into the flask filling area ready for export.

A number of fuel elements within the dry store cells were unable to be recovered with the discharge grab because of: damage to the lifting button (see Fig. 6) preventing acquisition by the standard grab or the elements are jammed in the storage tubes, meaning that the discharge winch is unable to supply sufficient force to free the stuck elements.

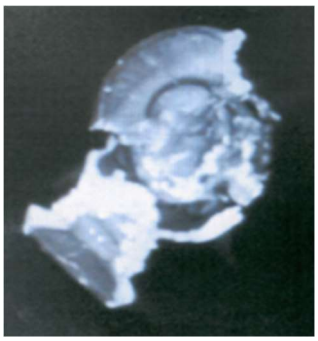

FIG. 6 An element with a damaged button in DSC 1.

As Wylfa neared the end of its operational life, the need to recover the stuck fuel elements became more necessary, partly to aid final defuelling and decommissioning of the station, but also to enable cross-reactor refuelling to take place, an activity that was vital to being able to extend the operational life of the facility.

Cross reactor refuelling meant permanently shutting down reactor one owing to a lack of new Magnox fuel, and transferring the best fuel from reactor one into reactor two to enable the remaining reactor to continue to operate efficiently. A prerequisite to the cross-reactor refuelling process was the retrieval and export of all the stuck or damaged fuel elements from dry store cells 1 & 2 to enable sufficient free positions to be obtained in the cells to allow the necessary shuffling of fuel elements to take place between the reactors. In 2012 JFN were awarded the contract to design and manufacture a machine to recover these stuck fuel elements from the dry store cells to enable cross-reactor fuelling to commence. JFN has previously worked alongside Wylfa to recover damaged fuel elements from their secondary dry store cells.

Working closely with the client's operators, engineers and safety case team a specification for the machine was drawn up, with the fundamental requirement being to enable a pull force of up to 1000 kg to be incrementally applied to free the stuck elements, without piercing the magnesium alloy outer casing of the element. The machine also had to satisfy a number of other significant requirements including knowing the position of the grab and recovered fuel element at all times, monitoring the force applied to the element, knowing when the element is free and preventing the accidental raising of an element onto the Pile Cap.

Using these requirements as the basis for the design, a number of potential solutions were devised and discussed with the client. The preferred solution from this optioneering process was an electro-mechanical grab, deployed by a single umbilical from a hoist unit on the Pile Cap.

With the concept design chosen, it was decided to embark on a series of trial based activities early in the scheme design stage. Initial trials focussed on the critical interaction between the fuel element casing and the grab jaw profile, to ensure that sufficient force could be exerted to free the element without piercing the casing. A number of different jaw profiles were machined and tested, ranging from abrasive diamond coated jaws to tool steel jaws with a series of teeth designed to bite into the element casing.

These jaws were tested on a Magnox element sample using a simple test rig. A fixed grip force was applied across the jaws and the resultant pull force prior to jaw slippage was recorded. The fuel element was then visually inspected, and the outcomes recorded. Not only did this initial testing allow the jaw profile to be optimised, but it also provided useful data for

the design of the grab itself, such as the grip force required, which in turn provided a starting point for the sizing and scoping calculations for the grab design. The trial also highlighted the need for the grip force to be continually applied as the grab was loaded, again providing key data for the sizing and rating of drivetrain components.

The outcome of this initial trial was then used as the basis for the design of the grab unit itself, with the geometry and profile of the jaws following that which was used on the test rig. This design was underpinned with calculations to verify the functionality and structural integrity of the grab. The final grab design featured an electric motor and gearbox, with a lead screw and nut arrangement used to provide the jaw closure force. Torque limits on the motor ensured that more force was always available to open the jaws than to close them, ensuring that the fuel element could always be released from the jaws after successful recovery. The jaws themselves were manufactured from 'Vanadis 4 Extra', a tool steel which has the necessary mix of hardness to ensure the teeth retained their sharp edge with repeated use and toughness to withstand the forces experienced by the jaws during use.

The grab also contained a number of notable design features, including a set of removable pins to enable the jaws of the grab to be remotely released via a fixture in the fuel discharge route in the unlikely event of grab failure. This meant that any element held in the jaws could be released and the grab recovered to the Pile Cap for maintenance. One jaw of the grab was deliberately lengthened to aid locating the grab over the fins of the fuel element. The grab also contains integral cameras and lights to provide full visibility of the remote operations.

Once the design of the grab was complete, and had been subjected to the necessary governance, it was manufactured and assembled (Fig. 7) ahead of the completion of the rest of the design. This approach enabled the performance of the grab to be verified early in the manufacturing phase, gave the opportunity for stakeholder buy-in and also enabled early engagement of the site operation team.

FIG. 7. Completed grab mechanism.

The design of the remainder of the machine followed a more traditional design process, in keeping with the reduced level of technical risk in these areas. The grab was designed to be suspended from a bob weight that not only aided deployment through the fuel route, but also contains a set of switches to detect the presence of an element in the grab. The hoist unit is based around a servo-motor powered drum and reeving mechanism, onto which the combined electrical and load bearing umbilical was wound. The servo-motor has variable speed and torque settings which can be adjusted incrementally as required, together with a mechanical clutch to provide overload protection.

The hoist is mounted on four load cells to give an indication of the pull force being applied to the element, and a combination of a manual cam box and encoder readout provide position indication. Three diverse systems are employed to prevent an element being accidentally brought to Pile Cap, the first using a combination of hoist position and bob weight switch position, the second using a Canberra gamma detector and the third using a physical stop that is installed around the umbilical and can only be removed once the gamma gate is closed below the grab, which cannot be physically achieved with an element in the grab. The control system for the machine has been designed to have a similar look and feel to the standard hoist control system, to assist with operators' familiarity and training.

Following the completion of the machine, an intensive series of testing and trials was undertaken at JFN's facilities. These trials not only enabled the performance of the machine to be verified and the function of all interlocks checked, but they also enabled the operators to become familiar with the control of the equipment in a non-critical environment. In addition the trials were also used to support the writing of procedures and processes for on plant operations, and enabled the equipment to be readily demonstrated to all stakeholders, giving them confidence in the performance and reliability of machine ahead of delivery.

Following successful completion of factory acceptance testing and operator training, the machine (Fig. 8) was transported to Wylfa, where JFN supported both pre-operations and inactive commissioning activities. The system has now been used to successfully recover 53 damaged fuel elements from the Dry Store Cells, winning a Magnox i4 implementation award in the process. This was a key enabler to permitting cross reactor refuelling to be carried out, which in turn, has enabled the generating life of the station to be extended well into 2015. During 2013 alone, Wylfa was able to deliver 3.63 TWh of electricity to the grid, a figure that could not have been achieved without cross-reactor refuelling [4].

FIG. 8. The finished machine in operation on the pile cap at Wylfa.

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IAEA'S ACTIVITY ON DEVELOPMENT AND APPLICATION OF A SAFETY CASE FOR DUAL PURPOSE CASKS

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Abstract

In 2011, following the previous International Conference of Spent Fuel Management in 2010, a Working Group (WG) was established to develop guidance for Member States for an integrated safety case for a Dual Purpose Cask (DPC), which is used for the transport and storage of spent fuel, with the support of both the Transport Safety Standards Committee (TRANSSC) and the Waste Safety Standards Committee (WASSC). The WG activity continued until April 2013. During this activity, a technical document was developed, which considers aspects such as acceptance criteria, ageing of spent fuel and materials used for the safety-related cask structures, and methods for inspecting cask condition for transport after storage. An integrated safety case for both transport and storage aims to support the application for the package design approval for transport and the application for the licensing of the storage cask. The technical document — to be further published as a TECDOC — provides useful technical aspects on this topic that can supplement the existing IAEA safety standards by providing a collection of scientific and technical arguments including safety assessments in support of: 1) The demonstration of compliance with the IAEA transport regulations SSR-6 [1], Regulations for the Safe Transport of Radioactive Material, 2012 Edition for off-site transport, including transport after storage; and 2) The demonstration of compliance with the international standards including GSR Part 5 [2], Predisposal Management of Radioactive Waste and SSG-15 [3], Storage of Spent Nuclear Fuel, and national requirements for dry storage of spent fuel as they apply to the DPC package during its storage period. As an outcome of this three year project, an "International Workshop on the Development and Application of a Safety Case for Dual Purpose Casks for Spent Nuclear Fuel" was organized in May 2014. The workshop was organized in order to enhance the participants' understanding of the proposed concept of an integrated safety case for DPCs, to analyze the gap between the current practices within Member States and the proposed concept, and to discuss ways of further improving the application of the DPC concept. At the workshop, several Member States presented their national legal framework and experiences on the interim storage of spent nuclear fuel in casks including both storage-only casks and DPCs. The following major topics were discussed during the workshop: 1) The adoption of a DPC strategy involves the provision of an applicable regulatory framework and technical considerations for long term management of DPCs. This includes contingency plans if there is a need to unload the spent fuel due to technical problems, or transport from the storage facility earlier than planned or other reasons; and 2) The long-term management of DPCs requires a safety case that evolves during the whole duration of storage period and forthcoming transport after storage, in order to provide sufficient information and justification to ensure the safety of the system. Some of the important aspects include: plausible and practical regulatory framework applicable to the whole storage period; cask and storage facility designs that consider monitoring and inspection plans, ageing management, possible changes of storage plans, measures for future incident; research and development programmes (mainly the ageing of spent fuel and the DPC) to ensure the proposed safety case is robust enough even after long term storage. The workshop also addressed a number of non-technical concerns such as record keeping and public acceptance that will affect sound management of the DPC concept over many decades. These concerns will require further consideration on an ongoing basis in addition to the technical aspects discussed at this workshop. Recommendations for organizing potential future IAEA activities were also discussed which will be contemplated to ensure and enhance the safety of DPC and other dry storage systems. In this presentation, overview of draft TECDOC and outcome of the workshop will be introduced.

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AGING MECHANISMS INFLUENCE ON TRANSPORT SAFETY OF DUAL PURPOSE CASKS FOR SPENT NUCLEAR FUEL

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Abstract

When storage of spent nuclear fuel (SNF) is done in dual purpose casks (DPC), the effects of aging on safety relevant DPC functions and properties have to be managed in a way that a safe transport after the storage period of several decades is capable, and can be justified and certified permanently throughout that period. The effects of aging mechanisms (like e.g. radiation, different corrosion mechanisms, stress relaxation, creep, structural changes and degradation) on the transport package design safety assessment features have to be evaluated. The consideration of these issues in the DPC transport safety case will be addressed. Special attention is given to all cask components which cannot be directly inspected or changed without opening the cask cavity, what are the inner parts of the closure system and the cask internals, like baskets or spent fuel assemblies. The design criteria of that transport safety case have to consider the operational impacts during storage. Aging is not subject of technical aspects only, but also of 'intellectual' aspects, like changing standards, scientific/ technical knowledge development and personal as well as institutional alterations. Those aspects are to be considered in the management system of the license holders and in appropriate DPC design approval update processes. The paper addresses issues which are subject of an actual IAEA TECDOC draft "Guidance for preparation of a safety case for a dual purpose cask containing spent nuclear fuel".

This paper has been removed as large proportions of the content have been published elsewhere. Interested readings should consult the following reference:

[1] WOLFF, D., PROBST, U., VÖLZKE, H., DROSTE, B., RÖDEL, R., Safety Aspects of Long-term Dry Storage of Type-B Spent Fuel and HLW Transport Casks, RAMTRANS, **15** (2004) 207-213.

REGIONAL COOPERATION ON SPENT FUEL MANAGEMENT: STATUS AND PROSPECTS IN EUROPE, ARAB REGIONS AND ASIA

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Abstract

Safe and secure management of spent fuel is a major responsibility for all countries operating nuclear plants and a major challenge for potential new entrants to nuclear power. The greatest challenge is implementing a long-term strategy leading to final disposal of the spent fuel or the high level wastes derived from reprocessing. Implementation of deep geological repositories is technically challenging, lengthy, costly and sensitive. Countries with few or no nuclear power plants may have difficulties in making available sufficient financial and human resources to meet this challenge and the potential economic, safety and security benefits of shared repositories are widely recognized. The last few years have seen growth in initiatives aimed at exploring the opportunities for shared storage or disposal facilities for countries in the same geographical region. Regional concepts have moved ahead through efforts supported by the IAEA, the European Commission and the Arab Atomic Energy Agency, and through study projects run by the NTI, CSIS and IFNEC. A significant part of the work promoting regional concepts has been performed by the Arius Association, which provides the secretariat for the self-funded Working Group on a European Repository Development Organisation (ERDO-WG) and also runs projects exploring regional concepts in the Arabian Gulf region, North Africa and South East Asia, with financial support for the non-European work provided by the Sloan and Hewlett Foundations. The most advanced initiative currently is in Europe, where the ERDO-WG has since 2009 been coordinating the efforts of ten countries. This paper summarizes advances during the last years in each of these global regions.

1. INTRODUCTION

Safe and secure management of spent fuel is a major responsibility for all countries operating nuclear plants and a major challenge for potential new entrants to nuclear power. The greatest challenge is implementing a long-term strategy leading to final disposal of the spent fuel or the high level wastes derived from processing the fuel. An unavoidable element of such a strategy is ensuring that a deep geological repository will be available. However, implementation of deep geological repositories is a technically challenging, lengthy, costly and sensitive challenge for any nation with an inventory of long-lived radioactive wastes. Countries with few or no nuclear power plants may have difficulties in making available sufficient financial and human resources to meet this challenge. For this reason, the potential economic, safety and security benefits of shared repositories are widely recognized. Cooperation between countries has long been a feature of radioactive waste management. National programs participate in discussion forums and joint projects, many run under the auspices of international organizations such as the International Atomic Energy Agency (IAEA), the European Commission (EC) or the Nuclear Energy Agency (NEA). The national political and public sensitivities related to accepting radioactive waste from a foreign country have, however, caused initiatives for multinational repositories to progress only very slowly. Nevertheless, the last few years have seen continuous growth in interest in specific initiatives aimed at exploring the opportunities for making shared storage or disposal facilities available to countries in the same geographical region. Regional concepts of this type have been moved ahead through efforts supported by international organizations, such as the IAEA, the European Commission (EC), the Arab Atomic Energy Agency (AAEA), and through specific study projects run by entities such as the NTI, the CSIS and the IFNEC Project. A significant part of the work promoting regional concepts has been performed by the Arius Association which provides the secretariat for the self-funded Working Group on a European Repository Development Organization (ERDO-WG) and also runs projects that are exploring regional concepts in the Arabian Gulf region, North Africa and South East Asia, with financial support

for the non-European work provided by the Sloan and Hewlett Foundations in the United States.

1.1. Are shared repositories ethical and credible?

Before moving to the documentation of specific initiatives aimed at promoting multinational concepts, it is instructive to re-visit two objections that have often been raised concerning regional shared repositories. It has been asserted that they are not ethical, since each country should look after its own waste, and that they are not credible, since there are, as yet, no volunteer host countries.

1.2. Ethical aspects

Implementing disposal implies that the present and immediately following generations accept operational risks and invest resources in order to protect far-future individuals. The issue is then whether we are being fair to present and future generations — i.e. a question of intergenerational equity. Does waste disposal really present unique ethical issues? There are, in fact, other activities today for which the same dilemma arises. Global warming due to CO_2 is the most topical subject, but there are numerous older examples for which the issue of fairness to future generations has not been recognised explicitly enough. A clear case is the exploitation of natural resources in Earth's crust. However, it must be acknowledged that the prevailing atmosphere of nuclear fear in many countries results in a debate on the long-term aspects of radioactive waste disposal that is much more intensive than for other cases. When we move to the issue of transfer of wastes to another country, the ethical debate has often become even more intense.

The principal argument put forward against multinational repositories is that they are ‘unethical’, since each country using nuclear technologies should dispose of wastes on its own territory. This is certainly not the view of the numerous countries considering the regional option, nor of international organisations like the IAEA and the EC, both of which are on record as recognising that such transfers are not only ethical, but can also be environmentally beneficial. What is perhaps unethical is for major nuclear nations to transfer nuclear technologies to smaller nations, without consideration of the long-term challenges resulting from managing the radioactive wastes produced. The large nuclear nations have been happy to export nuclear fuel cycle products and services (including reactors, fuel fabrication, reprocessing, etc.) to any country that would purchase these. It is self-serving and inconsistent to single out disposal as the one part of the nuclear fuel cycle that may not be internationalised. Arguments for national nuclear self-sufficiency ring rather hollow in countries that depend upon imported uranium and thus avoid all problems associated with mining — the fuel cycle step that has the most environmental impact.

1.3. Credibility of regional concepts

A further objection often raised by those sceptical of regional repositories, that they are not credible because no volunteer host country or site has been identified at present. This is perhaps even more eccentric. If it is to be a criterion for credibility, then virtually all national spent fuel and HLW waste geological disposal programmes are not credible, since only Finland, Sweden and France have identified deep disposal sites. In reality, no site or country is currently nominated for a multinational repository for exactly the same reason that national programmes do not nominate a site at the very beginning of their efforts. Both national and multinational programmes must fulfil some important prerequisites before taking the important step of selecting preferred sites:

- Getting all participants on board concerning the desirability of finding a common site (or sites) — that is, recognition of a common need;
- Identifying and transparently documenting all of the technical and non-technical criteria that a site would need to satisfy;
- Establishing, documenting and discussing with the involved public the advantages (and drawbacks) that a site would experience;
- Building trust in the organisations that are charged with identifying and developing the site.

Only then should one move to discussion of specific siting options. This is the gradual process envisioned by the Arius Association and in the ERDO-Working Group, whose activities are described below. Neglecting to satisfy adequately all of these prerequisites before moving to repository siting has led to setbacks or failures in various national disposal programmes around the world.

2. A BRIEF HISTORY OF MULTINATIONAL INITIATIVES AT THE BACK-END OF THE FUEL CYCLE

A comprehensive history of early proposals for multinational storage and disposal is given in IAEA documents from 2004 and 2010 [1, 2]. These reports contain details of the list of older initiatives given below:

- Regional Nuclear Fuel Cycle Centres (RFCC) (1975−77);
- International Spent Fuel Management Group (1975/1982);
- International Plutonium Storage (IPS) 1980;
- OECD/NEA Study (1987);
- Synroc Study Group in Australia (mid-1980s);
- IAEA Expert Groups (1994/95, 2001/02);
- International Working Group (late 1990s);
- Marshall Islands (1995−97);
- Wake Island/Palmyra Island (mid 1990s);
- Pangea (1997−2002);
- Non Proliferation Trust (NPT) (1998−2000).

Around the turn of the century, interest arose again in multinational or multilateral approaches in the fuel cycle. This was partly due to increasing concerns about ensuring safety and security in a world where nuclear power seemed to be set to expand. Various specific project proposals were made, as listed below, and specific support actions were initiated by international organisations, as is detailed in section 3.

- Russian proposals (2001−present)

Minatom was involved in several of the proposals mentioned above and over the past few years Russia has become increasingly serious about spent fuel import. It is the only country that has publicly supported this at government level. There is some ambiguity regarding the options for returning the wastes resulting from reprocessing in Russia to the client country, but definite agreements have been made in some cases for take-back of Russian fuel with no return of wastes (e.g. with Iran).

- Kazakhstan proposals (2001, 2002)

The Government of Kazakhstan declared its intention to host an international repository in the Mangistan region. Once again, however, political opposition to the proposal led to the concept being dropped.

- 12 NFC Proposals at the IAEA

Around the time of the IAEA General Conferences in 2005 and 2006, twelve proposals related to multinational fuel cycle concepts were put forward [3]. These all focussed on nuclear security and non-proliferation and they mostly focussed suggestions related to the front end of the fuel cycle.

The exceptions were the Global Nuclear Power Infrastructure (GNPI) concept from Russia and the Global Nuclear Energy Partnership (GNEP) from the USA. Both these proposals included options for fuel supplier countries to take back the spent fuel or, at least, to organise a take-away option to a multinational facility in a third country. However, GNEP effectively dropped this option and transformed itself into the International Framework for Nuclear Cooperation (IFNEC), which continues to organise discussion groups, including one devoted to consolidated spent fuel management. The Russian initiative continues to be active and the possibility of being offered a spent fuel take back arrangement may be functioning as a powerful argument for new entrant nuclear countries to opt for Russian technology.

- SAPIERR projects (2003-2009) [4]

The SAPIERR I Pilot Project for European Regional Repositories, initiated by Arius, studied potential options for regional collaboration and for regional repositories to be identified, though it did not extend to site identification. Following this pilot study, the SAPIERR II project assessed the feasibility of European regional waste repositories.

> "*The tasks were:*
>
> - *Preparation of a management study on the legal and business options for establishing a multinational repository organisation.*
> - *Study on the legal liability issues of international waste transfer within Europe;*
> - *Study of the potential economic implications of European regional storage facilities and repositories;*
> - *Outline examination of the safety and security impacts of implementing one or two regional stores or repositories relative to a large number of national facilities;*
> - *A review of public and political attitudes in Europe towards the concept of shared regional repositories;*
> - *Development of a Strategy and a Project Plan for the work of the multinational organisation*" [5].

Organisations from around half of the 28 EU Member States participated in some part of the SAPIERR work and, at its conclusion, several of these countries joined the European Repository Development Organisation Working Group (ERDO-WG), which is described below.

3. CURRENTLY ACTIVE COOPERATION INITIATIVES

3.1. IAEA support for multilateral initiatives

The IAEA was an early champion of multinational cooperation in nuclear fuel cycle issues, as indicated above. With the rise in interest around the turn of the century, direct support was expressed at the highest levels in the Agency. The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, finalised in 1997, recognised that each country has a direct responsibility for managing its own radioactive wastes, but acknowledged "…that, in certain circumstances, safe and efficient management of spent fuel and radioactive waste might be fostered through agreements among Contracting Parties to use facilities in one of them for the benefit of the other Parties, particularly where waste originates from joint projects". The IAEA has also published a series of technical documents addressing the key issues related to implementation of shared storage or disposal facilities. These include [1, 2, 6−8].

The most recent of these reports describes phased development of a multinational repository and also directly discusses the risks attached to such projects. The 2004 document introduced an important classification of multinational approaches, by identifying three potential options. The first is a cooperation scenario in which a shared repository is developed by a group of partner countries. If the countries are adjacent or close, the repository is labelled as regional; otherwise, it is a multinational repository. The second option is an add-on scenario, which assumes that a host country that has already implemented at national repository offers, at some later stage, to complement its national inventory of wastes for disposal by wastes imported from other countries. The third variant is an international or supranational scenario in which a repository (or network of repositories) would be fully in the hands of an international body and each host country would cede control of the necessary site to the specified international body. This last option was judged the least feasible, but increasing concerns over proliferation and nuclear security today might imply that such an approach could become credible.

The IAEA document with the greatest weight was, however, the report on Multilateral Approaches to the Nuclear Fuel Cycle, produced in 2005 by a high level Expert Group at the request of the Director General [9]. This report looked at multilateral approaches that could be employed in enrichment, reprocessing and final disposal. Of most relevance here are the conclusions that the Expert Group drew concerning spent fuel disposal. These are reproduced here, since they reflect directly the situation that still prevails today.

> "*At present there is no international market for spent fuel disposal services, as all undertakings are strictly national. The final disposal of spent fuel is thus a candidate for multilateral approaches. It offers major economic benefits and substantial non-proliferation benefits, although it presents legal, political and public acceptance challenges in many countries. The Agency should continue its efforts in that direction by working on all the underlying factors, and by assuming political leadership to encourage such undertakings.*
>
> *Small countries should keep options open (national, regional or international), be it only to maintain a minimum national technical competence necessary to act in an international context*" [9].

The Agency has indeed continued its efforts. This is illustrated by the reports subsequently produced, by the efforts of the IAEA Technical Cooperation Department to support regional activities and by the on-going work on the INPRO project, which only some weeks ago ran a Dialogue Forum in Vienna on the topic Cooperative Approaches to the Back End of Nuclear Fuel Cycle: Drivers and Legal, Institutional and Financial Impediments.

3.1.1. EC back-end cooperation projects

For some smaller EU Member States, implementation of a national geological repository on anything other than very long timescales is not practicable, because inventories are too small and costs too high. For such States, timely access to safe and secure disposal facilities will remain problematic or even infeasible unless regional, shared repositories can be implemented.

A significant strategic development took place in 2010 when the EC adopted the Directive on the Management of Spent Fuel and Radioactive Waste [10]. In the context of the present paper, the main message is that the option of EU Member States sharing repositories is kept open by Clause 3 in Article 4 on General Principles, which states that "Radioactive waste shall be disposed of in the Member State in which it was generated, unless at the time of shipment an agreement has entered into force between the Member State concerned and another Member State or a third country to use a disposal facility in one of them." The Directive implies that regional cooperation could be an important aspect of the detailed plans that the EC expects Member States to produce within 4 years. Nevertheless, the binding text of the Directive reflects the earlier views in the IAEA Join Convention and emphasises that countries should not use the prospects of regional disposal as a justification for remaining inactive.

The Radioactive Waste management sub-group of the European Nuclear Energy Forum (ENEF) has subsequently produced guidance for EU Member States [11] on how to meet those requirements of the Waste Directive that are focussed on R&D. In practice, the EC has provided support for numerous multilateral R&D projects that can contribute to enhancing cooperation at the back-end of the fuel cycle. An important step was the introduction of a Technology Platform on Implementing Geological Disposal [12], aimed primarily at helping the advanced EU national programmes to move towards construction and operation of such facilities. It is noticeable that all current EC activities tend to focus specifically on R&D. However, for many Member States with less advanced programmes, long-term R&D on radioactive waste management disposal solutions is a less immediate need than the necessity for real-time strategic advice and help. The extensive EC support to countries with advanced geological disposal research should be paralleled by support to the many other Member States who are seeking practical solutions today for immediate issues affecting European nuclear safety and security, associated with existing spent fuel and radioactive wastes.

3.1.2. The Arius association

Arius [13] was established in 2002 as the first formal body dedicated to supporting concepts for shared disposal facilities. A key objective is to explore ways of making provision for shared storage and disposal facilities for smaller users, who may not wish to — or may not have the resources to — develop facilities of their own. A decision was taken early to focus on the European region, since cooperation frameworks already existed. Arius was then instrumental in managing the SAPIERR projects described above. These projects led directly to the establishment of the ERDO-WG, whose activities are described below, and Arius

currently provides the secretariat for the ERDO-WG. But efforts have not been restricted to Europe.

Every one of the comprehensive list of IAEA documents referred to above has had major input from Arius. In fact, Arius was the key advisor to the Expert Group set up by the Director General when it debated options for cooperation at the back-end. Arius has also been requested to provide input on multinational approaches to a number of projects or studies that have examined the future of nuclear power, or developments in nuclear fuel cycles. These include initiatives by the following organisations:

- AAAS: The American Academy of Arts and sciences (AAAS) supports, as part of its Global Nuclear Futures initiative a project which focuses on the merits of regional storage centers for spent fuel, in particular in South East Asia [14];
- NTI/CSIS: Arius has participated in workshops in Paris and in Taipei on "New Approaches to the Nuclear Fuel Cycle" co-hosted by the Nuclear Threat Initiative (NTI) and the Proliferation Prevention Program at the Center for Strategic and International Studies (CSIS) [15]. The Taipei meeting was followed by a CSIS-NTI Workshop focusing on proliferation issues associated with multinational back-end initiatives;
- IFNEC/CFS: International Framework for Nuclear Energy Cooperation (IFNEC) has a Reliable Nuclear Fuel Services Working Group, which organized a meeting on Developing Options and Pathways for Disposal of Spent Nuclear Fuel and High-Level Waste. Arius was invited to participate in this meeting. Key conclusions drawn included the following: For several reasons, including safety, security and economy, the concept of a multinational repository merits being addressed within an international forum, like IFNEC.

Wide and sustained Arius engagement in multinational initiatives has also been made possible over the past several years with the financial support of the US Sloan and Hewlett Foundations. Original funding from the Hewlett Foundation was to support the European efforts, whereas the Sloan Foundation funding was explicitly to look at multinational opportunities outside Europe. Subsequently, both Foundations provided equal funding for a pilot project assessing the feasibility of applying the ERDO concept outside Europe. This study was completed in early 2011 and a follow up project was developed, concentrating on feasibility studies in the Arabian Gulf region and in South East Asia. This four year feasibility study is now nearing completion and its conclusions feed into the comments in Section 4 on the possible global future of multinational disposal concepts.

Arius took various steps to present the European work to other groupings in Arab regions and in South East Asia.

Workshops on Regional Collaboration on Radioactive Waste Management in MENA Countries were organised by the IAEA together with Arius, with significant input from the AAEA, in the UAE and in Tunisia. These events made it clear that the priorities in those Arab regions with active nuclear power development differ significantly from those in less wealthy Arab states that are concerned mainly with ensuring safe storage and disposal of spent radiation sources, NORM and other materials. In both cases, however, there is a strong interest in partnering initiatives that pool resources and benefit from economies of scale. Arius has also been involved in discussions in Vietnam and in Indonesia amongst countries in South East

Asia that are interested in moving into nuclear power and, accordingly, need to establish credible waste management strategies [16].

3.1.3. The ERDO Working Group

The European region was identified as the most promising starting place for concrete planning because a political framework already existed, the European Parliament had expressed positive views and the binding Waste Directive of the EC explicitly includes sharing facilities between Member States to be an acceptable approach to fulfilling waste management responsibilities. Accordingly, the ERDO-WG was formed with the mission of preparing the groundwork for a truly multinational waste management organization. The national waste management strategy favored by ERDO-WG members is a "dual track" approach in which a national disposal concept is worked on in parallel with working with partner countries to assess the feasibility of implementing shared multinational facilities. The dual track approach has been explicitly structured by the ERDO-WG.

> "*Ten EU countries have been involved in ERDO-WG activities. The ERDO-WG reacted to the publication of the EC Radioactive Waste Directive by preparing guidance for small EU Member States with small nuclear programmes and subsequently by submitting to EU governments structured proposals for a multinational European waste management organization*" [17].

Some of the ERDO-WG members have already taken a decision to include the dual-track approach as a part of their national waste management strategy. The key issues that will determine the success or otherwise of the ERDO initiative are the political and public acceptance of transferring spent fuel to another country and the economic benefits that can be derived from multinational cooperation. The organizational documents represent the first stages of the ERDO business plan, which will be required by any potential partner countries before reaching any final decisions on the establishment of a formal ERDO domiciled in one of the participating countries.

3.1.4. Status of the 'take-back' option

Another spent fuel management option that could contribute to global security and could ease nuclear problems for new entrants would be for nuclear-fuel suppliers to take back the spent fuel under a fuel 'leasing' arrangement, in which they would provide fresh fuel and take it back after irradiation, or for a large nuclear power program to accept spent fuel from smaller countries as an 'add-on' to its national inventory. They would then add this spent fuel to their own larger stocks to be stored for later disposal, or for reprocessing and recycling into new fuel. In fact, the political challenges for any potential service providers may be insuperable until such time as they have implemented their own disposal facilities Consequently, whilst conceptually attractive, leasing and take-back seem to remain as far out of reach of the emerging and nuclear power nations (and of the possible supplier nations) as it has been for the last 30 years. In addition, depending on how they were to be set up contractually, leasing arrangements might only solve part of the problem of spent fuel management, as long-lived wastes from recycling might be returned to the user countries for disposal.

4. PROSPECTS FOR FURTHER PROGRESS

Today, the right of nations to pursue a dual track approach which considers both national and multinational options is recognised widely. In addition, the potential benefits of smaller

programmes proceeding jointly have been emphasised by independent bodies, such as the National Academy of Sciences in the USA or the Royal Society in the UK, as well as by some larger programmes that will certainly need to implement their own facilities (e.g. USA and UK). The interesting question is whether the current initiatives and debates will lead to further progress towards implementation. We address this question below for different global regions.

4.1. Europe

The most structured approach currently is certainly that of the ERDO-WG in Europe. Advances have been made, as described above, and in principle the countries participating in the Working Group could decide at any time to establish an actual repository implementing organisation that would work in parallel to the national waste management organisations in other EU Member States. This step would involve appointing a small dedicated staff (perhaps including delegated experts from the participating countries) and setting these to work in a central location domiciled in one of the participant countries (without, however, prejudicing the later choice of repository sites). This first step will, however, be a major milestone and it is unlikely that it could be taken without intensive prior debate on issues such as the remit of the organisation, the siting strategy to be followed, the required funding levels and the allocation of costs to different sizes of user organisations. Current indications are that some smaller nuclear power programmes, such as the Netherlands and Slovenia, may be close to being able to make such commitments, but that others will require more time. Meanwhile, the ERDO-WG has become increasingly aware that its focus must be broadened beyond the long-term issue of final disposal and should cover more immediate strategic issues related to safe and secure management of all radioactive wastes. The WG recently, together with further EU Member States, such as Portugal and Greece, proposed to the EC that some funding should be allocated to support this. As pointed out earlier, however, EC support for cooperation on radioactive waste management currently focusses on R&D issues rather than strategic planning, so that funding avenues must be further explored.

4.2. Arab regions

As already pointed out, the challenges of ensuring safe radioactive waste management are different in differing Arab nations. Several non-nuclear countries in North Africa have expressed interest in introducing nuclear power and have also been involved, through the AAEA and the IAEA, in joint discussions on waste management. However, the current unrest across the region and the lack of human and financial resources make it unlikely that much progress will be made over the coming years. The most dynamic new nuclear power programme in the world at present is, perhaps, that of the UAE, where up to 8 nuclear power plants will be built. Work is in progress on the first four and ambitious start-up dates are scheduled. The UAE have been acting as a role model for new nuclear programmes in that they follow very strictly the advice provided by the IAEA in its milestones documents. The UAE have also publicly announced that they are following a ‘dual track’ disposal strategy, which keeps open both options – a national repository and a shared regional facility. Arius was contracted to examine in detail the implications of the regional approach.

An attractive concept for the six countries comprising the Gulf Cooperation Council region (which includes the UAE and Saudi Arabia) is the launch of a joint project looking into the feasibility of shared storage and/or disposal systems. Despite Jordan investigating a fuel send back arrangement as part of their power plant agreement with Russia, they will still require a geological reporistory for other long-lived wastes.

Arius has submitted to the Gulf Coordination Council proposals for an in-depth study of the benefits and challenges associated with establishing a multinational Waste Management Organisation (WMO) that would be co-owned by all interested States in the region.

4.3. Asia

In Asia, there are a few major nuclear power users (Japan, South Korea, Taiwan, and China) and many other nations that could potentially introduce nuclear energy. For the small programmes alone, there exists already one organisational framework within which the issue of cooperation on spent fuel and radioactive waste could be discussed. This is the ASEAN network, which includes Vietnam, Indonesia, Malaysia, Thailand, Philippines, Singapore, Cambodia, Brunei, Laos and Myanmar. In addition, Bangladesh has definite nuclear plans and Turkey, Mongolia and Kazakhstan, which straddle the Asian — European boundaries, have also all expressed interest in, or are already initiating, nuclear power programmes. Lastly, Australia has a traditional anti-nuclear power position, but returns repeatedly to the issue. There is also a network, the Japanese led Forum for Nuclear Cooperation in Asia (FNCA), which has participants from Australia, Bangladesh, China, Indonesia, Japan, Kazakhstan, Korea, Malaysia, Mongolia, the Philippines, Thailand and Vietnam. To date, however, only Vietnam and, perhaps, Malaysia from the ASEAN countries are sufficiently serious about nuclear power to be concerned already about spent fuel management options. What, then, are the prospects in Asia for multinational cooperation at the back-end?

4.4. Other global regions

The most obvious further region of the globe in which cooperation on spent fuel management could be valuable is in Central and South America. Mexico could ultimately send its spent fuel to the USA. Argentina and Brazil currently have nuclear power reactors in operation and have agreed to develop further nuclear power reactors jointly. In 2011, an agreement was signed under which they will jointly build two research reactors. Linking their spent fuel management strategies seems an obvious further step. In South America, Venezuela also has nuclear plans and has established a nuclear cooperation agreement with Russia. Chile has also expressed interest and has established cooperation with France. The opportunities for considering regional South American cooperation on management of spent fuel at the outset are apparent.

Finally, interest in introducing nuclear power to sub-Saharan Africa has been expressed at times by Kenya, Nigeria and Namibia. The obvious approach here would be for South Africa, the only experience nuclear power country in the region, to take a leading role in promoting cooperation, if and when the intention of these countries turn into specific plans.

5. CONCLUSIONS

This round-up of recent developments highlights a widespread interest in regional solutions at the back-end. In Europe, a good indicator of the further progress will be given over the next few years as EU Member States fulfil their requirements under the Waste Directive. It is certain that some Member States will include in their submissions a dual track approach.

> "*For the regional partnering approach to be as credible as a purely national strategy, the path towards establishment of a jointly owned European Waste Management Organisation should be clear*" [17].

The ERDO-WG provides a framework for European countries that wish to make progress in this direction. In the other global regions where shared solutions are currently being considered, the European approach might continue to be used as a role model, leading to further partnerships, e.g. one for the GCC, one for other Arab countries and one in Asia. There may be benefits in this approach for other regions too, such as Central and South America, or sub-Saharan Africa.

Despite the setback to global nuclear energy that resulted from the Fukushima disaster, many countries are expanding or introducing nuclear power programs. Moreover, large numbers of countries make use of nuclear technologies in medicine, research and industry that also lead to long-lived radioactive waste arisings. It is imperative that all such wastes are handled, treated and disposed of in a manner that minimizes safety and security risks. For many small countries, this can be best achieved by pooling their efforts in a regional, multinational framework. Nevertheless, large, advanced nuclear programs could help more than they currently do. The greatest help would, of course, be to take back or take away spent fuel from small countries. If this continues to be politically unacceptable, then moral, technical and even financial support to the multinational management and disposal initiatives of new nuclear power nations that purchase their expensive facilities from the large players would be valuable.

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INDUSTRY EXPERIENCE ON BACK END TRANSPORT

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Abstract

The back end of the nuclear fuel cycle covers the operations concerned with the spent fuel and wastes after the use of the fuel in the reactor. These materials are transported, treated, stored and eventually recycled or sent to a repository site. The transport of spent fuel and wastes is therefore crucial for a sustainable strategy on the long term spent fuel and waste management. The framework of regulations applying to the safe and secure transport of radioactive materials is a complex set of requirements, recommendations and standards, which apply at national or international levels, based on the IAEA Regulations for the Safe Transport of Radioactive Material (SSR-6). The World Nuclear Transport Institute (WNTI) represents the collective interests of the industry involved in the safe, secure, efficient and reliable transport of radioactive materials. One way the WNTI helps develop transport methodologies is through the use of Industry Working Groups, bringing together WNTI industry members with common interests, issues and experiences. The Back-End Transport Industry Working Group focuses on issues which include for instance dual purpose casks, transportable storage casks for spent nuclear fuels, including the very long term storage of spent fuel. In addition, the WNTI organised last December a workshop dedicated to the transport for the back end. After providing an overview of the framework of transport regulations for the spent fuel, the paper will deliver a broad overview of past experience, current issues and future challenges for the packaging and transport of spent fuel and wastes and the associated work of the WNTI and its members and discuss the importance of transport in a sustainable long term spent fuel and waste management strategy.

1. INTRODUCTION

Nuclear energy currently supplies more than 11% of the world's electricity.

> "*The use of nuclear reactors to produce electricity has required a wide range of radioactive material transports over several decades. These transports have supported all stages of the nuclear fuel cycle from uranium mining, to fuel processing and transport to reactor sites, to fuel reprocessing for recycling and spent fuel storage. The back end of the nuclear fuel cycle covers the activities taking place after the nuclear fuel has been used in the nuclear power plants*" [1].

The materials concerned by the back end include the spent fuel and wastes.

The industry has a long-lasting experience in the transport of radioactive materials, which spans for over 50 years. This excellent safety record has been noted in the International Atomic Energy Agency (IAEA) General Conference Resolution, Measures to strengthen international cooperation in nuclear, radiation, transport and waste safety,

> *"Recognizing that, historically, the safety record of civilian transport, including maritime transport, of radioactive materials has been excellent"* [2].

Every year, over 20 million consignments are routinely transported by road, rail, sea and air around the world. Most of these transports concern the radioisotopes for medical and industrial use. Five percent concern the transport for the nuclear fuel cycle, of which part regards the transport for the back end of the cycle.

After introducing the regulatory framework, this paper will review the industry experience in the safe, efficient and reliable transport in spent fuel and wastes, and will introduce the current topics discussed by the members of the World Nuclear Transport Institute (WNTI).

2. A STRONG REGULATORY FRAMEWORK

The transport of radioactive materials, including

> "*The transport of spent fuel and wastes, is strictly governed by an established system of international regulations and their adoption has led to an impressive record of safety. In over half a century, there has never been a single transport incident which has resulted in significant radiological damage to mankind or the environment.*
>
> *In 1961, the IAEA Regulations for the Safe Transport of Radioactive Material (SSR-6) or "IAEA Transport Regulations" [3] were published on the basis of expertise provided by Member States and international organisations. By 1969, the IAEA Regulations were adopted or used as a basis for regulations in many Member States. The principal international organisations having responsibility for transport by land, sea, and air have incorporated the IAEA Transport Regulations into their own regulations. The United Nations Recommendations on the Transport of Dangerous Goods [4] have always referred to the IAEA Transport Regulations, and fully integrate them. As a result, the Transport Regulations apply to the transport of radioactive material almost anywhere in the world."*[1]

The Transport Regulations are based on the fundamental principle that radioactive material being transported should be packaged adequately to provide protection against the various hazards of the material under both normal and potential accident conditions. Safety, therefore, relies primarily on the packaging whatever the transport mode. The prime objective is to protect people, property and the environment against the direct and indirect effects of radiation during transport.

The provisions of the IAEA Transport Regulations are not only reflected in the national requirements of Member States, but also in the regulations relative to each mode of transport as issued by international or regional bodies.

In 1965, the International Maritime Organization (IMO) published a major international instrument known as the International Maritime Dangerous Goods Code (IMDG Code) [5]. This Code is for the carriage of dangerous goods of any kind by sea. The IMO provisions for radioactive material are based on the IAEA Transport Regulations. In 1993, the IMO also established the Code for the Safe Carriage of Irradiated Nuclear Fuel, Plutonium and High Level Radioactive Wastes in Flasks on Board Ships (INF Code) to complement the IAEA Transport Regulations. Although the package design remains the primary safety measure, this Code introduces recommendations for the design of vessels transporting radioactive material. These complementary provisions address such issues as stability after damage, fire protection, and structural resistance. In January 2001, the INF Code was made mandatory and renamed the International Code for the Safe Carriage of Packaged Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Waste on Board Ships [6].

The International Civil Aviation Organization (ICAO) has responsibility for all aspects of international civil aviation. It develops regulations through the development of Annexes to the

1944 Convention on International Civil Aviation [7]. In 1981, the ICAO adopted Annex 18 covering the air transport of dangerous goods and, in addition, published a set of Technical Instructions (TI) [8] detailing the requirements for these transports. The TI contains a list of dangerous goods, as well as requirements for packing, marking, labelling and documentation fully consistent with the IAEA Transport Regulations. The International Air Transport Association (IATA), a trade association representing airlines, publishes annually the Dangerous Goods Regulations (DGR) [9] which is consistent with the ICAO TI as well as the IAEA Transport Regulations.

Land Transport Regulations tend to be regional rather than international. The United Nations Economic Commission for Europe (UN/ECE) publishes the European Agreement concerning the International Carriage of Dangerous Goods by Road (known as ADR) [10]. The IAEA Transport Regulations have been adopted to apply to the transport of radioactive material under the ADR. The Intergovernmental Organisation for International Carriage by Rail (OTIF) is responsible for the regulations concerning the International Carriage of Dangerous Goods by Rail (RID) [11]. These are included in the convention concerning International Carriage by Rail. The MERCOSUR/MERCOSUL Agreement of Partial Reach to Facilitate the Transport of Dangerous Goods [12] signed by Brazil, Argentina, Paraguay and Uruguay regulates the road and rail transport of dangerous goods, including radioactive material, between these States and is consistent with the IAEA Transport Safety Regulations.

3. INDUSTRY EXPERIENCE IN BACK END TRANSPORT

The World Nuclear Transport Institute (WNTI) was established in 1998, by BNFL of the United Kingdom (now International Nuclear Services — INS), Cogema of France (now AREVA) and the Federation of Electric Power Companies (FEPC) of Japan to represent the collective interests of the radioactive material transport sector and those who rely on safe, secure and efficient transport of radioactive materials. The WNTI is currently formed of 43 member companies.

> "*Through its non-governmental status, the WNTI supports the work of key intergovernmental organisations in promoting an efficient, harmonised international transport safety regime. The WNTI has consultative status with the IMO and observer status with the IAEA, in particular attending the General Conference and the Transport Safety Standards Committee (TRANSSC). In addition, the WNTI has consultative status with the United Nations Committee of Experts on the Transport of Dangerous Goods, observer status at the International Civil Aviation Organization (ICAO) Dangerous Goods Panel, Category B Liaison Membership with the International Organization for Standardization (ISO), and information status with the American National Standards Institute (ANSI) N14 Committee.*
>
> *The WNTI work benefits also from the on-going exchange between complementary industrial organizations and has established close links with FORATOM, the Nuclear Energy Institute (NEI), the World Institute for Nuclear Security (WINS) and the World Nuclear Association (WNA)*" [13].

In addition to a bottom-up approach, offering the industry feedback to regulators on operational experience accumulated through many years by the industry worldwide, the WNTI proposes a top-down approach to its members and stakeholders, by supporting a common understanding and interpretation of the Transport Regulations within industry, through the development of good practice documents, capturing industry's best practices, and

the organisations of thematic workshops, such as the one the WNTI organised last December on the future challenges in back end transport.

A dedicated working group was created by industry members of the WNTI to address subjects relating to the back end transport. Thanks to the collective work of WNTI members, the WNTI has produced several factsheets, information papers and good practice guides relevant to the transport of spent fuel and wastes. These documents and more information are freely available on the WNTI website [14].

The workshop on the challenges in back end transport organised by the WNTI last December saw the participation of a large range of stakeholders involved in the transport and storage of nuclear materials worldwide. The main conclusions of the workshop are described below.

4. FUTURE CHALLENGES IN THE BACK END TRANSPORT

Transport is a key element of the nuclear fuel cycle; it has sometimes been described as the 'blood' of the nuclear fuel cycle. Therefore transport should form an integral part of any strategy for the interim and longer term management of spent fuel and wastes. Considering transport at an early stage in a back end management strategy should ensure a safe, secure and efficient transport and interface between the transport and storage activities.

Firstly, it is important to consider the environment in which the transport activities are taking place. The environment is changing and these developments should be considered when developing strategies for the future. These changes are not specific to transport and could actually apply to storage too. Changes that the back end transport should face in the future include:

- The evolution of technical knowledge;
- The evolution of tools (computational tools, big data, etc.);
- The evolution of knowledge, organisations and practices, as practice benefits from lessons learnt from experience for instance.

A large number of stakeholders should be taken into account when considering the environment for the transport activities. These include back end policy makers and implementers, Competent Authorities, Technical Support Organizations and industry.

Challenges faced by transport can be principally grouped under six items:

- Challenges in packing for transport and storage;
- Challenges in packaging licensing and transport approvals;
- The need to prepare now for tomorrow;
- Demand for packages and transport;
- Adequacy between Transport Regulations and real conditions of transport and handling;
- Appropriate level of security.

The challenges in packing for transport and storage can be formulated through the following question: How to ensure safe, secure, efficient and reliable packaging of spent fuel and wastes for both transport and storage? For instance, information on waste streams would help facing the challenge.

Challenges in packaging licensing and transport approvals would address the issue of transport after long term storage. Solving these challenges would benefit from harmonization of the regulatory framework for transport and waste.

In order to prepare for tomorrow, a clear long term strategy, the importance of knowledge management and a long term stability of the regulations should contribute to a safe, secure and efficient transport.

This preparation would in turn help design packages which are fit for purpose, in adequate number, as the current fleet and model could be renewed.

While security has always been addressed in the transport of spent fuel and wastes, it has become a topic of greater attention in the last fifteen years. Security recommendations have been reviewed and implementation documents further developed. The IAEA Nuclear Series Nr. 13, Recommendations, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5) [14] addresses transport. It is completed in particular by Security in the IAEA Nuclear Security Series No. 9, Implementing Guide, Transport of Radioactive Material [15]. In regards to security, the WNTI suggests that a graded approach should be preferred.

Some of the challenges will be further detailed in the coming paragraphs.

4.1. The '4Cs'

The '4Cs' represent four key elements of the management of spent fuel and wastes which should be considered when addressing transport. The four components are:

- Categorisation: products' groups such as irradiated fuel; unirradiated fuel; encapsulated casings; unencapsulated casings; package internal furniture; reprocessed product; reprocessing waste; processing intermediate; pond/pool sludges; decommissioned plants; research materials; process/decommissioning wastes;
- Conditioning: such as high level wastes, intermediate level waste and low level waste;
- Characterisation: facility characterisation; physical characteristics of the plant; waste categories and volumes; radiological and non-radiological contamination; and
- Classification: what is being packaged/transported.

Quality management is of particular importance for the classification, as is the alignement of classification with the Transport Regulations in order to ensure the optimal selection of package type for the material transported. The classification will ultimately allow the determination of the correct package type and conditions of acceptance.

4.2. WNTI 'Good Practice Guide on Inventory Principles'

In order to support packaging for transport, the WNTI with the support of its members is producing a good practice guide, WNTI Inventory Principles — A move towards reliable packaging and transport data [16].

This guide proposes nine principles which are critical for packaging, transport, treatment and disposal. They are: training, accuracy and review, auditability, assurance – nuclear site reviews, fit for purpose data, dedicated resources, dynamic waste inventory, clarity and alignment.

The guide will be freely available on the WNTI website in the coming weeks.

4.3. Challenges in the regulatory framework for transport and wastes

Challenges in the regulatory framework can be seen in four dimensions:

- Harmonisation between countries;
- Harmonisation between countries' implementation of the regulations;
- Harmonisation between wastes and transport regulations;
- Harmonisation between safety and security.

The IAEA will play a pivotal role in tackling the potential issues of interface between the various aspects of harmonisations. In its role as 'world's centre for cooperation in the nuclear field', the Agency acts as a key facilitator for the discussions between the various stakeholders involved in these areas requiring harmonisation.

Equally, since industry implements the regulations set by Competent Authorities around the world, the World Nuclear Transport Institute is well placed to provide the return of experience of the Regulations to Regulators in meetings of the Agency for example.

4.4. Guidance on package design approval

The IAEA Transport Regulations, with the associated guidance, and in particular the Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2012 Edition), IAEA Specific Safety Guide SSG-26 [17], provide the requirements and guidance for the design of packages.

In addition to these documents, a cross-boundary action to harmonise the application for design approval has been helped by the Competent Authorities action to develop common guides. In North America, RD-364: Joint Canada — United States Guide for Approval of Type B(U) and Fissile Material Transportation [18] can be mentioned. In 2006, a bilateral agreement between France and the United Kingdom Competent Authorities was signed for mutual recognition of packages requiring multilateral approval.

These early examples have been expended with the Technical Guide 'Package Design Safety Report (PDSR) for the transport of radioactive material' [19] developed by the competent authorities and their support organizations responsible for the transport of radioactive material of Belgium, France, Germany, Spain and the United Kingdom and the World Nuclear Transport Institute (WNTI) and AREVA as industry representatives. The IAEA version of the PDSR is based on the experiences of these documents.

4.5. Dual Purpose Casks (DPC)

Dual purpose casks should be part of an integrated back end strategy, which considers the various options for the back end including, reprocessing, interim storage in pools, dry interim storage and final repository storage.

Through the IAEA, experts have worked in joint working groups to develop an integrated safety demonstration for DPCs. The IAEA Joint Working Group on Guidance for an Integrated Transport and Storage Safety Case for Dual Purpose Casks for Spent Fuel has developed a (draft) document and reported recommendations to TRANSSC and the Waste Safety Standard Committee (WASSC).

During the review of the Transport Regulations, a dedicated working group proposed changes to the Transport Regulations to address the DPCs. This proposal should include a gap analysis for transport after long term storage. It has been submitted as part of the 2015 review cycle and will be considered in TRANSSC in June.

4.6. Transports of large objects

The back end transport also includes the transport of decommissioned parts of nuclear power plants. These large objects, such as team generators, have been the subject of proposed changes to the Transport Regulations. A working group was gathered under the auspices of Canada, and included representatives of the WNTI, to propose a new text addressing these large items. This new text is part of the proposals to be reviewed by TRANSSC this June, as part of the 2015 review cycle.

In addition, the WNTI published a factsheet aimed at presenting a jargon free description of these transports [20]. This factsheet is freely available from the WNTI website.

Also to note, a particular point to take into account when transporting these components is early route planning as these components are usually of very large size.

4.7. Fissile exception requirements

In the 2012 Edition of the Transport Regulations, requirements for fissile exceptions have been reviewed. This revision has sometimes been received as requiring further support in its interpretation.

For this matter, the WNTI has issued an information paper [21], which is available from the WNTI website.

In addition, the IAEA is also working on a guidance document, on the basis of the WNTI information paper and other documents.

5. CONCLUSION — KEEPING THE ROADS OPEN

Transport is the only nuclear activity happening in the public domain. While industry experience in transporting has been spanning over several decades, it is important that routes remain open to our transports. Communication should be maintained and a stakeholder engagement strategy should be planned from the start, when a back end strategy is developed.

For that purpose, the public needs to be well informed to accept the transports. In this aim, communication is key. In 2014, the WNTI issued a good practice guide on communicating for transport. This guide is available from the WNTI website.

While transport's safety record is outstanding, it is important that it is not taken for granted. Emergency preparedness and response should be carefully planned, with regular table top and full size drills.

The acceptance of international back end transport is usually also facilitated thanks to cross-national nuclear liability and insurance frameworks, such as the Paris and Vienna Conventions, and the Convention on Supplementary Compensation, which apply to transport equally as to nuclear installations but also include specific provisions for transport.

In conclusion, transport should be considered from the early stage of the development of a strategy for the management of spent fuel and wastes, considering the interface between transport and storage.

The industry is well equipped to face the challenges to come for the transport of back end materials, and through the WNTI will continue to gather the experience in the transport, with a view to ensure safe, secure, efficient and reliable transport of back end materials.

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LICENSING OF THE DEEP GEOLOGICAL DISPOSAL OF SPENT NUCLEAR FUEL IN SWEDEN

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Abstract

The Swedish Nuclear Fuel and Waste Management Co., SKB, was established in the 1970s and is owned by the companies that own the Swedish nuclear power plants. SKB operates facilities for interim storage of spent nuclear fuel (Clab) in Oskarshamn and a final repository for low level waste (SFR) at Forsmark in Östhammar as well as a transportation system for the radioactive waste and spent nuclear fuel. SKB also has the task to develop and implement the systems for final disposal of spent nuclear fuel and long-lived waste. This encompasses an extensive Research Development and Demonstration-programme that includes large scale facilities for canister technology development, buffer and backfilling technology and an underground research laboratory (Äspö) in Oskarshamn. On 16 of March 2011, an important milestone was reached when applications for licences to build a Spent Fuel Repository in Forsmark and an encapsulation plant in Oskarshamn were submitted to the Swedish Radiation Safety Authority and the Land and Environmental Court in Stockholm. The applications are based on results from extensive research, site investigations and analyses that have been going on for decades. The applications show that the legal requirements on the facilities can be fulfilled for a KBS-3-system where the spent fuel is placed in copper-steel canisters that are deposited at about 500 m depth in crystalline rock at the selected site at Forsmark. The repository will remain safe throughout the operating life and after closure if constructed and operated according to the specifications laid down in the plans and documentation of the licensing application.

1. INTRODUCTION

The licensing processes according to the Act on Nuclear and the Environmental Code, is now on-going. The Swedish Radiation Safety Authority has so far asked SKB for additional information regarding e.g. copper corrosion, canister design, groundwater flow modelling and biosphere modelling. The Land and Environmental Court has received input from many stakeholders including the concerned municipalities, environmental groups and many others regarding their requests for additional documentation or explanations by SKB. These requests cover issues concerning site selection, alternative disposal methods, environmental consequences due to groundwater discharge and increased traffic, just to name a few. The Court will now make its own judgement and prepare the formal demand on SKB for additional information.

It is presently estimated that the final statements to the government by SSM and the Environmental Court will come early 2017. Political decisions, provided that the safety authority and the court have given a green light, by the municipalities of Östhammar and Oskarshamn could then possibly be taken in 2018 followed by a final decision of the Swedish government, whereby construction of the repository could start around 2020. The local public opinion is still very positive with about 80% of the population being positive to the plans of SKB. A broad dialogue with all stakeholders will continue to be of high priority for SKB in order to arrive at a situation in a few years from now where all necessary decisions can be taken based on a credible and transparent understanding of the critical issues.

While the licensing process is still under way, SKB continues to prepare for implementing the KBS-3-system. Among other things, this entails building up an organization and industrial production system for all parts of the final disposal process.

2. THE REGULATORY FRAMEWORK

There are four main legislative instruments that regulate final disposal of waste from nuclear power:

- The Act on Nuclear Activities;
- The Radiation Protection Act;
- The Act on Financing of Management of Residual Products from Nuclear Activities;
- The Environmental Code.

Any new nuclear facility must be licensed according to the Act on Nuclear Activities and the Environmental Code. In both cases the government grants the license on the basis of recommendations and reviews by the competent authorities. The authorities are engaged in the Environmental Impact Assessment (EIA) in connection with the site selection process.

The local communities in Sweden have a high degree of self-governance. They can veto a decision to build a nuclear facility in their municipality. The veto can be over-ruled by the national government under extreme circumstances of national interest, but this is in practice not an option. The veto right facilitated for the local communities to interact with SKB and its owners without risk of becoming committed against their will. They could always pull out of the site selection process.

The Act on Nuclear Activities in 1984 included a formal requirement that the owners of nuclear reactors also submit an up-dated RD&D programme every third year. Since 1986 SKB has submitted nine RD&D reports to the Government, KBS-3 being the main alternative for the disposal of SNF.

3. THE SITE SELECTION PROCESS

Between 1976 and 1983 SKB has been carrying out an extensive mapping of the Swedish geology.

> "*The intention was not to find a location for a repository, but primarily to obtain data from large depth in different areas spread all over the country. On the basis of area survey studies SKB concluded that it is possible to find many places in Sweden with a suitable geology to host a final disposal for spent nuclear fuel*" [1].

Between 1992 and 2000 SKB performed feasibility studies in no less than eight municipalities. It has been decided that site investigations will be done in both candidate municipalities, Östhammar and Oskarshamn. The third municipality that SKB suggested for site investigations, Tierp, voted against further cooperation with SKB in their municipality council. Between 2000 and 2009 SKB performed drillings to 1000 meter depth, gathered data that later were compiled in an EIA. Safety analyses have also been performed in both sites. In preparing the EIA report and the rest of the investigations needed to make the final selection of the site, extensive consultations with all stakeholders took place [2, 3].

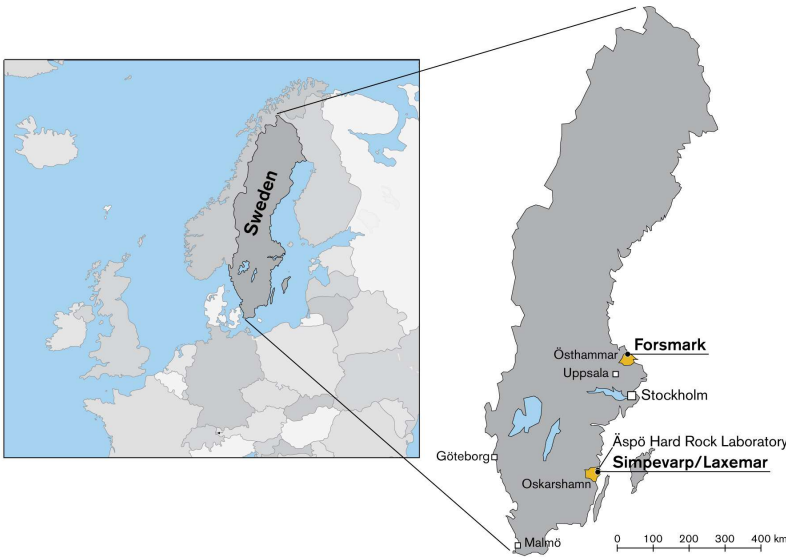

FIG. 1. Repository sites.

2009 SKB decided that the location of the final disposal will be at Forsmark in the municipality of Östhammar and that SKB will build an encapsulation facility in Oskarshamn.

4. SUBMITTING THE APPLICATION

"*The site selection for a final repository is a milestone for the Swedish nuclear waste programme. In March 2011 an application for permits to build the final repository and the encapsulation facility was submitted to the government [4].*

The applications according to the Act on Nuclear Activities and to the Environmental Code are formally the bases for two separate legal examinations, and we therefore had to draw up two documents. The content of the documents is to a great extent the same, but there are also differences since there are number of issues that are considered only according to one of the regulations.

The Environmental Court will after some preparatory procedures hold a main hearing for all stakeholders. The Court will later submit its statement to the Swedish Government. If the application is accepted, the Court will hold a new hearing. Thereafter the Court will grant permit and stipulate conditions pursuant to the Environmental Code.

In Parallel, The Swedish Radiation Safety Authority, SSM, is expected to submit its statement according to the Act on Nuclear activities. SSM will subsequently stipulate conditions pursuant to the relevant regulations" [1].

The government would have, prior to the authorities issuing their conditions, heard the municipalities on their veto decisions.

5. CURRENT STATUS OF THE REVIEW

"*The factual review by the authorities started at the end of May 2011. In conjunction with this, the application documentation was also sent out to experts in a broad national referral, both by SSM and by the Land and Environmental Court. Environmental organisations, concerned municipalities and county*

administrative boards, universities and colleges, other authorities and more were allowed to give their opinions about SKB´s application.

At SSM, the review work on the application is also divided up between its own personnel and hired external experts. The Authorities have requested supplementary information and clarification from SKB under the last three years since their review started. SKB has also been offered the opportunity to respond to the statements that different stakeholders have expressed.

The Government also requested that an independent international review of the applications should be made. This was carried out by the OECD´s Nuclear Energy Agency (NEA) between May 2011 and June 2012 [5]. The team of experts reviewed SKB´s safety analysis on the long term radiation safety as well as the selection of site and the choice of method, KBS-3. In their final statement you can read 'from an international perspective, SKB´s post closure radiological safety analysis report, SR-Site, is sufficient and credible for the licensing decision at hand. SKB´s spent fuel disposal programme is a mature programme… at the same time innovative and implementing best practice capable in principle to fulfil the industrial and safety related requirements that will be relevant to the next licensing step"" [1].

6. NEXT STEPS TOWARDS IMPLEMENTATION

In parallel with the licensing work, SKB has to plan for the phase after the permit to construct the final disposal for spent nuclear fuel has been granted. Main items of work are:

- To build up the organisation responsible for on-site preparations and detailed design and in charge of the forthcoming construction phase in Forsmark;
- To continue the technology development of the KBS-3 system with a detailed design, industrialized production and control systems;
- To further strengthen the scientific basis on processes of importance for long term safety;
- To keep and develop public confidence in the project of spent nuclear fuel management.

7. CONCLUDING REMARKS

There are numerous challenges facing all involved parties in the licensing process. Taking into account that the review seems to be stretched over time, new regulations might get adopted leading to changes in requirements. That alongside generational renewal of experts both at SKB but also within the safety authority, SSM, requires attention. This since SSM and SKB are interacting for a first of a kind facility with no previous references.

There are still a number of scientific issues that need further attention to strengthen the scientific basis on processes of importance for long term safety and to reduce uncertainties and facilitate better estimates of risk in future safety assessments. The issues will be addressed by the R&D programme during the coming years.

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CHALLENGES IN DEVELOPING THE BASIC DESIGN OF THE KBS-3 SYSTEM INTO A QUALIFIED AND INDUSTRIALLY VIABLE OPERATION

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Abstract

The programs for final disposal of spent nuclear fuel are similar in Sweden and Finland, and there has been extensive cooperation between the waste management organizations in the two countries over the years. This cooperation will now be deepened, aiming when possible for the same technical design. While a technically feasible reference design and layout is presented for the repositories in the two countries, detailed designs adapted to an industrialized process designed to fulfilling specific requirements on quality, cost and efficiency need still be developed. Also the repository layout needs to be adapted to the local conditions found when constructing the repository at depth. Both SKB and Posiva have developed design requirements and other conditions and presented these to the designer. However, the formulation of requirements such that they lead to designs that both meets long term safety and can be verified is not trivial and revision and harmonization between the organizations is needed. Essentially the detailed technical design need to be completed in time for the detailed design of the planned facilities in the KBS-3 repository system, i.e. the encapsulation plant, the facility for buffer and backfill bentonite component production and the underground repository.

This paper has been removed from the proceedings as it has been published elsewhere. Interested readings should refer to the following reference.

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SAFETY ASPECTS OF SPENT FUEL MANAGEMENT

(Session 3)

Chairpersons

F. BESNUS

France

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KEYNOTE SPEECH 03: SAFETY AND TECHNOLOGICAL ASPECTS OF SPENT FUEL MANAGEMENT

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Abstract

For an open or closed fuel cycle, the management of spent fuels is characterized by several phases until final storage. Between these phases, some operations have to be performed, such as conditionings, transports or recycling. For the safety, a global strategy for spent fuel management must be developed. This strategy is important to verify the consequences of choices made at one stage on the following stages, but also to define requirements of the parameters that will be used to justify the safety of the following stages and to anticipate knowledge needs. Furthermore, the strategy implantation requires significant time for the design, the licensing and the commissioning of facilities (typically more than 10 years for each facility). A strategy must be elaborated to give time for this process and for the dialogue with stakeholders. Finally, it's important to check that the capacities of each stage are sufficient. Regarding safety, the facilities for spent fuel management use different processes with specific safety issues. Moreover, the designs of installations with the same objectives can be different. This diversity leads to produce a safety analysis for each installation. This analysis should be at least based on shared objectives and principles. The SSG-15 guide of IAEA is a good tool for this. This approach requires also particular safety skills for the operators, the safety authorities and their technical supports. These skills are varied, take time to be developed and must be maintained. Another aspect of the safety of such installations is that the analysis of severe accidents is less developed than for NPPs. The feedback of Fukushima accident shows the necessity to work on some severe accidents. In addition to the vigilance about the normal operating conditions of these facilities, this approach should be continued because it provides safety improvements. Other challenges are the ageing of these installations and how to improve their safety. Indeed, they will be operated during a very significant time and are generally complex to modify. Periodic safety reviews are a good mean to define improvements. In any case, the operating experience feedback should be shared. In this regard, the ability to monitor the safety functions and equipment is crucial. In the parts of installations where irradiated materials are handled, controls are difficult to be performed. This issue must be taken into account in the design and, for the older installations, require developments.

REGULATORY EXPERIENCES FROM IMPLEMENTATION OF SNF DISPOSAL PROGRAMME FROM SITE SELECTION TO CONSTRUCTION OF DISPOSAL FACILITY — THE FINNISH CASE

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Abstract

In 1983, the Government made a strategy decision on the objectives and target time schedule for the research, development and technical planning of nuclear waste management. The decision gave the timeline for the milestones on the way to an operating disposal facility by around 2020. This has been a long-term project with over 30 years of parallel development of the project and the regulatory approach to SNF management. Posiva, the implementer, submitted the construction license application (CLA) and supporting documentation to the authorities at the end of 2012. The Radiation and Nuclear Safety Authority of Finland (STUK) started the review and assessment with an initial review in early 2013 and STUK's safety evaluation of Posiva's CLA was presented to the Finnish government in February 2015. The CLA and STUK's review cover aspects of safety, security and safeguards. The scope of application is operational safety of facilities and post-closure safety of disposal. In parallel with the CLA review process STUK has implemented an inspection program focusing on the applicant's management system and readiness for construction. To support regulatory decision making, STUK has used a wide range of national and international experts in the CLA review process.

This paper has been removed from the proceedings as it has been published elsewhere. Interested readers should refer to the following reference.

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DEVELOPMENT OF LONG-TERM SAFETY REQUIREMENTS FOR AN ALTERNATIVE DESIGN VARIANT (KBS-3H) FOR SPENT FUEL DISPOSAL

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Abstract

In 2012, Posiva Oy submitted a construction licence application for a spent nuclear fuel disposal facility to be constructed at Olkiluoto, Finland. A safety case (TURVA-2012) was compiled to support the licence application. The disposal concept is based on the KBS-3 method. The reference design is the KBS-3V design, where the spent nuclear fuel canisters are emplaced individually in vertical deposition holes positioned along deposition tunnels. Posiva Oy is also studying, in collaboration with its Swedish counterpart SKB, an alternative design variant, KBS-3H, where the canisters are emplaced horizontally in 100−300 m long deposition drifts. This design variant was also included in the application as a potential alternative. In order to compare these two alternatives, a safety case, following Finnish regulatory requirements, is being produced for the KBS-3H design. The main objective is to determine whether KBS-3H can be shown to fulfil the long-term safety requirements with the same level of confidence as for KBS-3V. To this end, long-term safety related requirements specific to the KBS-3H design are being defined following Posiva's requirements management system (VAHA). VAHA includes five levels of requirements spanning from legal and stakeholders' requirements (level 1) to system requirements (level 2), performance targets and target properties (level 3), design requirements (level 4) and finally design specifications (level 5). The level 1 requirements, since they stem from laws and regulations, are identical for both designs. At lower levels, the differences in the designs have an increasing effect on the details of the requirements and design specifications. The set of release barriers is partly different in the two designs, as are the types and dimensions of the emplacement areas and their construction methods. The development of the KBS-3H-specific requirements starts by defining the barriers of the KBS-3H design and assigning safety functions for the individual barriers. The safety functions will then give rise to performance targets, and subsequently to the more detailed requirements and specifications at lower levels. The safety case for KBS-3H will then evaluate whether the horizontal design fulfils these requirements. The requirement definition includes interesting aspects related to the fact that KBS-3H has been developed over decades in parallel to the reference design KBS-3V, and it includes several novel solutions and unique components not included in KBS-3V. The iteration among requirements formulation, safety assessment and design development is particularly visible in this project.

1. INTRODUCTION

The KBS-3 method has been developed and analyzed for decades in Sweden and Finland as a potential solution for disposal of spent nuclear fuel from domestic nuclear power plants. The reference method is KBS-3V with vertical emplacement of canisters in individual deposition holes positioned along deposition tunnels (Fig. 1, on the left). Posiva Oy (in Finland) and the Swedish Nuclear Fuel and Waste Management Company (SKB) are also developing an alternative design variant, KBS-3H, where the canisters are emplaced in horizontal deposition drifts (Fig. 1, on the right). The deposition drifts will replace the deposition tunnels and deposition holes employed in the KBS-3V design, thus reducing significantly the required volume to be excavated and, subsequently, backfilled. Between 2003 and 2007, the first safety assessment for KBS-3H was produced by Posiva Oy and SKB, summarized in Ref. [1]. At that time, Posiva's requirements management system (VAHA) was not yet in place, and the safety assessment was based on the concepts of safety function indicators and associated criteria following the methodology adopted in the Swedish SR-Can safety assessment [2].

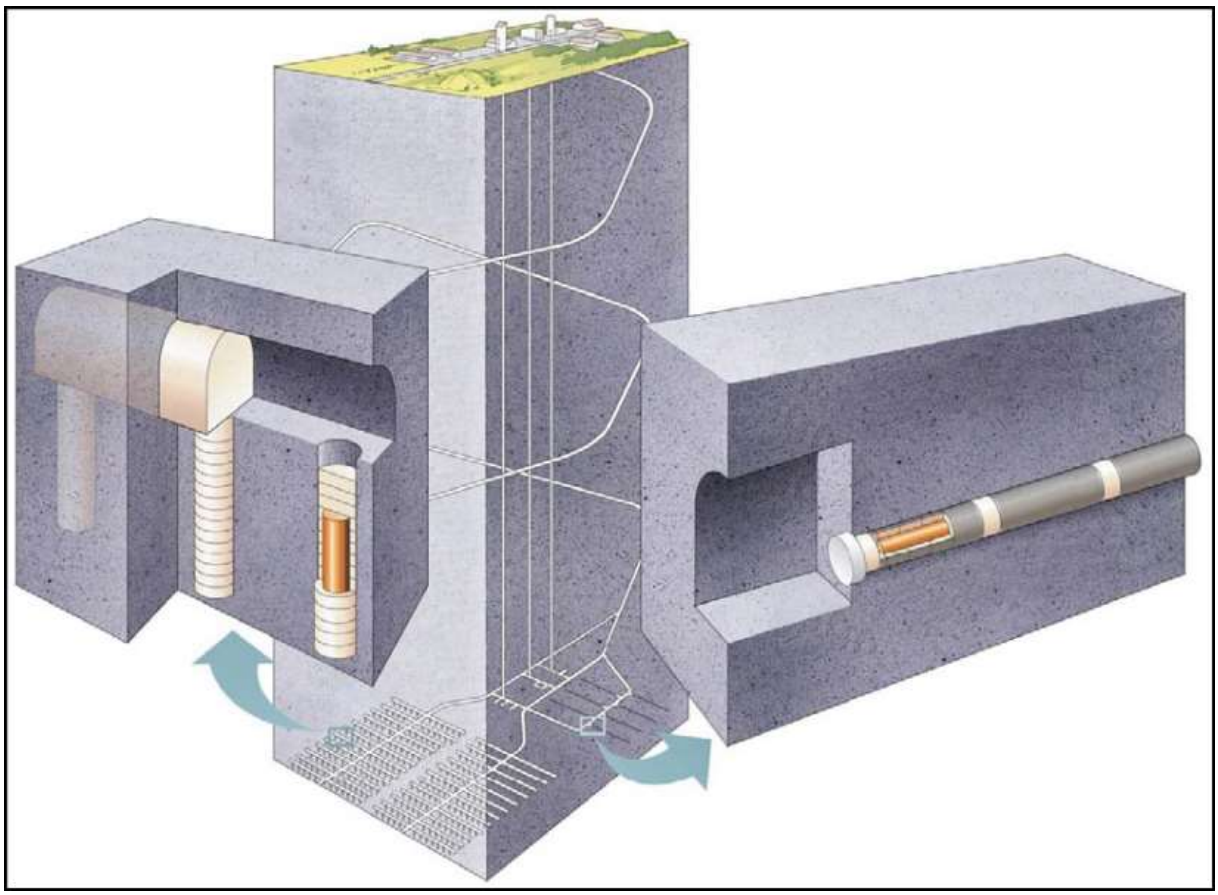

FIG. 1. Schematic drawing of the KBS-3V reference design (to the left) and the KBS-3H alternative (to the right). Courtesy of SKB, Illustrator: Jan Rojmar.

Posiva's construction licence application for a spent nuclear fuel disposal facility to be constructed at Olkiluoto − submitted in 2012 and supported by a safety case, TURVA-2012 [3] − was based on the KBS-3 method. The reference design was the KBS-3V, and the KBS-3H design variant was included in the application as a potential alternative, described in a topical report [4]. After reviewing the safety case, the Radiation and Nuclear Safety Authority (STUK) stated that they have not found any factors that would prevent the KBS-3H design from fulfilling the safety requirements, and a change of design from KBS-3V to 3H could be handled as a change in the construction licence currently in process [5].

2. CHARACTERISTICS OF THE KBS-3H DESIGN VARIANT

The horizontal KBS-3H deposition drifts are planned to be 100−300 m long, circular drifts with a diameter of approximately 1.85 m, and they will be excavated mechanically, not by drill and blast as the KBS-3V deposition tunnels. Each canister is pre-packaged along with bentonite buffer in a metallic shell to facilitate and control the canister and buffer installation in the drift. The canister, buffer and metallic shell are referred to as *supercontainer*. The metallic shell is currently planned to be made of titanium to minimize any harmful interactions with the bentonite barriers of the disposal concept. The supercontainer shell is perforated to allow the buffer to swell out of the supercontainer as it takes up water. About 20−30 supercontainers will be emplaced horizontally in the drifts, depending on the drift length and canister type, separated by additional bentonite buffer components, *distance blocks*. The centre-to-centre distance between adjacent supercontainers, and thereby also the

length of the distance blocks, is mainly determined based on temperature limits set for buffer performance and thermal properties of the fuel, engineered barrier system (EBS) and rock. In addition to providing thermal separation, the distance blocks are also intended to separate the supercontainers hydraulically from each other.

In addition, bentonite is planned to be used as various *filling components* in the drifts, at locations that are unsuitable for supercontainers and distance blocks, either due to long-term safety or operational reasons. Filling components are used, for example, at positions of unacceptably high groundwater inflow, and immediately adjacent to the plug(s) in the drift. Each drift will be sealed with a *drift plug* made of titanium, and all drifts longer than approximately 150 m will also be divided into two compartments using a titanium *compartment plug*. In addition to these, there are some minor components in the drifts, for example the pipe system related to artificial watering and air evacuation procedures according to the current KBS-3H reference design, DAWE (Drainage, Artificial Watering and air Evacuation). The majority of these pipes will, however, be removed before closure. The DAWE system is a novel solution to accelerate the saturation and swelling of the bentonite components, resulting in a more uniform and predictable initial state of the drift components.

The components of the KBS-3H design variant and the artificial watering procedures are described in more detail in Ref. [6].

3. LONG-TERM SAFETY REQUIREMENTS AND THE VAHA SYSTEM

The VAHA ('Vaatimusten Hallinta', i.e. 'Requirements Management' in Finnish) system was officially launched in 2007 to develop the requirements database [7]. Requirements in VAHA are hierarchically organized according to five levels:

- Legal and stakeholders' requirements (level 1);
- Requirements applying to the whole disposal system and safety functions for the barriers (level 2);
- Barrier-specific long-term performance targets and target properties (level 3);
- Barrier-specific design requirements (level 4);
- Barrier-specific design specifications (level 5).

The level 1 requirements arise from laws, decrees, decisions-in-principle and other high-level *stakeholder requirements*. Level 2 consists of *system requirements* as defined by Posiva on the basis of the requirements from Posiva's owners and the regulatory requirements listed on level 1. The level 2 requirements define the EBS components and the functions of the EBS and host rock [7]. The system requirements also define, for example, the maximum quantity of spent nuclear fuel to be deposited.

Safety functions are high-level descriptions of the role of each barrier in providing long-term safety. *Performance targets* for the engineered barriers and target properties for the host rock are derived from each safety function, as required in the Finnish regulations (STUK YVL D.5, paragraph 407). The actual *design requirements* and *design specifications* are ultimately defined to enable the fulfilment of the performance targets during the long-term evolution of the disposal system.

The performance targets have been set to provide the link between the initial state and long-term safety. The performance targets form the basis for the definition of design requirements. The initial state of the disposal system can be affected through the design requirements and system implementation practices (up to the closing and sealing of each deposition hole and

tunnel in KBS-3V, or deposition drift in the case of KBS-3H); the degree to which the performance targets are met is evaluated through performance assessment (See Ref. [8] for KBS-3V, under preparation for KBS-3H). In this sense, VAHA is a tool to translate high-level, long-term safety related requirements into design specifications to follow during design, construction and operation of the disposal facility.

> "*The design basis refers to the current and future environmentally induced loads and interactions that are taken into account in the design of the disposal system, and, ultimately, to the requirements that the planned disposal system must fulfil in order to achieve the objectives set for long-term safety. The Design Basis report (Ref. [7] for KBS-3V, under preparation for KBS-3H), which is part of the safety case, documents the safety bases of the requirements at levels 3 and 4 in VAHA.*
>
> *In defining the design basis, Posiva must, by regulation, on the one hand assess the likelihood of different scenarios and, on the other hand, identify those deemed reasonable, and assess those that may be possible but are considered highly unlikely. Although only scenarios deemed reasonable are used as design basis scenarios, scenarios that are deemed unlikely also need to be assessed in the safety case. The performance assessment evaluates the fulfilment of the performance targets by the proposed design. Uncertainties identified in the performance assessment drive the formulation and assessment of scenarios potentially leading to radionuclide releases*" [7].

4. PROCEDURE FOR DEFINING THE REQUIREMENTS FOR KBS-3H

> "*In order to evaluate whether the KBS-3H alternative can fulfil the long-term safety requirements with the same level of safety as KBS-3V, a safety case is being produced for the Olkiluoto site as a joint project between Posiva and SKB during 2014−2016. As the main objective is to allow the safety of a KBS-3H repository to be compared with that of a KBS-3V repository, the safety case aims to use, as far as possible, the same assumptions as were used in the TURVA-2012 safety case*" [3].

for KBS-3V, but modified to apply to the special characteristics of KBS-3H and, when necessary, to new data and improved methods.

The long-term safety requirements defined for KBS-3H are intended to follow the structure of VAHA, Posiva's requirements management system. The VAHA requirements have originally been developed for the KBS-3V design, thus the requirements need to be adapted to account for, e.g., the unique components of the KBS-3H variant. The level 1 requirements, since they stem from laws and regulations, are identical for both KBS-3V and 3H. Also, the safety functions, target properties of host rock and long-term performance targets remain largely similar to those defined for KBS-3V. At lower levels, the differences in the designs have an increasing effect on the details of the requirements and design specifications.

4.1. Barriers and safety functions

The first task in the development of KBS-3H-specific requirements is to define the barriers of the KBS-3H design and to assign safety functions to the individual barriers. These need to be defined taking into account the level 1 requirements which are rather general. The barriers and their overall function are defined on level 2. The safety functions will then give rise to

performance targets and target properties (level 3), and subsequently to the more detailed design requirements (level 4) and design specifications (level 5).

In the safety case project, the barriers of the KBS-3H concept are currently defined to be: (1) canister; (2) buffer (including the buffer in the supercontainers and the buffer in the distance blocks); (3) filling components; (4) compartment plugs and drift plugs; (5) closure; and (6) host rock. Of these, only the canisters are exactly identical in both concepts. The closure components, i.e. the backfill materials and plugs used in the disposal facility outside deposition tunnels/drifts, are largely identical, but there are some geometrical changes in the immediate vicinity of the deposition drifts, although the materials and most of the requirements can be assumed to be the same. The site is naturally the same, but there will be some changes in the host rock requirements due to the different dimensions of the repository openings and the different construction methods. Filling components and compartment and drift plugs are unique to the KBS-3H concept. It should be noted that the supercontainer shell is not considered to have a long-term safety function, as it is needed only for the operational phase, and the bentonite within the shell is meant to swell and extrude through the perforations soon after supercontainer emplacement. It will, however, have to be designed so as not to impair the barriers, meaning that there will be lower-level requirements for, e.g., its material selection.

Safety functions are then defined for all barriers. The safety functions are partly common with KBS-3V (see Table 1; the requirements common for 3V and 3H span over both columns) and partly based on the safety functions and safety function indicators defined in earlier project phases for KBS-3H (e.g. Ref. [6]), with some new considerations suggested. The safety functions on the right hand column of Table 1 are specific to 3H. In addition to the definition of the EBS, VAHA level 2 requirements for 3V also contain general system requirements, as mentioned in Section 3, which are considered to apply to 3H as well.

4.2. Performance targets and target properties

Based on the defined barriers and their safety functions, the next level comprises the set of performace targets (for the engineered barriers) and target properties (for the host rock), which also need to take into account the loads to which the disposal system may be exposed to during its long-term evolution. The starting point is the VAHA level 3 requirements for KBS-3V, which needs to be adapted to 3H. The performance targets for canister and closure are directly applicable to 3H, although the closure definitions need to be slightly modified due to the geometrical differences mentioned above. For the host rock, the actual target properties are the same for 3V and 3H, although there are changes in terminology due to the different repository geometries. The requirements for a 3V deposition hole are taken to apply, as far as possible, to a section of the 3H drift that contains one supercontainer and half a distance block on both sides of the supercontainer, and this drift portion is termed a supercontainer section. It is noteworthy that, while in KBS-3V the deposition hole is a separate opening, in KBS-3H the supercontainer section is part of the deposition drift. The requirements on the whole drift, thereby, also apply to the supercontainer sections, which then also have a set of their own requirements. Despite this overlapping, it is sensible to preserve this structure of presenting the requirements to allow easier comparison with 3V.

The performance targets of the buffer need to be modified to some extent to take into account the two types of buffer components present in 3H, i.e. the buffer in the supercontainers and the buffer in the distance blocks. For the most part, similar performance is required from 3V and 3H buffer, but the distance blocks have some requirements of their own, e.g. the hydraulic separation of supercontainers.

As the filling components and compartment and drift plugs are completely KBS-3H specific, their performance targets cannot be derived from the 3V requirements, although the filling components have a few functional similarities with both buffer and 3V deposition tunnel backfill (a component that does not exist in 3H at all), and the drift plug has a function roughly similar to the 3V deposition tunnel plug, although their quite different design plans imply some differences in requirements. Defining requirements for these 3H-specific components requires substantial iteration with long-term safety based considerations and development of design (this iteration is discussed in more detail in Section 5 below). Factors to consider include, for example, the erosion resistance of the filling components, as the consequences of significant bentonite erosion can be more pervasive in 3H drifts than in separate 3V deposition holes. Chemical erosion of bentonite is one of the key long-term issues that are being studied in the current KBS-3H safety case.

TABLE 1. SAFETY FUNCTIONS ASSIGNED TO THE BARRIERS IN POSIVA'S KBS-3V REPOSITORY [7] AND PRELIMINARY SAFETY FUNCTIONS PROPOSED FOR THE BARRIERS IN A KBS-3H REPOSITORY

Barriers	Safety functions for KBS-3V	Preliminary safety functions for KBS-3H
Canister	• Ensure a prolonged period of containment of the spent nuclear fuel. This safety function rests first and foremost on the mechanical strength of the canister's cast iron insert and the corrosion resistance of the copper surrounding it.	
Buffer	• Contribute to mechanical, geochemical and hydrogeological conditions that are predictable and favourable to the canister; • Protect canisters from external processes that could compromise the safety function of complete containment of the spent nuclear fuel and associated radionuclides; • Limit and retard radionuclide releases in the event of canister failure; • In addition, the buffer in the distance blocks shall; • Hydraulically and thermally separate the supercontainers from each other.	
Deposition tunnel backfill	*(see Ref. [7] for safety functions)*	*(barrier not present in 3H)*
Closure	• Prevent the underground openings from compromising the long-term isolation of the repository from the surface environment and normal habitats for humans, plants and animals; • Contribute to favourable and predictable geochemical and hydrogeological conditions for the other engineered barriers by preventing the formation of significant water conductive flow paths through the openings; • Limit and retard inflow to and release of harmful substances from the repository.	
Host rock	• Isolate the spent nuclear fuel repository from the surface environment and normal habitats for humans, plants and animals and limit the possibility of human intrusion, and isolate the repository from changing conditions at the ground surface; • Provide favourable and predictable mechanical, geochemical and hydrogeological conditions for the engineered barriers; • Limit the transport and retard the migration of harmful substances that could be released from the repository.	
Filling components	*(barrier not present in 3V)*	• Contribute to favourable and predictable mechanical, geochemical and hydrogeological conditions for the buffer and canisters; • Limit and retard radionuclide releases in the possible event of canister failure; • In addition, the filling blocks (at inflow locations) shall; • Isolate possible transmissive sections of the drift from the canisters and buffer.
Compartment plugs and drift plugs	*(barrier not present in 3V)*	• Contribute to favourable and predictable mechanical, geochemical and hydrogeological conditions for the filling components, buffer and canisters by keeping the drift components in place.

4.3. Design requirements and design specifications

The design requirements form the VAHA level 4, and the possibilities to use common requirements in both KBS-3V and 3H follow the same outline as presented above for performance targets. Once again, the requirements for the filling components and compartment and drift plugs are unique, although all applicable 3V requirements (mainly for the deposition tunnel backfill and plug) are also included in some form or another, to be able to argue that the KBS-3H concept fulfils the same long-term safety requirements as KBS-3V. The design requirements are still somewhat general in nature, and many of them can be fulfilled with slightly different design solutions. For example, there is an identical design requirement for both 3V and 3H that *the buffer shall be designed to be self-sealing after initial installation and self-healing after any hydraulic and mechanical disturbances*. Even though in 3H the buffer that is initially inside the supercontainer must swell through the perforations of the shell to seal the drift section in question, it is considered possible to achieve this soon after emplacement, with the help of artificial watering mentioned above.

VAHA level 5 (design specifications) is the most detailed and generally the most quantitative of all VAHA requirements, giving the actual specifications for manufacturing and implementing the engineered barriers, and for constructing the underground openings, so that both the EBS and the host rock fulfil the requirements presented on higher levels. For the canister, the specifications are identical in 3V and 3H. For the buffer, there are obviously some changes due to the different geometries and partly different initial properties of the 3H buffer, for example regarding the 3H-specific buffer component, i.e., the distance block. The filling components and compartment and drift plugs require completely new specifications, although the filling component requirements are considered to be fulfilled by the same material, bentonite, as is used in the buffer, and therefore some specifications can be identical with those of the buffer. The design specifications for these 3H-specific components are currently being defined by the design experts on the basis of preliminary higher-level requirements and they are not further discussed here.

5. ITERATION BETWEEN REQUIREMENTS FORMULATION, SAFETY ASSESSMENT AND DESIGN

The development of a hierarchical and comprehensive system of long-term safety requirements for a disposal alternative that has been innovated and refined for more than a decade unavoidably entails significant iteration between requirements formulation, safety assessment and design development. A similar iterative process was also identified in the development of the TURVA-2012 safety case, which fed into the Preliminary Safety Analysis Report compiled for the construction licence application (Fig. 2). The roles of the design basis and performance assessment are described in Section 3.

The continuous iteration between long-term safety requirements formulation, design and implementation is necessary yet challenging as design development often occurs at the same time as requirements development. Ideally, the requirements should come first and the design develops later but, in practice, it is done in parallel and even design requirements may be set prior to long-term safety requirements. This was noticed, for example, already at the time the VAHA requirements management system was established. One key task at the beginning of the VAHA work was to collect requirements of various types from various sources and to develop the five-level structure for requirements that were at least partly formulated long before the VAHA project started. Some design requirements and specifications were developed from earlier iteration loops of long-term safety, design and production that were only partially reported. For example, the thickness of the buffer rings around the canister was

defined around 1983 based on the canister diameter assumed at that time and on the diameter of a deposition hole that was considered feasible to construct using the boring methods available in 1978, i.e. when the 1.5 m deposition hole diameter was first proposed [9, 10]. Subsequent safety analyses have shown that this thickness is adequate and it has remained the same for over 30 years. Clearly, that would not be the case if the thickness had been found unsuitable for long-term safety. It is also important to acknowledge that the KBS-3 method, with its barriers and their specifications, aims specifically at providing long-term protection from the hazards of the spent nuclear fuel. All design work, therefore, has its basis on long-term safety requirements, even if these requirements have been considered implicitly in some stages, only to be explicitly tested in safety assessments. Furthermore, the regulatory requirements are themselves evolving along the development of the repository programme and this introduces additional hurdles to requirements management and design development work. If the design is developed before the final long-term safety requirements are available, this introduces the risk of developing a design that no longer fulfils these requirements. A close cooperation between long-term safety, design and implementation is to be encouraged to avoid such conflicts. Furthermore, a close cooperation among barrier-specific experts is also to be sought since setting requirements on a given barrier has implications on other barriers. In Posiva's case, the VAHA system can be conceived as a tool to manage the requirements and provide traceability and as a communication tool between long-term safety and design and development work.

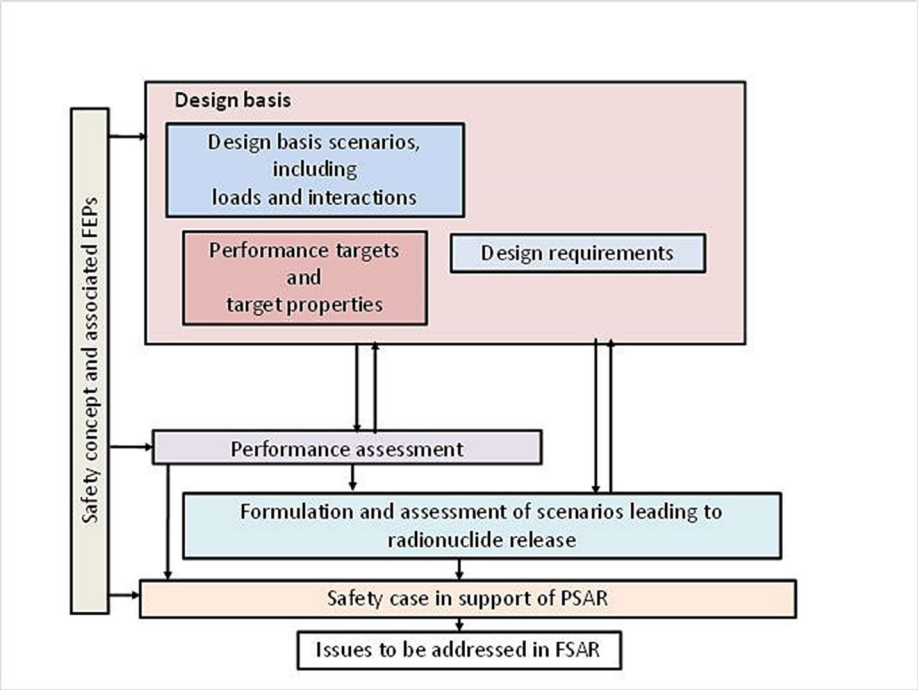

FIG. 2. The development of the repository system as an iteration between requirements, designs and safety assessments (source: Figure 1−7 in Ref. [7]). PSAR = Preliminary Safety Analysis Report for construction licence application and FSAR = Final Safety Analysis Report for operating licence application (the main safety documents required by the Finnish authorities); FEPs = features, events and processes.

As regards the KBS-3H concept, there are additional aspects in this iteration work which need to be taken into account. Even though the current requirement formulation gains from previous 3H safety and design iterations (e.g. the 2007 safety assessment mentioned in Section 1), the time span of the current iteration is very short, as the VAHA system for this

design variant is intended to be developed as part of the safety case project rather than preceding it. The KBS-3H design and production reports are being produced at the same time as the requirement formulation work is ongoing, calling for increased communication, and favouring rapid decisions whenever an improvement to the design is found that is more compatible with the long-term safety requirements being developed. An example of a design change that has been brought about by long-term safety considerations is the change of supercontainer shell and plug material from steel to titanium [6]. Furthermore, several other improvements in the 3H design have taken place since the 2007 safety assessment in response to the previous iterative loop. For example, the DAWE design (see Section 2), which used to be an alternative option in the design, has been selected as the reference design. Resulting from this development work, the KBS-3H design is at a quite mature state as a technical solution. In addition to having been developed from earlier horizontal designs (see, e.g., Ref. [11]), it has, at the same time, benefitted greatly from the vast amount of site investigation data, R&D and safety assessments produced in the KBS-3V programme. This has saved several rounds of iterations, allowing the KBS-3H design to mature rapidly.

6. CONCLUSIONS AND WAY FORWARD

As discussed above, the formulation of requirements for a KBS-3H repository differs to some extent from the situation encountered when initially setting up the VAHA system for KBS-3V. Several rounds of iteration between long-term safety and design development have already taken place in the KBS-3H concept, even before the first attempt to develop a comprehensive requirement system. This introduces some risks to the project as the design may no longer fulfil the long-term safety requirements if the design is developed based on only preliminary requirements. A close cooperation between long-term safety, design and implementation is to be encouraged to avoid such risk.

A further challenge to requirements formulation is that KBS-3V requirements development indirectly affects the KBS-3H requirements because of similarities in design and usage of the same host rock. Yet, there are several components unique to the KBS-3H design that need to be considered in requirements formulation. Furthermore, these unique KBS-3H components do not have the same function: some are not intended to be left in the drifts after closure (e.g. most of the DAWE pipes), but should, nevertheless, be composed of materials that are compatible with the barriers. Some others are components that remain in the repository and have, therefore, higher compatibility requirements with the safety-relevant barriers, even though their purpose is operational (e.g. the supercontainer shells). And finally, there are components that do have safety functions but are different from any 3V components, either by material choice (titanium plugs) or by function (filling components). As it has been found necessary to give some of these components a barrier role, it is also imperative to formulate requirements for them to carefully address all VAHA levels where barriers are treated individually.

> "*The requirements management system VAHA is continuously maintained and updated. The purpose of the database is to ensure that all requirements have been sufficiently taken into account in the development of the disposal system and the establishment of the design basis of the nuclear waste facility*" [7].

Posiva's requirements are also being revised

> "*in response to STUK's comments received in conjunction with the construction licence application decision submitted to the Ministry of Employment and the Economy. During the system development work, Posiva*

collaborates closely with SKB, which is currently developing a similar system for its own purposes" [7].

A joint Posiva-SKB project has been recently established to harmonize the requirements at level 3 (performance targets) and level 4 (design requirements) to facilitate further cooperation in the design and production of our respective geologic repositories.

After the long-term safety requirements have been defined for a specific safety case, the next important task is a performance assessment, which is used to analyze the fulfilment of the performance targets (see Section 3). The performance assessment is carried out based on the requirements and the designs that are available at the time of this assessment. As there are likely to be developments in requirements both for KBS-3H and KBS-3V while the performance assessment is carried out, it is important to maintain a close connection between the work ongoing in both design alternatives of the KBS-3 method. Any update to the long-term safety requirements relevant to KBS-3H will be discussed in the performance assessment report with the aim to identify the main implications on the conclusions. The KBS-3H safety case can thus serve as an extra iteration loop to test the latest updates to long-term safety requirements that are common with KBS-3V.

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TRANSPORT ASPECTS OF SPENT FUEL MANAGEMENT

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Abstract

There are different concepts for management of spent nuclear fuel, including wet storage in pools and dry storage in welded canisters or in dual-purpose transport and storage casks. But all of them require transport of the spent fuel, at least transport to the final destination after the interim storage period. This presentation analyzes how this safe transport of spent fuel after several decades of storage is reflected in the transport regulations and how it can be achieved for different storage concepts. For any transport package it must be ensured that the conditions of the packaging and contents at the beginning of shipment meet the conditions the package design safety analysis is based on. For dry storage of spent fuel this requires that the ageing and the ageing management have to be considered as early as in the design phase for the storage and transport arrangements and have to be considered for the package design approval and the approval of the storage site. The need of consideration of ageing for packages to be stored for a long time before transport can be derived from the current IAEA transport regulations already, but these regulations or the corresponding guidance could be improved to strengthen this requirement, as is discussed in the presentation. On the other hand Germany has developed and applied a concept for transport related ageing assessment and management during dry storage of spent fuel in transport and storage casks. This concepts consists of approval of the package design for transport as a prerequisite for transport and for the interim storage license, followed by review and renewal of the certificate during interim storage period and use of certificate at the end of the interim storage period, including pre-defined actions to be taken before the shipment of the DPC from the facility to the final destination. Experiences from this regulatory concept are presented in the second part of the presentation, as well as consequences for other storage concepts.

1. INTRODUCTION

There are different concepts for management of spent nuclear fuel, including wet storage in pools and dry storage in welded canisters or in transport and storage dual-purpose casks (DPCs). However, all of them require transport of the spent fuel, at least transport to the final destination after the interim storage period. Transport of spent fuel after several decades of storage is challenging due to the ageing of the spent fuel and of the packaging used for storage as well as possible changes in the regulations and the technical knowledge related to the package design safety demonstration. Considerations about an integrated transport and storage safety case for a DPC including the assessment for transport after storage has been the topic of an IAEA working group [1].

Any transport package during shipment must meet the requirements of the transport regulations that are valid at the time of shipment. This means that:

- The demonstration of compliance of the package with the regulations shall be in line with the transport regulations and the state of the technical knowledge at the time of the shipment;
- During shipment the conditions of the packaging and contents shall meet the conditions the package design safety analysis is based on.

For the shipment of spent fuel after an interim storage period these requirements are challenging due to ageing processes and possible changes in the regulations and technical knowledge. Regarding ageing management the dry storage in DPCs is less flexible than wet storage in pools or even dry storage in welded canisters (that allow modifications to the outer packaging at any time), but DPCs have other advantages regarding their safety functions.

Germany has developed and applied a concept for transport related ageing assessment and management during dry storage of spent fuel in DPCs.

"*This concept consists of approval of the package design for transport as a prerequisite for transport and for the interim storage license, followed by review and renewal of the package design approval certificate during the interim storage period and use of this certificate at the end of the interim storage period, including pre-defined actions to be taken before the shipment of the DPC from the facility to the final destination*" [2].

First experience supports this concept.

The need to consider ageing for packages to be stored for a long time before transport can be derived from the current IAEA transport regulations already, but these regulations or the corresponding guidance could be improved to strengthen this requirement, as discussed below.

2. GENERAL PRINCIPLES OF AGEING MANAGEMENT FOR TRANSPORT PACKAGES

The ageing of the package must be taken into account for any transport package during package design and demonstration of compliance with the regulations, as well as during operation and periodic inspection. Ageing considerations may in particular influence the specification of the contents, the design of the packaging and the instructions for handling and inspection of the packaging. Of course, ageing is discussed mostly with respect to packages intended to be transported after a long storage time, e.g. DPCs. But explicit or implicit ageing considerations are important for other packages as well if designed for a long life time period.

For a package used for frequent loading, transport and unloading ageing is usually accounted for as follows:

- Compliance of the package design with the regulations valid while designing the package is shown for the packaging as built and the contents at the time of loading;
- Ageing of the packaging is taken into account mostly as part of the program for quality assurance during use of the package:
  - The packaging is periodically inspected when empty. The inspection interval (in terms of time or number of shipments) and extent should be defined based on consideration of ageing of the packaging components and their safety functions;
  - Parts of the packaging subject to ageing may be inspected, replaced for precaution, or repaired or replaced when degradation is detected during inspection.

These measures aim at keeping the packaging in the condition that had been the basis for the demonstration of compliance with the regulations. Such a program needs to be specified in the design phase of any package:

- If the properties of the contents or packaging may change significantly during the transport period (e.g. by build-up of daughter nuclides or radiolysis) this may be accounted for in the design (limitation of contents or other countermeasures) and in the handling instructions (limitation of transport duration, special procedures for opening of the package after long shipments);
- Changes in the transport regulations or in the technical knowledge related to the safety demonstration are reviewed during the periodic renewal of the package design approval. In case of significant changes, the package design may be adjusted or replaced by a new design.

The same general principles apply to DPCs and other packages intended to be transported after a long storage time, but inspection of the empty packaging is not possible during storage and the possibilities for later changes to the design are limited. Therefore, the ageing management for such packages should be as follows:

- The ageing of the packaging and the contents during the planned storage time should be taken into account from the very beginning in the design and in the demonstration of compliance of the package design with the regulations valid while designing the package. The design should be such that compliance with the regulations should not depend on inspection of the interior of the package during storage. On the other hand, well known storage conditions including monitoring of the packages, if applicable, may be taken into account for the safety demonstration;
- Ageing of the packaging during storage should additionally be taken into account as part of the instructions for handling and periodic inspection that need to be specified in the design phase:
  - The storage conditions for the package should be fixed to the conditions taken as the basis for the compliance demonstration;
  - During storage the effect of changes to the transport regulations and to the technical knowledge related to the safety demonstration of the package should be monitored and assessed ('gap analysis');
  - Before shipment after storage an inspection of the package (including review of the storage records) and maintenance, if required, should be carried out to ensure compliance with the regulations during shipment.

An example of this method is the German concept of enabling transport of DPC after storage.

3. THE GERMAN CONCEPT FOR ENABLING TRANSPORT OF DPC AFTER STORAGE

3.1. General

> "*The concept of interim storage in Germany is based on DPCs which are placed in storage facilities. For such a cask a transport package design approval according to applicable national transport regulations consistent with the IAEA Regulations for the Safe Transport of Radioactive Material is necessary to transport the cask to the interim storage facility, to be able to ship the cask if necessary during the interim storage period and finally to transport it to the final destination when the interim storage period is finished. Such a transport package design approval is also a key element for the safety demonstration of the interim storage facility. This German regulatory concept for transport package design approval and ageing management for DPCs has been described in*" [2].

It distinguishes the following three phases:

3.2. Design phase

In Germany, DPCs can only be stored in an interim storage facility if they have a valid Type B(U)F package design approval certificate. The package design is subject to the normal approval procedure for transport taking into account additionally some specific aspects resulting from the use of the package as a storage cask, as follows:

- The package design has to be defined as part of the DPC design which includes all technical specifications which are necessary to meet the safety criteria for transport and interim storage;
  - "*The long term behaviour of safety related material properties and safety functions of the package design has to be considered as far as possible in the assessment of the package design. Credit can be taken from the well-known environmental conditions the cask will be subject to within the interim storage facility and also from the well-known conditions of the dry atmosphere inside the cask*" [2];
- As part of the quality assurance documents, a plan for inspection of the loaded package before transport after storage has to be developed.

3.3. Storage phase

The package design approval certificate for a DPC in interim storage has to be renewed every 5 or 10 years. In the case of 10 years validity the certificate holder has:

- To evaluate the effect of any new transport regulations on the safety analyses report the certificate is based on; and
- 5 years after issuing the package design approval certificate to confirm that all safety related technical provisions, specifications and safety demonstrations the safety analysis report is based on are still valid. This includes assessment of changes to the applicable regulations, technical standards, and methods of safety demonstration, operational experience and ageing effects.

These evaluation reports have to be sent to the competent authorities and get approved by them. This provides a well monitored process to maintain the validity of the package design approval certificate for transport over the whole storage time. If a deficiency is identified during this process, then measures will be derived and included in:

> "*the package design approval certificate. Their implementation at the concerned casks will then take place at the end of the storage period before the transport from the storage facility will start*" [2].

3.4. Pre-shipment inspection at the end of the interim storage period

Before shipment after storage all actions and measures specified in the plan for inspection of the loaded cask before transport must be performed. This includes actions derived from the review process during interim storage as described above.

The plan for inspection of the loaded cask before transport may contains measures like:

- Inspection of the records of the leak-tightness supervision system during storage;
- Check of all cask related documents including the package design approval certificate for completeness and consistency;
- Visual inspection of the cask, inspection of the containment system and leak-tightness test, inspection of the trunnions including tests.

If additional measures need to be performed as derived from the review process described above then such measures will be included in the plan for inspection before transport or separately specified within the Type B(U)F package design approval certificate.

All measures to be performed before transport are supervised by the competent authority:

> "*Only if all measures and actions are successfully completed the transport of the cask from the interim storage facility can take place*" [2].

4. WAY TO IMPROVE THE TRANSPORT REGULATIONS (SSR-6)

Currently there is no explicit wording about consideration of ageing in the transport regulations SSR-6. Nevertheless an ageing management (the extent of which should depend on the package design) in the form of maintenance is already required to be part of the management system that needs to be implemented according to para. 306 of SSR-6.

The international discussion has shown, that the IAEA transport regulations should be improved by including directly into SSR-6 the requirements for additional actions necessary for packages intended to be transported after storage. Taking into account the considerations about ageing management described above and the German experience from the established procedure the following changes to SSR-6 are proposed:

- It should be required that the design of any package shall take into account ageing processes. The case of a package intended to be used for shipment after storage should be directly mentioned to clarify that ageing considerations are especially important for these package designs;
- It should be explicitly stated that for a package intended to be used for shipment after storage the application for package design approval should consider ageing of the packaging and the contents in terms of safety demonstration. The management system for such a package should account for ageing, including appropriate instructions for operation and maintenance (with the inspection before shipment after storage as a key element) as well as a periodic gap analysis program as discussed above.

This means that the areas of package design, demonstration of compliance with the regulations and package design approval should be improved. The regulatory text for implementation of these principles could read like this:

614bis. The design of the package shall take into account ageing mechanisms considering also the time period between preparing and consigning the package, so that even after storage the package meets all applicable requirements at the time of transport.

809 (j): For the design of a package intended to be used for shipment after storage, long term safety considerations to provide evidence that the design is adequate to meet all applicable requirements after storage at the time of transport. These include considerations of ageing mechanisms in safety analysis and in the management system, in particular in the specification of operational and maintenance instructions as well as a gap analysis program to consider potential changes in regulations, technical knowledge and state of the package design during storage.

838 (t) A specification of the applicable management system, as required in para. 306, which shall also take into account appropriate requirements according to para. 809 (j).

5. CONCLUSIONS

For packages intended to be shipped after several years of storage it is important to consider ageing processes in the design as well as in the instructions for handling and periodic

inspection of the packaging as part of the management system. For storage and transport of spent nuclear fuel in Germany a method for obtaining and maintaining the transport package design approval considering ageing has been developed for DPCs. Up to now, application of this concept has shown good results.

For clarifying and emphasising that ageing needs to be taken into account for all packages intended to be used for shipment after storage the IAEA regulations for the safe transport of radioactive material SSR-6 should be modified. Improvements are currently under discussion and could be introduced during the current review cycle of the IAEA transport regulations.

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SPENT FUEL TRANSPORT CONTAINER C-30, PART II: USING THE OLD TYPE TRANSPORT CASK C-30 FOR AN IMPROVED FUEL VVER-440

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Abstract

The transport cask C-30 with basket T-12 was developed in former East Germany for VVER-440 fuel with enrichment 3.6%. Later was in Slovakia developed the new compact basket KZ-48 for VVER-440 fuel with enrichment 4.4% and maximal burnup 55 MWd/kgU. The present licence is for VVER-440 fuel with enrichment 4.4% and maximal burnup 60 MWd/kgU. The future licence need to be for VVER-440 fuel with average enrichment 4.87% and maximal burnup 70 MWd/kgU. In this paper are criticality, inventory and shielding analyses described.

1. TRANSPORT CASK C-30

The transport cask C-30 was developed in former East Germany in 80-ties last century for transport spent fuel VVER-440 type (hexagonal assemblies). The transport cask consists from metal body with lids and inner basket. Two baskets were possible: T-12 for non-defected fuel and T-13 for defected fuel in hermetic casings. The transport cask C-30 is wet type.

The basket T-12 is very simple — assemblies are only in water with pitch 22.5 cm (Fig.1). The capacity is 30 assemblies. The basket T-12 was a dual type — spent fuel assemblies were stored directly in Interim Spent Fuel Storage Facility in Bohunice NPP in T-12 baskets. At present the basket T-12 is not used.

In 90-ties last century was reconstruction of Bohunice ISFSF (Interim Spent Fuel Storage Facility) done. The seismic resistance has increased, and the capacity has increased too. To increase of capacity the new basket KZ-48 was developed in Slovakia. The basket KZ-48 is again dual — for use in transport cask and in ISFSF too. The basket KZ-48 (Fig. 2) is compact basket with boron steel rack with pitch 16.8 cm. The capacity is 48 assemblies.

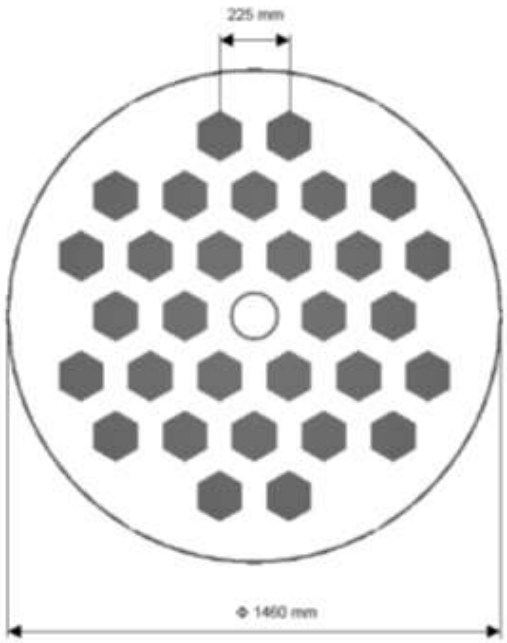

FIG. 1. The basket T-12.

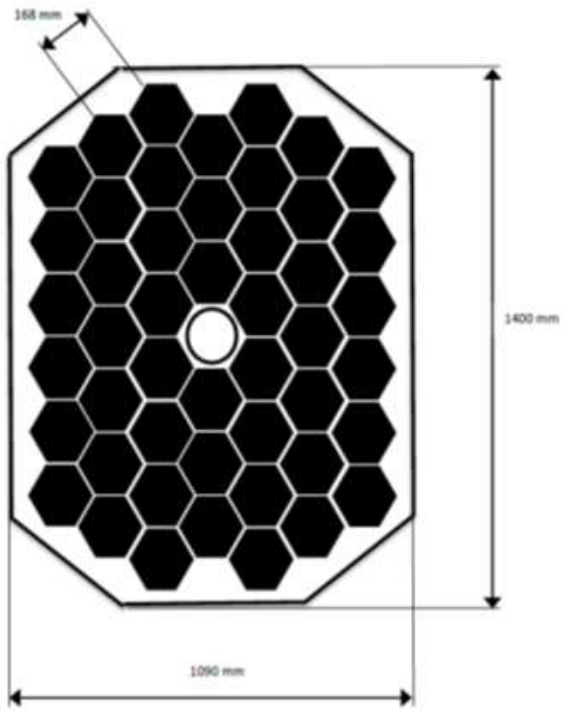

FIG. 2. The basket KZ-48.

2. FUEL

The reactors VVER-440 are more than 40 years in operation. During so long time the fuel was developed and improved. In this article are analyses only for 3 types:

- 3.6% = original fuel of 1st generation, original basket T-12. All fuel pins have the same enrichment 3.6% of ^{235}U;

- 4.25% = fuel 2nd generation with burnable absorber, fuel pins have radial profiled enrichment (maximal enrichment is 4.4% ^{235}U) and 6 pins contain burnable absorber Gd_2O_3. This fuel is in present transported in transport cask C-30 in the basket KZ-48. The present license for transport cask C-30 with basket KZ-48 is for fuel with maximal enrichment 4.4% and maximal burnup 60 MWd/kgU;
- 4.87% = present fuel 2nd generation with burnable absorber, fuel pins have radial profiled enrichment (maximal enrichment is 4.95% ^{235}U) and 6 pins contain burnable absorber Gd_2O_3. This fuel is in present in core and in 5 - 6 years will be transported in cask C-30 in the basket KZ-48. Estimated burnup should be 70 MWd/kgU. The safety report for this fuel will be prepared in 4 years.

3. USED CODES AND LIBRARIES

The SCALE 6.1.2 system [1] was used for criticality and inventory analyses. The criticality calculations were with the KENO VI module (Monte Carlo method) and library v7-238 (continuous energy) carried out. The inventory calculations were with the ORIGEN module and libraries vver440 (3.6), vver440 (4.25) and vver440 (4.87) carried out.

The TORT-DORT code [2] with library BUGLE [3] was used for shielding analyses.

4. CRITICALITY ANALYSES

The scope of analyses is defined in Regulation No. 57/2006 Coll. on details concerning the requirements for shipment of radioactive material [4]. In this presentation are only two cases: routine condition and accident condition with mechanical destruction. The limit for routine condition is $k_{ef} < 0.95$ and the limit for accident condition is $k_{ef} < 0.98$ [5].

The results are in Table 1. k_{ef} is calculated for conservative parameters without burnable absorber and includes $2\sigma_{MC}$ and $2\sigma_{exper.}$ ($k_{eff} = k^{cal}_{eff,conser.param.} + 2\sigma_{MC} + 2\sigma_{exper.}$)

TABLE 1. MULTIPLICATION COEFFICIENT K_{EF}, 1 CASK C-30

Enrichment and basket	Condition	
	Routine $k_{ef} < 0.95$	Accident $k_{ef} < 0.98$
3.6% in T-12	0.85727	1.22095
4.4% in KZ-48	0.87908	0.95381
4.87% in KZ-48	0.89909	0.97504

We see that for routine condition all cases meet the requirement for subcriticality. For accident condition the basket T-12 doesn't meet the requirement. In present in Slovakia the basket T-12 is not in use and hasn't licensee. In past, by safety analyses for safety report we have burnup credit used.

5. INVENTORY ANALYSES

In this article are analyses only for 3 types of fuel described in chapter Fuel. For inventory analyses is very important burnup and cooling time. The inventory analyses have been for following cases carried out:

- **3.6%:**
  - Burnup 42 MWd/kg, cooling time 2.8 y = average burnup in basket T-12 (basic value for relative comparison);
  - burnup 46 MWd/kg, cooling time 2.8 y = maximal burnup in basket T-12.
- **4.25%:**
  - Burnup 50 MWd/kg, cooling time 3.6 y = average burnup in basket KZ-48 (first license);
  - Burnup 55 MWd/kg, cooling time 3.6 y = maximal burnup in basket KZ-48 (first license);
  - Burnup 60 MWd/kg, cooling time 3.9 y = maximal burnup in basket KZ-48 (present license for higher burnup).
- **4.87%:**
  - Burnup 60 MWd/kg, cooling time 4 y = maximal burnup in basket KZ-48 in peripheral zone;
  - Burnup 70 MWd/kg, cooling time 5 y = maximal burnup in basket KZ-48 in central zone.

Results are listed in Tables 2–5 and shown (absolute value and relative value to original parameter). We see that by increasing cooling time is possible to receive similar value for activity and photons source. To keep decay heat should be to increase cooling time a little more. The problem is with neutron source – to keep similar level is necessary to increase cooling time several times, because main producer of neutrons is ^{244}Cm with $T_{1/2}$= 18.1 y.

TABLE 2. ACTIVITY

Enrich. [%]	Burnup [MWd/kgU]	Cooling time [y]	Activity		
			1 assembly [Bq]	1 cask [Bq]	1 assembly relative
3.6	42	2.8	4.703E+15	1.411E+17	1.000
	46	2.8	5.112E+15		1.087
4.4	50	3.6	4.593E+15	2.205E+17	0.977
	55	3.6	4.995E+15		1.062
	60	3.9	5.118E+15		1.088
4.87	60	4	4.890E+15		1.040
	70	5	4.971E+15		1.057

TABLE 3. DECAY HEAT

Enrich. [%]	Burnup [MWd/kgU]	Cooling time [y]	Decay heat		
			1 assembly [W]	1cask [kW]	1 assembly relative
3.6	42	2.8	457.2	13.72	1.000
	46	2.8	514.7		1.126
4.4	50	3.6	440.6	21.15	0.964
	55	3.6	502.3		1.099
	60	3.9	532.8		1.165
4.87	60	4	487.2		1.066
	70	5	529.1		1.157
		limit	800	24	

TABLE 4. PHOTON SOURCE

Enrich. [%]	Burnup [MWd/kgU]	Cooling time [y]	Photons		
			1 assembly [s^{-1}]	1cask [s^{-1}]	1 assembly relative
3.6	42	2.8	2.731E+15	8.194E+16	1.000
	46	2.8	3.043E+15		1.114
4.4	50	3.6	2.515E+15	1.207E+17	0.921
	55	3.6	2.814E+15		1.030
	60	3.9	2.881E+15		1.055
4.87	60	4	2.611E+15		0.956
	70	5	2.555E+15		0.936

TABLE 5. NEUTRON SOURCE

Enrich. [%]	Burnup [MWd/kgU]	Cooling time [y]	Neutrons		
			1 assembly [s^{-1}]	1cask [s^{-1}]	1 assembly relative
3.6	42	2.8	8.749E+07	2.625E+09	1.000
	46	2.8	1.264E+08		1.445
4.4	50	3.6	1.350E+08	6.480E+09	1.543
	55	3.6	1.956E+08		2.236
	60	3.9	2.693E+08		3.079
4.87	60	4	2.268E+08		2.592
	70	5	3.852E+08		4.403

6. SHIELDING ANALYSES

The limits are [4]:

- 2 mSv/h for surface;
- 0.1 mSv/h in distance 2 m from surface.

- **Fuel 3.6%:**

The original case (30 assemblies 3.6% in the basket T-12, average burnup in cask 42 MWd/kgU, maximal burnup of assembly 46 MWd/kgU) meet s limits without problems.

- **Fuel 4.25%:**

The first license was for maximal burnup 55 MWd/kgU and average burnup in basket KZ-48 50 MWd/kgU without any restriction for position assembly in basket.

The present license is for fuel with maximal enrichment 4.4% ^{235}U and maximal burnup 60 MWd/kgU. Two cases how are assemblies placed in the basket KZ-48 were evaluated:

- Assemblies are in two areas (Fig. 3):
    - Central area: 26 assemblies, maximal burnup 60 MWd/kgU;
    - Peripheral area: 22 assemblies, maximal burnup 55 MWd/kgU;
- Assemblies are only in 44 positions with maximal burnup 60 MWd/kgU and 4 positions are empty (Fig. 4).

For both cases the dose meets limits.

- **Fuel 4.87%:**

For present fuel with average enrichment 4.87% and maximal burnup 70 MWd/kgU we have preliminary 3 cases evaluated:

- Assemblies are in two areas (Fig. 5):
    - Central area: 26 assemblies, maximal burnup 70 MWd/kgU;
    - Peripheral area: 22 assemblies, maximal burnup 60 MWd/kgU;
- Assemblies are in two areas and 4 positions are empty (Fig. 6);
    - Central area: 26 assemblies, maximal burnup 70 MWd/kgU;
    - Peripheral area: 18 assemblies, maximal burnup 60 MWd/kgU;
    - Peripheral area: 4 empty positions;
- Assemblies are in three areas (Fig. 7);
    - Central area: 26 assemblies, maximal burnup 70 MWd/kgU;
    - Peripheral area: 18 assemblies, maximal burnup 60 MWd/kgU;
    - Peripheral area: 4 assemblies, maximal burnup 55 MWd/kgU.

For all three cases the dose meets limits.

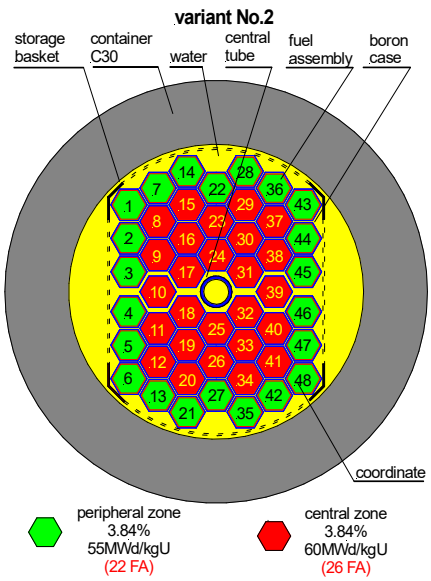

FIG. 3. Two-zone configuration (fuel 3.84%/4.25%).

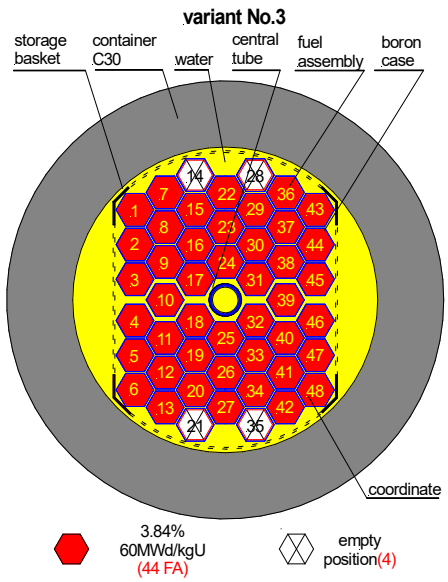

FIG. 4. Single-zone configuration with empty positions (fuel 3.84%/4.25%).

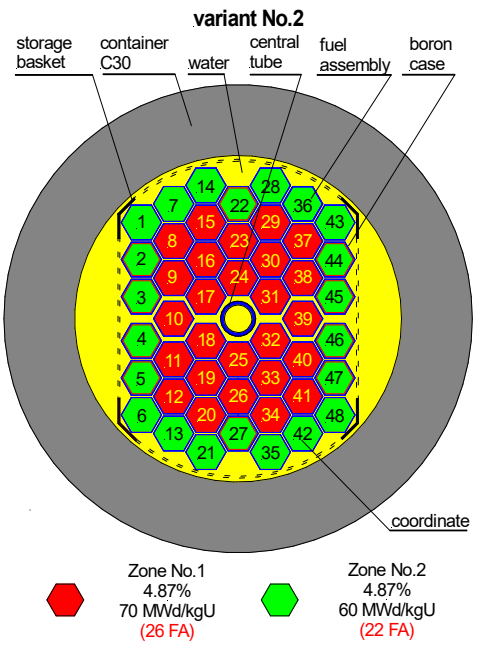

FIG. 5. Two zone configuration (fuel 4.87%).

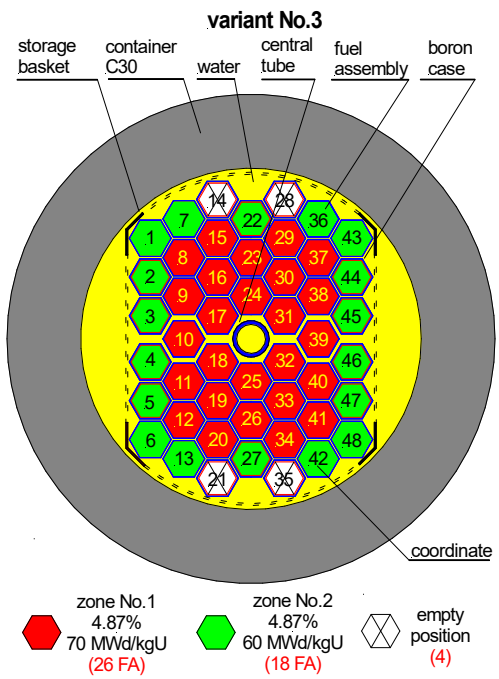

FIG. 6. Two zone configuration with empty positions (fuel 4.87%).

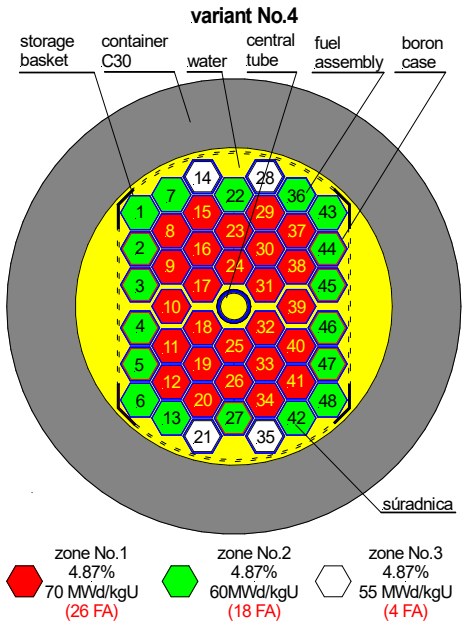

FIG. 7. Three zone configuration (fuel 4.87%).

7. CONCLUSION

By using a new improved fuel in old type of transport cask it is necessary to check safety limits and use some new actions (and/or):

- Criticality: if is necessary is possible to use burnup credit;
- Decay heat;
  - Prolong cooling time;
  - Decrease number of assemblies in cask;
- Dose;
  - Prolong cooling time;
  - Decrease number of assemblies in cask;
  - Areas according burnup;
  - Additional shielding.

It is possible to use the very old design transport cask C-30 for improved fuel with higher enrichment and higher burnup. To keep safety limits is necessary to prolong cooling time and have some restriction. The other way is to develop a totally new transport cask.

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THERMAL FIELD MODELING OF SPENT FUEL TRANSPORT CONTAINER C-30

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Abstract

In Slovakia we use nuclear energy for more than forty years. In the past spent fuel was transported to former Soviet Union. Since 1987 we store all spent fuel in Interim Spent Fuel Facility at Bohunice site. So the spent fuel is stored before its final deposition or reprocessing. For transport of spent fuel we use transport container C-30. The paper describes thermal calculations of C-30. The capability of removal of residual heat from spent fuel is very important feature for every device or facility. It is therefore important to understand the process of heat removal. First calculation of thermal field of transport container C-30 was done by the manufacturer of C-30 in eighties. The calculation was used as an approval for the ability of sufficient heat removal. In nineties and in the beginning of 21st century the use of new type of fuel raised question, whether C-30 could be used also for transportation of spent fuel with higher burnup and residual heat. In the application for the type approval of C-30 new calculation was realized, in order to demonstrate the ability to safely divert the residual heat. The evaluation of information technology enabled to use more detailed model then in previous calculations. Despite technological advance all models used one common assumption. The inside of the container was considered to be a homogenous heat source. This assumption may be sufficient for type approval; however, in order to get most accurate results more detailed analysis of inside of C-30 became desirable. New calculation of residual heat removal considers the inventory of C-30 as a system of separate spent fuel assemblies — separate heat sources. The paper describes the results of thermal field modeling calculations.

1. INTRODUCTION

The use of nuclear energy started in Slovakia in 1972. Spent fuel was transported to former Soviet Union. Since 1987 all spent fuel is being stored in Interim Spent Fuel Facility at Bohunice site before its final deposition or reprocessing.

During transport and storage of spent nuclear fuel we have to ensure three main functions of transport and storage facilities:

- Ensure subcriticality;
- Ensure protection against radiation and protection of environment, and
- Ensure *residual heat removal*.

The paper describes the evaluation of modeling and calculation of thermal field for transport container C-30.

2. TRANSPORT CONTAINER C-30

Transport container C-30, used for transportation of VVER-440 spent fuel, was designed and manufactured in former German Democratic Republic, VEB Germania, Karl-Marx-Stadt, in years 1984 to 1990. Spent fuel is inserted into transport container in casks T-12, T-13 or KZ-48. Fig. 1 shows cross-section of transport container C-30 with spent fuel in cask KZ-48.

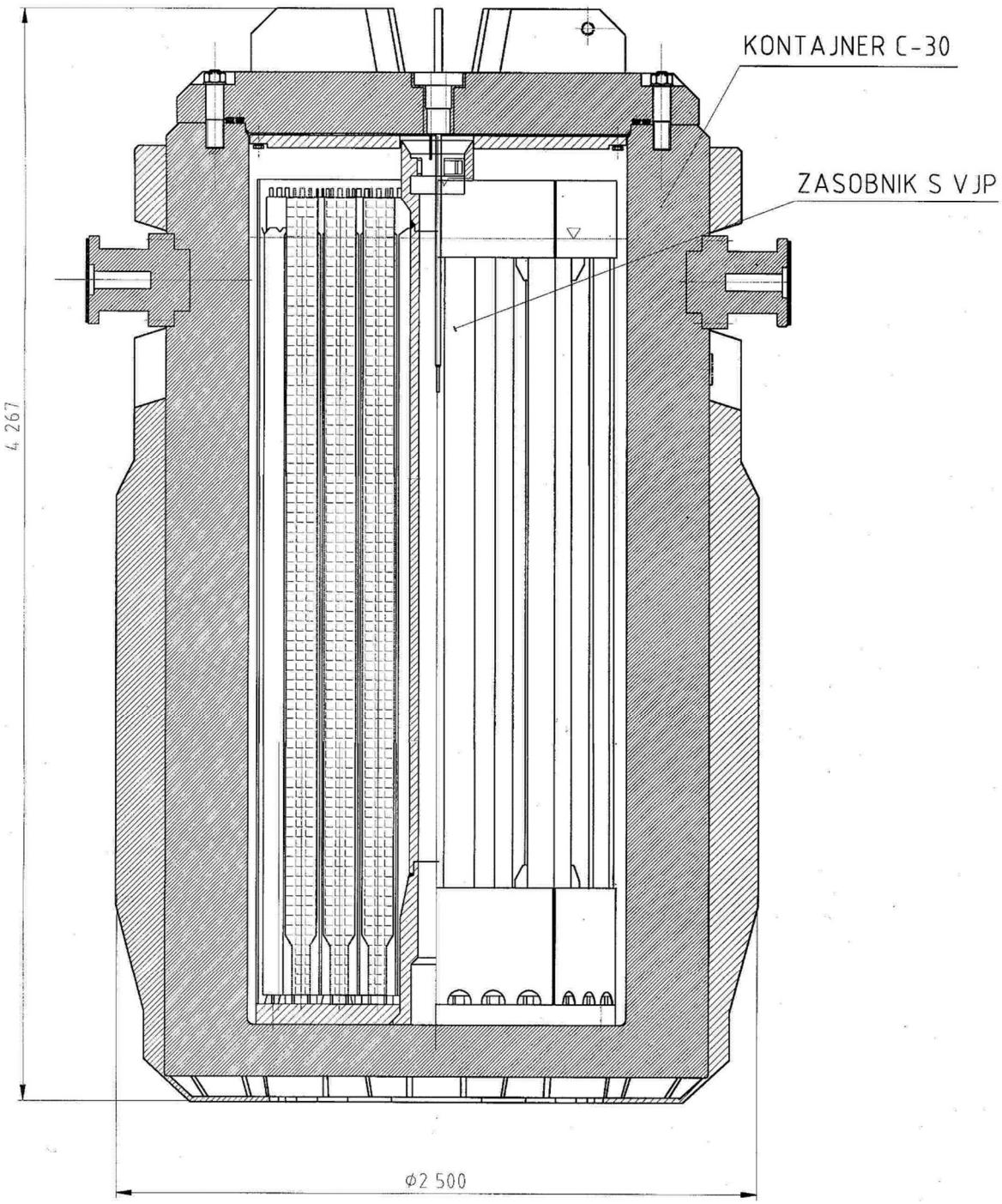

FIG. 1. Cross-section of transport container C-30 with spent fuel in cask KZ-48.

3. THERMAL CALCULATIONS

3.1. Original thermal calculations

Original technical documentation contains thermal calculations of the container in Volume III. For calculation it was considered transport of 30 spent fuel assemblies with total residual

power of 5, 8, 12 and 15 kW for wet transport and 5 and 8 kW for dry transport. All calculations were performed by using special computer code TEMPN 2. Fig. 2 shows the calculation model.

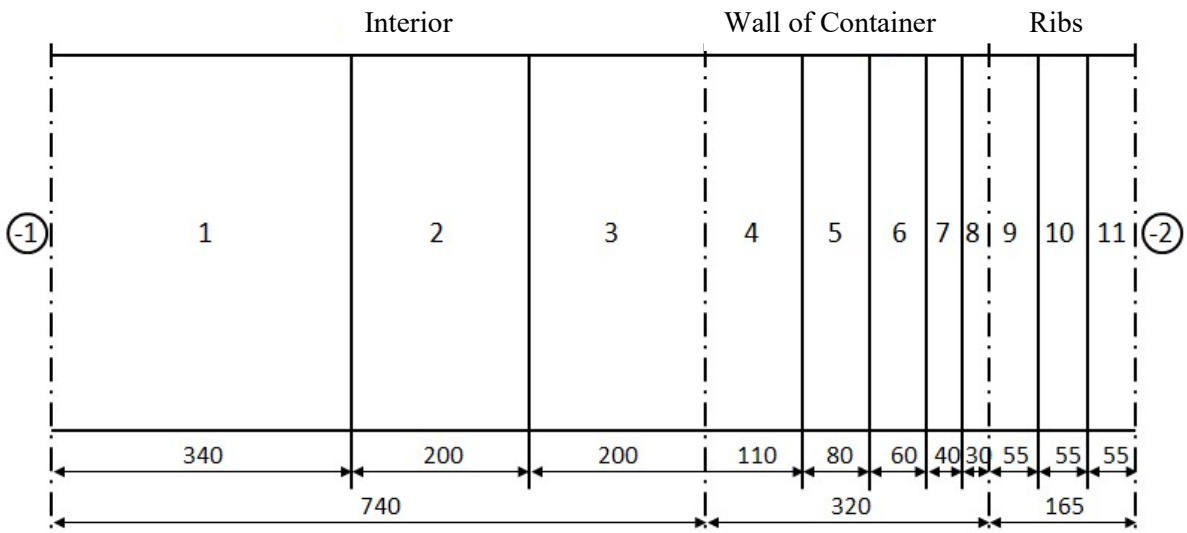

FIG. 2. Calculation model for transport container C-30.

Elements 1 to 3 represent the interior of the container, elements 4 to 8 represent the wall and elements 9 to 11 represent ribs of the container. -1 in the circle represents the axis of symmetry and -2 in the circle represents environment.

3.1.1. Input data

As input data for the calculation dimensions and material properties of the container, characteristic values of radiation and environment parameters, enthalpy and thermal conductivity and radiation coefficient between container and environment were taken into consideration. The calculation for heat output 5, 8, 12 and 15 kW was performed. It was considered for this one-dimensional model, that heat is dissipated thru the wall of container. The ambient temperature was considered between +38°C to -40°C. As required by the shipment type B the influence of sunlight was also considered.

3.1.2. Results

The calculations gave following results:

(1) For ambient temperature +38°C, wet transport, without considering sunlight, the surface temperature of the container was +50.5°C by the heat output of 5 kW and +67.5°C by 15 kW;
(2) For ambient temperature +38°C, wet transport, with considering sunlight, the surface temperature of the container was +74.1°C by the heat output of 5 kW and +87.1°C by 15 kW;
(3) For ambient temperature -40°C, wet transport, the surface temperature of the container was -25.8°C by the heat output of 5 kW and -7.0°C by heat output 15 kW;

(4) For ambient temperature -40°C, wet transport, cannot be excluded, that water will freeze inside the container. In order to avoid water freezing the minimum heat output should be higher than 8.2 kW for closed transport means and 13.5 kW for opened transport means, by maximal transport velocity 40 km per hour;
(5) Permissible temperature of spent fuel assemblies 300°C will not be exceeded.

The calculation showed, that most the dangerous transport conditions are low ambient temperature and low heat output. Such conditions may cause water freezing which could damage the fuel assemblies or even the transport container.

3.2. Thermal calculations in 2002

In 2002, Slovak Electric, joint stock Company, applied for type approval of transport container C-30. The Preliminary Safety report describes in Chapter 6 thermal and pressure stability of transport container.

3.2.1. Transport requirements

Transport requirements were determined according to legislation in force as follows:

(1) Temperature in range from -40°C to +38°C, considering sunlight;
(2) Maximal pressure less than 700 kPa;
(3) Maximal surface temperature less than 85°C;
(4) Transport container should withstand outside temperature minimal +800°C for 30 minutes without damage to it and to spent fuel.

3.2.2. Thermal stability analysis

For purposes of thermal analysis of transport container C-30 a 2D model was evaluated. The model represents horizontal cross-section of container the place with the most adverse foreseeable temperature distribution. Finite-element network is shown on Fig. 3.

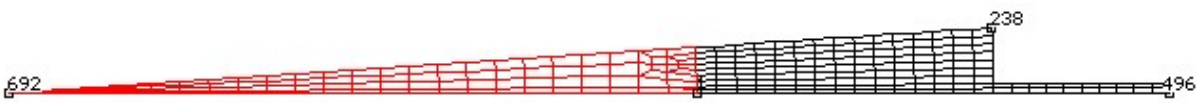

FIG. 3. 2D model with finite-element network.

The model consists of a total of 262 eight-node quadratic elements related to the 889 grid points.

3.2.3. Input data

Dimensions and material properties of the container, characteristic values of radiation and environment parameters, enthalpy and thermal conductivity of container were taken into consideration. Transport conditions were determined according to 3.2.1. The inventory of the container (spent fuel, cask, and water) was considered as a homogenous entity with internal

heat source. Thermal analysis was implemented in stationary mode. It was considered, that heat dissipation from the surface of the container is performed through radiation and convection.

3.2.4. Results

Based on the described 2D model temperature of the container wall and inventory were calculated under normal conditions of transport and test conditions as required in legislative in force. The calculations gave following results:

(1) For ambient temperature +38°C, wet transport, with considering sunlight, KZ-48 cask, the maximal heat output is limited by maximal surface temperature +85°C by 24 kW;
(2) For ambient temperature -40°C, wet transport, maximal transport velocity 100 km per hour and heat output below 15 kW, water will freeze inside the container;
(3) For overheating test with ambient temperature +38°C, the transport container is exposed for 30 minutes to the temperature 800°C, and then it is naturally cooled for 24 hours considering sunlight. Maximal temperature is reached at the end of overheating respectively during the cooling phase;
(4) In terms of meeting the requirements of the thermal stability of the container under normal transportation conditions at ambient temperature 38°C, the maximum permitted residual power of spent fuel is set to 24 kW.

Maximal spent fuel heat output in transport container C-30 is 24 kW. For ambient temperature -40°C, wet transport, maximal transport velocity 100 km per hour, and heat output below 15 kW, water will freeze inside the container, what could lead to damage of container due to brittle fracture and loss of tightness.

3.3. Thermal calculations in 2009

After license expiration in 2009 the Nuclear and Decommissioning Company, joint stock Company, applied for type approval of transport container C-30. They submitted Document for Type Approval of Transport means — Transport Container C-30. Chapter 6 describes the thermal and pressure calculations.

3.3.1. Transport requirements

Transport requirements are the same, as in 3.2.1.

3.3.2. Thermal stability analysis

The analysis was performed by the computer code COSMOS, which allows solving stationary and non-stationary, linear and non-linear thermal calculations using finite element method at the specified geometrical, physical, initial and boundary conditions. The calculations verify the maximum total residual power 24 kW. The resulting course of temperatures was determined using the computer code NISA/HEAT. 3D geometrical model of the container was developed for the calculation purposes. The inventory of the container was considered as a homogenous entity with internal heat source. Thermal analysis of transport container C-30 was implemented as a stationary task.

3.3.3. Input data

Input data are the same as in 3.2.3.

3.3.4. 3D model

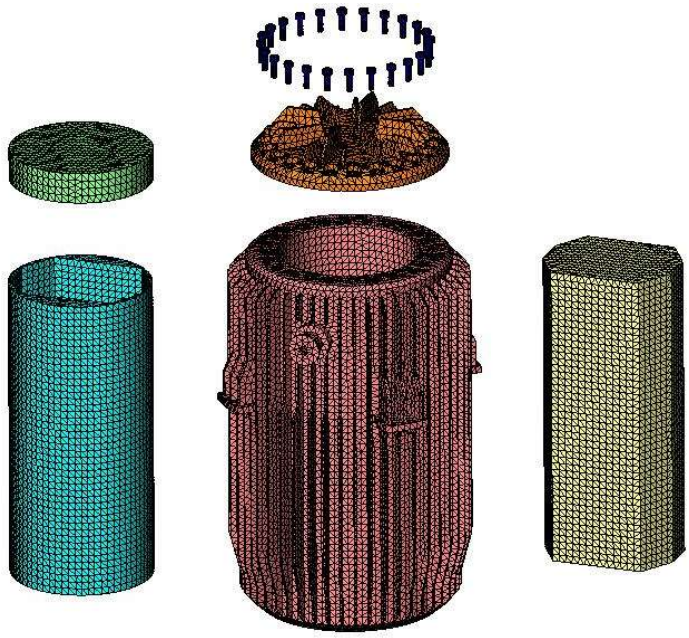

FIG. 4. Geometry of the calculation model with typical finite element mesh.

Fig. 4 shows geometry of the calculation model with typical finite element mesh. Model has around 450 000 nodes and 300 000 elements.

3.3.5. Results

Following simulations were performed:

(1) Container in an environment with a temperature of +38°C. Maximal surface temperature by heat output 24 kW does not exceed 85°C. In this case the maximal temperature of fuel cladding is 118°C. Fig. 5 shows temperature of individual parts of the container;

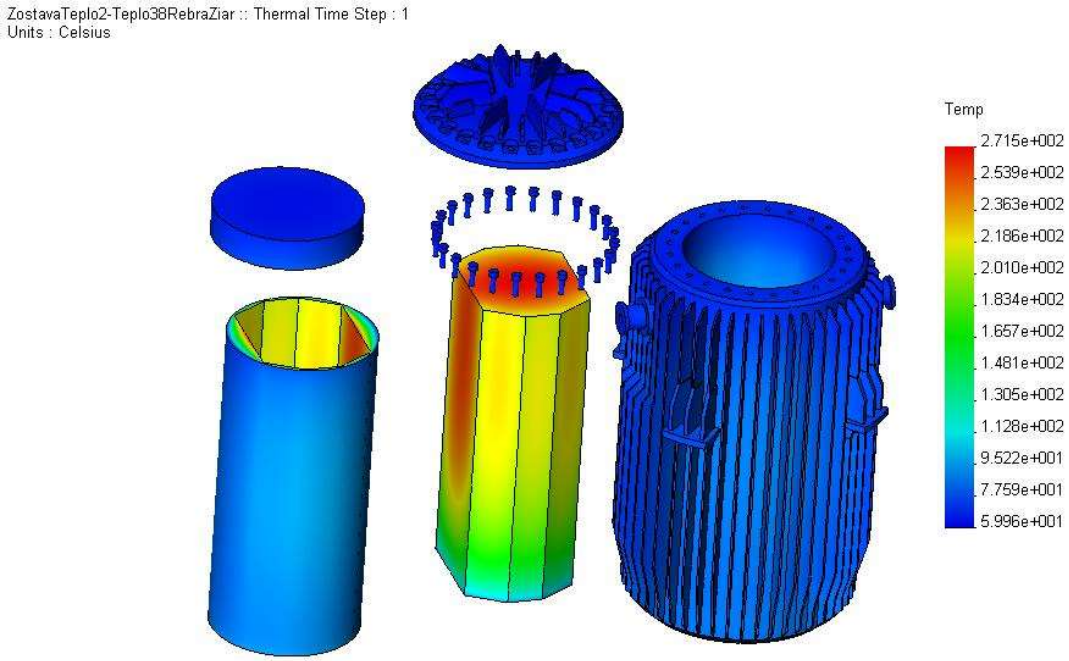

FIG. 5. Temperature of individual parts of the container.

(2) Container in an environment with a temperature of -40°C. When transporting the container on the wagon by ambient temperatures of -40°C, and heat output below 15 kW, the temperature of the coolant falls under 0°C. This results in a local freezing of the cooling water in the container;

(3) Container in an environment with a temperature of +38°C, considering sunlight with heat flux 800 Wm^{-2} for 12 hours;

(4) Container in the flame with a temperature of 800°C for 30 minutes followed by stay in an environment of 38°C and exposed to the sunlight for a period of 6 hours with a heat flux of 800 Wm^{-2}. During cooling phase, the maximal temperature of fuel cladding does not exceed 187°C.

Numerical analysis shows that the maximum inventory temperature does not exceed at any operating, conditions or thermal tests, the value of 283°C. Given the fact that the maximum permissible temperature of fuel pins is 750°C, it can be concluded that during transportation of fuel assemblies with a total residual power 24 kW none thermomechanical damage to fuel rods occurs.

3.4. Thermal calculations in 2014–15

The main reason for starting new thermal field calculations was the fact, that all previous calculations considered the inventory of the container (spent fuel assemblies, cask, and water as a homogenous entity with internal heat source. Another reason was that computing power of computer has increased in such way, that such analysis became possible.

3.4.1. Input data

Input data are the same as in 3.2.3.

3.4.2. 3D model

The main difference between 2009 and 2014−15 3D model consists in way of modeling the inventory of the container. In all previous calculations the inventory was considered as homogenous entity with heat source. The 2014−15 3D model is more complex, and it models in a simply way fuel assemblies as well as cooling medium flow. Fig. 6 shows temperature distribution in transport container horizontal cross-section for total heat output of 20 kW.

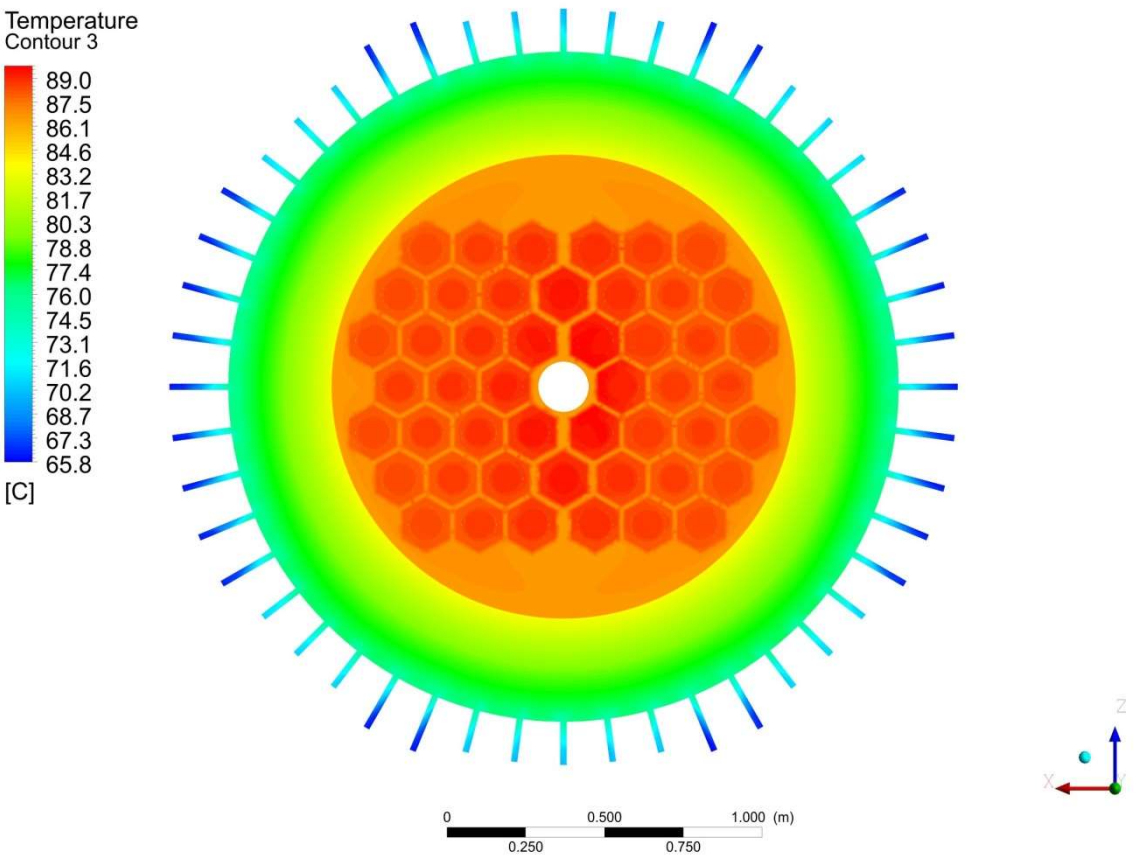

FIG. 6. Temperature distribution in transport container horizontal cross-section, total heat output of 20 kW.

Each spent fuel assembly is divided into two parts. The central circular part represents the area of water between fuel pins. This part does not produce any heat. The rest of the assembly is bordered by hexagon on the outer periphery and by a circle inside of the fuel assembly. Only this part is responsible for heat production. The model takes into consideration also the coolant (water) flow.

3.4.3. Results

As for now only basic calculations have been made. As the model consists of more than 9 000 000 tetrahedrons, each calculation needs quite long time — days even weeks. Full results will be available by the beginning of next year. Fig. 7 shows thermal field model of transport container vertical cross-section with total heat output of 20 kW.

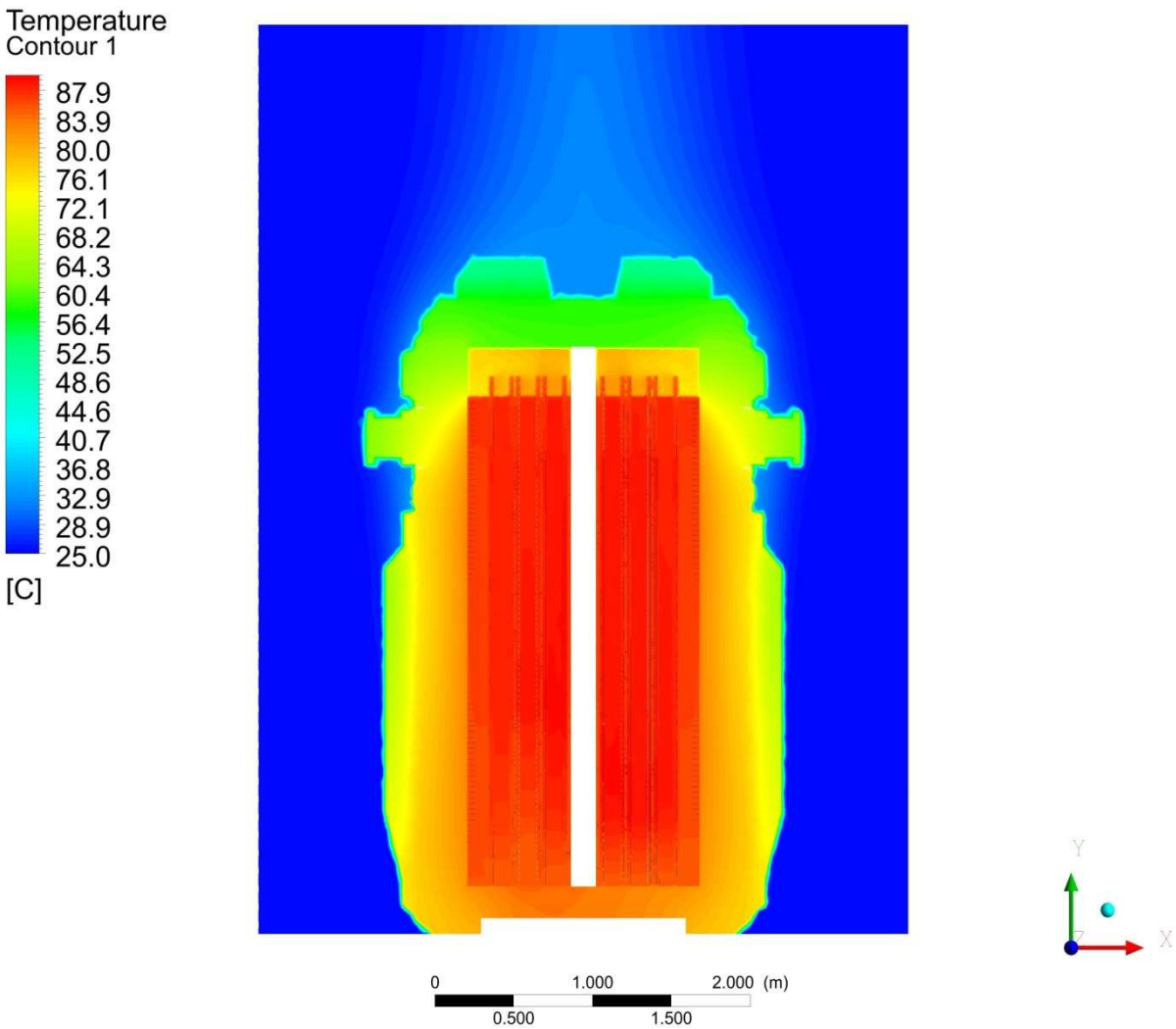

FIG. 7. Thermal field model of transport container vertical cross-section, total heat output of 20 kW.

4. CONCLUSION

The aim of thermal calculations of transport container is to prove, that residual heat produced by spent fuel could be safely lead away without any damage to the fuel and to container. First calculations were made thirty years ago. We can see, that improvement of computing power allows creating more complex geometrical models of container. During last years also, the enrichment and burnup of the nuclear fuel has increased. All these facts cause the need for further, more detailed analysis of the inventory of the transport container C-30.

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ACTIVITIES OF THE OECD/NEA/CSNI WORKING GROUP ON FUEL CYCLE SAFETY IN SPENT FUEL AND HIGH-LEVEL WASTE MANAGEMENT

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Abstract

The Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) assists member countries in ensuring the adequate safety of existing and future nuclear installations, through maintaining and developing the knowledge, competence and infrastructure needed to regulate and support the complete life cycle. NEA's Working Group on Fuel Cycle Safety (WGFCS) brings together representatives of regulatory bodies, their technical support organizations, and operators of nuclear fuel cycle facilities, which comprise of interrelated activities, including: uranium mining and milling; uranium refining and conversion to uranium hexafluoride; uranium enrichment; fuel fabrication and storage; spent fuel storage; spent fuel reprocessing; decommissioning of nuclear facilities; radioactive waste management and disposal options (including for spent fuel); and the research and demonstration facilities that support these activities. This paper discusses current WGFCS activities in the area of spent fuel and high-level waste (HLW) management. Consistent with its mandate, the WGFCS, in cooperation with the NEA Working Group on Fuel Safety, held a workshop on the safety of long term interim storage (LTIS) facilities in Munich, Germany, in May 2013. The workshop covered national approaches, safety requirements, regulatory framework and implementation issues, technical issues, operational experience, and research and development for LTIS. As follow-on to recommendations from the workshop, and recommendations from NEA's Committee on the Safety of Nuclear Installations, the working group is developing a Technical Opinion Paper (TOP) that will gather information presented at the workshop. This information will be complemented by information supplied by WGFCS members to develop a picture of the current status of LTIS in NEA member countries. The TOP will provide an overview of LTIS requirements and technical needs, based on approaches in member countries.

Key Words: OECD, NEA, Long Term Interim Storage.

1. INTRODUCTION

This paper describes activities of the Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), Working Group on Fuel Cycle Safety (WGFCS) in the area of spent fuel and high-level waste management. Specifically, this paper provides background information on the OECD/NEA WGFCS, a summary of the International Workshop on the Safety of Long Term Interim Storage Facilities which was organized by the WGFCS, and a description of the WGFCS's current effort to develop a technical opinion paper (TOP) on this subject.

This presentation is made on behalf of the Organisation for Economic Co-operation and Development Nuclear Energy Agency (NEA). The author is the current chairman of the NEA Committee on the Safety of Nuclear Installations' Working Group on Fuel Cycle Safety.

2. OVERVIEW OF THE OECD/NEA WORKING GROUP ON FUEL CYCLE SAFETY

The Nuclear Energy Agency is a specialized agency within the Organisation for Economic Co-operation and Development, an intergovernmental organization of industrialized countries based in Paris, France. In the area of nuclear safety and regulation, the goal of the NEA is to assist member countries in ensuring high standards of safety in the use of nuclear energy, by supporting the development of effective and efficient regulation and oversight of nuclear installations, and by helping to maintain and advance the scientific and technological knowledge base. Within the NEA is the Committee on the Safety of Nuclear Installations

(CSNI), whose mission is to assist member countries in maintaining and further developing the scientific and technical knowledge base required to assess the safety of nuclear reactors and fuel cycle facilities. The Working Group on Fuel Cycle Safety reports to the CSNI, which approves the WGFCS's program of work.

The Working Group on Fuel Cycle Safety brings together representatives of regulatory bodies, their technical support organizations, and operators of nuclear fuel cycle facilities, which comprise a number of interrelated activities, including: uranium mining and milling; uranium refining and conversion to uranium hexafluoride; uranium enrichment; fuel fabrication and storage; spent fuel storage; spent fuel reprocessing; decommissioning of nuclear facilities; radioactive waste management and disposal options (including for spent fuel) and the research and demonstration facilities that support these activities.

The objective of the WGFCS is to advance the understanding for both regulators and operators of relevant aspects of nuclear fuel cycle safety in member countries. Consistent with its mandate, in the last several years, the WGFCS has undertaken activities to address the management of spent nuclear fuel (SNF) and high-level waste (HLW). Specifically, in May 2013, the WGFCS held a workshop on the Safety of Long-Term Interim Storage (LTIS). As a follow-on to that workshop, the WGFCS is now developing a technical opinion paper that summarizes all requirements which should be taken into account when preparing for the long term interim storage of SNF and HLW.

3. SUMMARY OF THE INTERNATIONAL WORKSHOP OF THE SAFETY OF LONGTERM INTERIM STORAGE FACILITIES

On 21–23 May 2013, the WGFCS, in cooperation with the NEA Working Group on Fuel Safety, held a workshop on the Safety of Long Term Interim Storage Facilities (the LTIS workshop), in Munich, Germany. The workshop was hosted by the Gesellschaft für Anlagen und Reaktorsicherheit (GRS) mbH. Ninety specialists from operators, industry, national nuclear authorities, technical support organizations and research institutes — representing 14 NEA countries and two international organizations - attended the workshop. A total of twenty nine papers were presented. The proceedings were published in January 2014 [1].

The objective of this workshop was to discuss and review current national activities, plans and regulatory approaches for the safety of long term interim storage facilities dedicated to SNF, HLW, and other radioactive materials with prolonged storage regimes. It was also intended to discuss results of experiments and to identify necessary research and development (R&D) to confirm the safety of fuel and cask during long-term storage. The workshop had three technical sessions: (1) National Approaches for Long Term Interim Storage Facilities, Safety Requirements; (2) Regulatory Framework & Implementation Issues; and (3) Technical Issues & Operational Experience, Needs for R&D.

3.1. Workshop general conclusions and recommendations from LTIS workshop

The summary below is from the LTIS workshop proceedings [1].

Presentations and discussions at the workshop indicated that there is, in many countries, an interest or need for long term storage of spent nuclear fuel and high-level waste, before disposing in a final repository or reprocessing the used fuel. In some countries a prolongation of the storage period, with respect to original planning, is already anticipated. The main reasons are delays in site selection or planning, and licensing of a final repository, sometimes in combination with a national nuclear policy to abandon reprocessing of spent nuclear fuel

from power reactors. Storage periods in the range of 50 to 100 years and even longer are being discussed. The workshop presentations covered a range of topics, perspectives and professional experience of the presenters, i.e. facility operators, cask vendors, national authorities, experts involved in safety analyses and/or R&D activities, and representatives from international organizations, who provided overview information on past experience and on-going work in their member states as well as international cooperation programs.

Dry cask storage often using dual purpose casks, dry vault storage, and wet pool storage were presented as the main storage concepts currently applied or under development. Generally, the presentations indicated a preference for dry cask storage, but the discussions revealed specific advantages of both the wet and dry concepts. The major advantage of dry cask storage is seen in its reliance on passive components and robustness against external impacts. On the other hand, pool storage provides direct access for monitoring and control of the stored fuel elements, and higher flexibility for further steps in spent fuel management, e.g., packaging and/or conditioning of the fuel elements after storage. It was also noted that use of passive cooling is a feasible option for wet storage as well. In any case, the required level of safety has to be demonstrated and provided by appropriate measures in both dry and wet storage approaches.

The presentations identified a number of important aspects and challenge connected to long term storage: These can be placed into four general categories: (1) technical aspects of ageing; (2) long term operational issues; (3) conceptual aspects of long term storage; and (4) regulatory requirements, licensing and surveillance.

Regarding the technical aspects of ageing, the long term behavior of the stored fuel and the performance of casks are the points of interest. As there is no direct access for monitoring of the fuel during cask storage, a demonstration of long term safety based on reliable data and experience is necessary. Current national practices of safety analyses and demonstration for licensing were presented, which are being approved usually for several decades. Results presented from international cooperation programs and gap analyses confirm the need for additional data and information on long term behavior of the fuel cladding, in particular in the range 50−100 years and beyond, where the fuel temperature decreases, and also for high burn-up and mixed oxide fuel. Also, data for material behavior for cask components, such as metal gaskets, lid bolts, and trunnions for lifting and transport, are needed. Also, different methods of monitoring fuel behavior and cask performance during storage are being developed and tested.

The second topic comprises all issues and tasks regarding long term operation of storage facilities. These are ensuring knowledge management and qualification of operation personnel for more than several decades, record keeping on the stored material with regard to storage and for further use or disposal after storage, a systematic ageing management program as part of the safety management system, monitoring and analyses of operation experience. Also consideration of social aspects (e.g. public confidence and political commitment) has been identified as an important issue.

The topic ‘conceptual aspects’ is related at first to the question of which type of storage should be preferred — dry cask or wet pool storage. As already mentioned, there are advantages and weaknesses in the different storage types, and a decision on the respective type of storage should consider site specific conditions. Another important point is the holistic aspect of long term storage. In order to minimize the number of handling and conditioning processes, and to avoid unnecessary radiation exposure to personnel, long term storage should

be part of an integrated spent fuel management concept, e.g. use of multi-purpose cask for storage shipment and disposal.

The last, but just as important, topic comprises regulatory issues of long term storage. Questions raised at the workshop related to national approaches for licensing long term storage, e.g. for more than 50 years, time limitation of operational licenses, requirements for systematic ageing management, periodic safety review, demonstration and approval of transportability of spent fuel casks for the whole storage period and beyond, anticipated changes in national and international requirements, and finally, observation of the (generally agreed) principle of not burdening unsolved problems of disposal to future generations.

Considering all these issues, which were presented and discussed with high expertise, from different points of view and based on different areas of experience, the following can be concluded:

- Important regulatory aspects of long term storage, such as time limitation of operation license, requirements for a safety case on long term storage, PSR, are compiled in the Specific Safety Guide SSG-15, issued recently by the IAEA;
- Long term storage, in general for 50 years and longer, should preferably be part of an integrated fuel/waste management program taking into account the further use or treatment of the stored fuel/waste beyond storage. A license for long term storage should be periodically reviewed and integrated in the overall spent fuel management, considering verification of long term safety and including social aspects. Appropriate margins and procedures for prolongation of an operation license should be provided;
- A holistic view of the whole process is needed, jointly analyzing all safety functions involved in all back-end stages up to final disposal, which would require a major regulatory effort. A license should as a minimum be connected to a commitment for developing further steps of SF management beyond storage within a certain short term;
- Regarding technical aspects of ageing, the deployment and progress should move towards closing the data gaps identified for long term behavior of fuel rods and fuel cladding, in particular for high burn-up and MOX fuels and new cladding materials. Retrievability and transportability of the fuel after storage is an important aspect. Also, data on cask performance from on-going experiences and testing should be collected, evaluated and published;
- Sufficient adherence should be paid to the non-technical aspects of ageing, in particular knowledge management, record keeping of stored fuel/waste, personal training and planning;
- As both wet and dry storage provide specific advantages, it is up to each country to decide which type of storage would be more favorable under their respective situation and condition;
- In order to assess mechanical behavior of fuel rods, such as hoop stress, creeping, effects of Hydrogen and/or temperature on cladding ductility, swelling of pellets, etc., the availability of relevant experimental data would be useful for improving and validating calculations, models and computer codes (e. g. TRANSURANUS, FRAPCON) for the long term range and for high burn-up fuel. Data for improving and benchmarking temperature calculation models for dry cask storage seems desirable as well;
- Appropriate solutions for long term storage of damaged fuel elements have to be developed;

- The delay in the final decision making results in lost opportunities for process optimization. Increasing the capacity of the pools and dry storage casks is not in the way of optimization.

4. TECHNICAL OPINION PAPER ON THE SAFETY OF LONG TERM INTERIM STORAGE FACILITIES

As follow-on to recommendations from the workshop, and recommendations from NEA's

Committee on the Safety of Nuclear Installations, the WGFCS is developing a technical opinion paper (TOP) that will gather information presented at the workshop. The information will be complemented by information supplied by WGFCS members to develop a picture of the current status of LTIS in NEA member countries. The TOP will provide an overview of LTIS requirements and technical needs, based on approaches in member countries. This activity is being led by the NEA member country of Spain.

The TOP will focus on: national approaches and expectations for LTIS management of spent fuel and HLW; spent fuel and HLW inventories and storage systems, and long term strategies being considered; regulatory framework, policies and regulations; licensing processes and procedures; knowledge, data and regulatory gaps and challenges, and national programs to address them; identification of needs for research and development; and considerations of lessons learned from the Fukushima accident. The task will not duplicate ongoing technical activities (e.g. by the International Atomic Energy Agency, the Electric Power Research Institute, the U.S. Department of Energy). Rather, the working group will coordinate with these organizations and provide a compilation and analysis of the member countries' position on long term interim storage, licensing requirements and gap identification.

As part of this effort, the WGFCS, led by Spain, has developed a questionnaire as a means to systematically collect the necessary information to successfully develop a TOP that provides a comprehensive overview of LTIS, the current national activities, regulatory approaches and requirements, technical needs, and R&D programs to overcome them. The questionnaire, which was sent to the NEA member countries in April 2015, consists of the following questions:

- Describe the national approach, including used or planned funding principles, and expectations for Long Term Interim Storage management of Spent Nuclear Fuel and High Level Waste in your country;
- Describe qualitatively and if possible quantitatively also the SNF/HLW inventory as well as the storage systems used;
- Describe the LTIS strategies being in use or considered;
- Describe the regulatory framework for LTIS: policy and regulations (a. Main limits and acceptance criteria for the environmental effects and public exposures for normal operation and accidents; b. Main safety design and radiation protection requirements)
- Describe the licensing process of the storage application in your country with focus on: supporting safety assessment – base case and specific requirements for LTIS; timeframe and renewal process; transportation considerations, if any, at the time of storage;
- Describe the main identified gaps and challenges for LTIS, as well as the national programs developed to address them. Please include considerations on how cross-cutting issues (knowledge management, recordkeeping) as well as non-technical aspects (e.g. public confidence and political commitment) are addressed;

- Describe the needs for R&D identified for LTIS;
- Describe the considerations on LTIS resulting from the Fukushima lessons learned.

The WGFCS expects to complete the TOP and submit it to CSNI for approval in December 2015.

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CONSEQUENCES OF LONG TERM BLACK OUT ACCIDENT ON SPENT FUEL POOL FOR VVER-1200

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Abstract

Natural disaster occurred at Japan Fukushima Daiichi NPP in March 2011 showed that combination of natural disasters with a long term black out which is not accepted as a design basis accident till that day could be how dangerous. In this study case, inventory calculation in the spent fuel pool, transmission from inventory to source term and atmospheric dispersion calculation of source term were investigated, in a case of long term black out accident that can occur in the AES 2006 design VVER-1200 type of NPP planned to be built in Turkey.

1. INTRODUCTION

On 12 May 2010 in Ankara, an intergovernmental agreement has been signed between Turkish Republic and Russian Federation for constructing four units of VVER 1200 type (AES 2006 design) a third generation pressurized water reactor which has 4800 MW(th) installed capacity at Mersin, Akkuyu site.

In this study, radiological consequences of a possible accident that may occur in the spent fuel pool of a VVER 1200 type NPP in a long term station black out accident which is similar to Fukushima Daichii accident is investigated.

During the accident because of external events it is assumed that electricity cannot be provided to the NPP and diesel generators are out of service.

2. INVENTORY & SOURCE TERM CALCULATIONS

ORIGEN-ARP software is used for inventory calculations [1]. For calculations typical AES 2006 design data given in Environmental Impact Assessment Report of Akkuyu NPP is used [2]. Technical parameters used for AES 2006 design VVER 1200 type power reactor inventory calculations are given in Table 1.

Capacity of the spent fuel pool used in the reactor design is sufficient for spent fuel of 10 years plus all the fuel in the core [2].

During the accident it was assumed that spent fuel pool is full and all the fuel in the core is in the pool.

According to accident scenario it was assumed that all the water covering the fuel is evaporated and the integrity of the fuel clad is lost. Decay heat released by the spent fuel in the pool causes to evaporate the water and the spent fuel in the pool started to uncover in 2.29 days.

The waiting period of the fuel in the spent fuel pool is used in inventory calculation. Additionally, spent fuel uncover period which is 2.29 day is also taken into account during spent fuel inventory calculations.

TABLE 1. TECHNICAL PARAMETERS FOR NPP

Nuclear Power Plant Life Time	60 yrs
Thermal Power	3200 MW(th)
Average burn up ratio	55.5 MWday/kg U
Average enrichment of the fuel assembly placed in to the core	4.79%
Number of fuel assembly in the core	163
Number of fuel assembly changed every year	42
Uranium amount in 42 fuel assembly	19 768 kg
37 of 42 fuel assembly stays in the core for 4 years	
5 of 42 fuel assembly stays in the core for 3 years	
Average time of stay in the core of 42 fuel assembly	1330 days
Amount of water in the spent fuel pool	1529.3 m^3 + 750 m^3

Since radionuclide specific release ratios are not available at containment outer boundary of VVER-1200 NPP, they methodology and numerical values of NRC NUREG 1150 [3] are utilized in this work.

According to NRC NUREG 1150 radioactive isotopes are divided in to 9 groups according to their chemical and physical behaviors. These groups and release ratios are given in Table 2.

TABLE 2. ISOTOPE GROUPS AND RELEASE RATIOS.

Groups	Release ratio
Noble Gas Group (Xe, Kr)	0.95
Halogen Group (I)	0.35
Alkali Metal Group (Cs, Rb)	0.25
Tellurium Group (Te, Se, Sb)	0.15
Barium Group (Ba)	0.04
Strontium Group (Sr)	0.03
Cerium Group (Ce, Pu, Np)	0.01
Rutenium Group (Ru, Co, Rh, Pd, Mo, Tc,)	0.008
Lanthanum Group (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Am, Cm, Y)	0.002

Source term calculated by using release ratios given in Table 2 and core inventory is given in Table 3.

TABLE 3. SOURCE TERM

Isotope	Activity (Bq)	Isotope	Activity (Bq)
Kr-85	1.04E+17	Sr-89	9.15E+16
Kr-85m	4.72E+12	Sr-90	2.92E+16
Kr-87	3.94E-01	Sr-91	4.32E+14
Kr-88	1.10E+10	Ce-141	5.01E+16
Xe-131m	3.70E+16	Ce-143	9.81E+15
Xe-133	4.73E+18	Ce-144	4.62E+16
Xe-133m	1.02E+17	Pu-238	5.27E+14
Xe-135	4.01E+16	Pu-239	3.50E+13
I-131	8.54E+17	Pu240	5.43E+13
I-132	8.09E+17	Pu-241	1.38E+16
I-133	1.71E+17	Np-239	2.20E+17
I-134	8.65E-09	Ru-103	3.64E+16
I-135	5.29E+14	Ru-105	9.20E+10
Cs-134	2.48E+17	Ru-106	1.66E+16
Cs-136	3.28E+16	Rh-105	5.43E+15
Cs-137	3.43E+17	Mo-99	2.08E+16
Rb-86	1.21E+15	Tc-99m	2.02E+16
Te-127	2.78E+16	La-140	1.06E+16
Te-127m	6.93E+15	Zr-95	1.06E+16
Te-129	1.41E+16	Zr-97	4.38E+14
Te-129m	2.19E+16	Nb-95	1.10E+16
Te-131m	1.39E+16	Am-241	2.10E+13
Te-132	3.36E+17	Cm-242	2.40E+14
Sb-127	2.28E+16	Cm-244	1.25E+14
Sb-129	4.62E+11	Y-90	1.95E+15
Ba-140	1.90E+17	Y-91	8.02E+15

3. RESULTS AND CONCLUSIONS

For atmospheric dispersion calculation PC COSYMA software based on gaussian plume model was used [4].

For 15 different atmospheric conditions, calculations were made for exposed doses against distance. For all the calculations it was assumed that day time, summer season and total dose was exposed in two days. Release height is assumed as 100 meters. Atmospheric conditions used for calculations were given in Table 4.

TABLE 4. ATMOSPHERIC CONDITIONS USED FOR CALCULATIONS

Condition	Stability class	Wind speed (m/s)	Condition	Stability class	Wind speed (m/s)
1	A	1	9	C	7
2	A	2	10	D	4
3	B	1	11	D	6
4	B	2	12	D	7
5	B	4	13	E	2
6	C	2	14	E	4
7	C	4	15	F	2
8	C	6			

Atmospheric dispersion calculation results performed for 15 different atmospheric conditions, for each condition exposed effective doses approximately at 10 km, 15 km and 20 km were given in comparison at Table 5.

Since release ratio from containment used for transmission from inventory to source term is not specific to VVER-1200 type NPP and accident, most of the uncertainties originate from there.

Most important assumptions made during atmospheric dispersion calculation were constant wind direction and single release of source term assumptions. In a more realistic calculation noble gas group which is very important part of the source term will be released from containment rapidly but it will spread over time for the release of heavy isotope. From this point of view made assumption is conservative.

According to the results of the calculations, except the conditions were in the most conservative stability class of E and F the wind speed is 2 m/s, it was observed that beyond 20 km annual dose limits for population was not exceeded. In these two conditions where dose limits were exceeded, short term sheltering is thought to be more than enough.

TABLE 5. COMPARISON OF EXPOSED EFFECTIVE DOSES AT DIFFERENT DISTANCES FOR DIFFERENT ATMOSPHERIC CONDITIONS

Conditions	Stability class	Wind speed (m/s)	Effective dose at ≈ 5 km (mSv)	Effective dose at ≈ 10 km (mSv)	Effective dose at ≈ 15 km (mSv)	Effective dose at ≈ 20 km (mSv)
1	A	1	4.386	1.809	1.315	0.8986
2	A	2	2.539	0.9697	0.7435	0.5169
3	B	1	5.505	2.290	1.538	1.094
4	B	2	3.246	1.288	0.8892	0.6492
5	B	4	1.733	0.7126	0.4907	0.3706
6	C	2	4.746	1.812	1.274	0.8767
7	C	4	2.610	1.032	0.7169	0.5116
8	C	6	1.796	0.7310	0.5193	0.3626
9	C	7	1.554	0.6379	0.4553	0.3243
10	D	4	4.176	1.433	0.9958	0.7125
11	D	6	2.880	1.018	0.7257	0.5173
12	D	7	2.493	0.8895	0.6379	0.4680
13	E	2	15.18	4.224	2.949	2.023
14	E	4	8.484	2.435	1.690	1.207
15	F	2	37.84	8.082	4.701	2.943

For conducting a specific study to the VVER-1200 type Akkuyu NPP which will be built in Turkey, design data which will be presented in Akkuyu Nuclear Power Plant Safety Analysis Report and site parameters which will be presented in Akkuyu Nuclear Power Plant Site Parameters Report are required. Since these data are not present and will be classified as confidential calculations were made as conservative as it could be.

Regarding the obtained results, more realistic calculations which can be conducted with Akkuyu NPP design data and site parameters assumed that effective dose beyond 20 km will not exceed annual public dose limit.

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INTERIM STORAGE OF SPENT FUEL IN GERMANY — HISTORY, STATE AND PROSPECTS

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Abstract

Since the concept of dry interim storage of spent nuclear fuel elements in transport casks had been developed by the former DWK (Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen e.V.) in the late 1970s and had been first evaluated by the Reactor Safety Commission in 1979, development has been going on. Dry interim storage passed a lot of modifications in different fields, but the fundamentals of this first concept on which the dual purpose cask used for transport and storage is based on are still up to date. Changes also affect the authorities, experts and operators by generation alternation. Thus, an essential aspect of age management is transfer of knowledge and experience. This paper gives an overview over the licensing activities in the field of dry interim storage of spent fuel in Germany within the last 35 years up to now and closes outlining future perspectives. The Federal Office for Radiation Protection is the competent licensing authority for storage of spent nuclear fuel since 1989.

1. CONCEPT OF DRY INTERIM STORAGE IN TRANSPORT CASKS — BEGINNING AND EARLY DEVELOPMENTS

The concept of dry interim storage of spent nuclear fuel elements in transport casks has been developed by the DWK (Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen e.V.) in the late 1970s [1].

In those years in Germany, interim storage of spent fuel in external storage facilities was projected as an interim step of spent fuel management. It was intended to be established to bridge the time gap until reprocessing of spent fuel or direct disposal without reprocessing could be realized in sufficient manner [2]. Thus, a large waste management center in Gorleben, including — among others — a reprocessing plant and a wet interim storage facility was projected and applied for in 1977. Another wet interim storage facility was planned in Ahaus to increase capacities [3, 4]. It was applied for in 1978.

The common concept for storage of spent fuel elements at that time was wet interim storage in water basins, as practiced in the nuclear power reactors. Then as a novelty, the DWK developed the concept of dry interim storage of spent fuel in transport casks [5]. In this concept, the transport cask ensures the main safety functions as well of the interim storage of its inventory. That implies the safe enclosure of the radioactive inventory and protection against mechanical impacts, as well as criticality safety, radiation shielding and decay heat removal. The cask construction with cooling fins on the cask surface allows passive cooling of the inventory by convection. The safe enclosure of the inventory is assured by a double lid system, consisting in two independent barriers with metallic gaskets, of which the leak tightness is constantly monitored by a pressure switch, which detects pressure changes in the interspace of the lid barriers. Dry interim storage according to this concept was considered to be safe for storage periods of at least 30 years. As advantages of this concept were considered especially the independence of active cooling systems and the safe enclosure of the spent fuel in the cask, enabling easier handling of the radioactive material.

This concept was examined by the Reactor Safety Commission[1], which in September 1979 confirmed feasibility and safety of dry interim storage in transport casks as an equivalent alternative concept for storage of spent fuel.

After this positive statement, the DWK filed applications for dry interim storage in storage facilities located in Ahaus and in Gorleben in October 1979 and in April 1980 respectively, each one for a storage period of 40 years with a capacity of 1500 tU [6]. These applications were filed to PTB (Physikalisch-Technische Bundesanstalt), the predecessor of BfS as licensing authority.

2. FIRST LICENSES FOR DRY INTERIM STORAGE IN GERMANY

The legal basis for interim storage of spent fuel is the article 6 of the Atomic Energy Act (AtG). This article specifies the licensing conditions, which have been unchanged on from the very first version of the Atomic Energy Act, dated on 23 December 1959, until now. Thus, a storage license has to be granted, if the need for the applied interim storage is demonstrated and if the licensing conditions, as following, are met:

- Reliability and technical qualification of the applicant and the responsible persons;
- Precaution against harm and damages caused by the storage according to the state of the art of science and technology;
- Sufficient financial security to cover the liability for damages caused by the storage;
- Protection against disruptive actions or other third-party intervention.

Additionally, the construction of the storage facility required a planning and building permission granted by the local building authority.

The licensing procedures for both the storage facilities in Ahaus and Gorleben were conducted according to the regulations for formalized licensing procedures, each including a public participation procedure. For Gorleben, the basic application documents were disclosed for public inspection in October and November 1980 [7, 8]. In total, 225 objections were raised against the project, which were discussed within a hearing of all parties in January 1981. For Ahaus, the disclosure of the application documents took place from February to April 1983. Concerning this project, 5186 objections were raised and discussed in June 1983.

The PTB as licensing authority verified the compliance of the applied storage facilities in Ahaus and Gorleben with the licensing conditions according to article 6 AtG supported by external experts. The Federal Institute for Materials Research and Testing (BAM) was assigned to the examination of all fields of cask safety. The technical inspection agency TÜV Hannover was assigned to the verification of safety of the storage concept, including e. g. the site and environs of the storage facility, the storage building and technical equipment, safety of operation and radiological aspects. Aspects of criticality safety were examined by PTB, and the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) examined all topics of physical protection of the storage facility.

At the end of 1982, the external experts BAM, TÜV Hannover and GRS finished their expertise reports and gave positive statements on both the projected storage facilities in

[1] The Reactor Safety Commissions (RSK) was founded in 1958 and advises the Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB) in safety-related matters and thus matters concerning the physical protection of nuclear installations and radioactive waste management (www.rskonline.de).

Gorleben and Ahaus. Referring to the expertise reports, also the Reactor Safety Commission (RSK) positively evaluated the projected interim storage facilities in Ahaus and Gorleben in a further statement in March 1983, and recommended to grant the licenses [3]. The expertise reports of TÜV Hannover and BAM were additionally disclosed for public inspection from February to April 1983.

The first license for dry interim storage of spent fuel in Germany was granted for the central storage facility in Gorleben (TBL-Gorleben) on 5 September 1983 by the PTB. It permitted the storage of spent fuel elements from light water reactors (PWR and BWR) in transport casks of the types CASTOR Ia, Ib, Ic, IIa. These cask types could take up four, respectively nine, fuel elements each. In total, it included the storage of 1500 tons of uranium in transport casks on 420 cask positions for a period of 40 years, until September 2023. As owner and operator of the TBL Gorleben had been founded the Brennelementlager Gorleben GmbH (BLG), who is in this position until now.

The license for dry interim storage in the storage facility in Ahaus (TBL Ahaus) was granted on 10 April 1987, permitting the storage of in total 1500 tons of uranium in transport casks on 420 cask positions for a period of 40 years, until April 2027. It was granted for storage of spent fuel elements from light water reactors (PWR and BWR) in transport casks of the types CASTOR Ia and IIa. As owner and operator of the TBL Ahaus had been founded the Brennelement-Zwischenlager Ahaus GmbH (BZA), who is in this position until now.

In October 1989, the responsibility for licensing of interim storage of spent fuel according to article 6 AtG went over to the newly founded Federal Office for Radiation Protection (Bundesamt für Strahlenschutz — BfS). The BfS entered into force as licensing authority and continued the current licensing procedures.

On 17 March 1992, a supplement to the storage license for TBL Ahaus was granted. Due to a supplementary application of 1984, additionally the storage of spherical fuel elements from the High-Temperature Reactor (HTR) in Hamm-Uentrop, a gas-cooled pebble bed reactor, was permitted. As an appropriate transport and storage cask for these spherical fuel elements, the CASTOR THTR/AVR had been developed. The licensing procedure for this matter included further public participation procedures in 1989 and 1990, including the participation of citizens of the Netherlands.

The first cask emplacement in the TBL Ahaus took place on 25 June 1992 with a transport- and storage cask CASTOR THTR/AVR with spherical fuel elements of the HTR. This was at once the first cask emplacement for dry interim storage in Germany at all. In the TBL Gorleben, the first cask was emplaced in April 1995. It was a cask of the type CASTOR IIa with spent fuel elements from the PWR in Philippsburg.

3. FURTHER DEVELOPMENTS IN THE 1990S

In June 1993, a license for one more storage facility was granted, targeted on interim storage of the spent fuel elements from the AVR test reactor in Jülich for a storage period of 20 years. The AVR test reactor was a gas-cooled pebble bed reactor, and the spherical fuel elements were to be stored in transport and storage casks of the type CASTOR THTR/AVR. This storage facility in Jülich went into operation in August 1993 with the first cask emplacement. Presently, in Jülich in total 288 161 spent fuel elements are stored in 152 casks, of which the last one was loaded in September 2009.

In 1993 and 1994, the BLG filed new applications to the BfS, intending a complete renewal of the existing storage license of the TBL Gorleben. This was due to the technical developments, which had taken place since the first license of 1983, and had led to a considerable number of amending licenses. The new application targeted to a broader matter of the license, including more different types of transport and storage casks and a higher total storage capacity of 3800 tons of heavy metals. In addition to the existing license, the storage of vitrified highly active waste from reprocessing of spent fuel elements was applied for [9]. For these matters, the BLG filed a new comprehensive safety report. On this basis, a public participation procedure was carried out in 1993. About 14000 objections were raised against this project, which were discussed in September 1993.

The new storage license according to article 6 of the Atomic Energy Act for the TBL Gorleben was granted on June 2nd of 1995, comprising the storage of up to 3800 tons of heavy metal in transport and storage casks on 420 cask positions, with a total activity of 2×10^{20} Bq and a total thermal output of maximum 16 MW. The storage period is limited to 40 years, until 31 December 2034. This license is still valid, in a version modified by presently four amending licenses – the latest one granted in January 2010. The amending licenses permit, for example, the use of new cask types for the storage of vitrified highly active waste, e. g. the French cask type TN85 or the cask type CASTOR HAW28M, each taking up 28 canisters of vitrified highly active waste.

Likewise, the BZA and the GNS in 1995 filed a new application for the TBL Ahaus to BfS, intending to obtain a new and comprehensive storage license for the TBL Ahaus. The applied matter included, besides spent fuel elements from light water reactors and from the HTR, also spent fuel elements from research reactors. A comprehensive safety report was filed in January 1996, on the basis of which a public participation procedure was carried out. 2317 objections were raised against the project, which were discussed in September 1996.

The new storage license according to article 6 of the Atomic Energy Act for the TBL Ahaus was granted on 7 November 1997. It permits the storage of up to 3960 tons of heavy metal in transport and storage casks on 420 cask positions, with a total activity of 2×10^{20} Bq and a total thermal output of maximum 17 MW. The storage period is limited to 40 years, until 31 December 2036. This license is still valid, in a version modified by presently six amending licenses — the latest one granted in May 2010. One of these, for example, permits the storage of spent fuel elements of the research reactor in Dresden-Rossendorf in transport and storage casks of the type MTR2.

On 5 November 1999, a storage license for a storage facility on the site of the NPP Greifswald in Rubenow, ZLN (Zwischenlager Nord), was granted. It was targeted on the interim storage of the spent fuel elements from the NPP Greifswald and the NPP Rheinsberg, both located in the former GDR. For the fuel elements originating from these NPPs, which were PWRs of the types WWER-440 and WWER-70, the new cask type CASTOR 440/84 was developed. ZLN is a part of a large storage facility for radioactive waste from the decommissioning of the NPP Greifswald. The licensed capacity of ZLN is up to 585 tons of heavy metal. It comprises 80 cask positions, a total activity of 7.5×10^{18} Bq and a total thermal output of 600 kW. The storage period is limited to 40 years, too, until 31 October 2039. Within the last years, this license has undergone seven amendments, the latest one in April 2010. These were targeted — among others — on the storage of further inventories, e.g. vitrified high active waste from the prototype reprocessing plant in Karlsruhe, especially for which a vitrification plant was erected and operated. Another amending license permits the storage of spent fuel elements from the KNK (compact natrium cooled reactor), which was

operated in Karlsruhe, and from the experimental nuclear powered ship “Otto Hahn” in transport and storage casks of the type CASTOR KNK.

From 1999 on, the phase-out of nuclear energy use in Germany was initiated, which lead to an amendment of the Atomic Energy Act in April 2002. Besides arrangements concerning the nuclear phase-out, the new version of the Atomic Energy Act defined a new waste management concept. Other than the precedent concepts, this new concept excluded reprocessing of spent nuclear fuel and determined direct disposal in a geological repository as the only legal way of waste management. The delivery of spent nuclear fuel elements to reprocessing plants finally was prohibited by 1st July 2005.

As consequently higher capacities for interim storage of spent fuel were necessary, the operators of the NPPs were obliged to establish interim storage facilities on the sites of their NPPs. Thus, twelve on-site interim storage facilities were projected and underwent licensing. The storage facilities were projected as storage halls of reinforced concrete, in which transport and storage casks of the types CASTOR V/19, which takes up 19 PWR fuel elements, and CASTOR V/52, taking up 52 BWR fuel elements, can be stored. Additionally, five short time storage facilities were projected and applied for to bridge the time until the newly erected interim storage facilities could take up operation. In these short time interim storage facilities, the casks could be stored horizontally in small concrete housings for a period of up to five years.

For the interim storage facilities, which had been applied to after 14 March 1999, an environmental impact assessment had to be carried out within the licensing procedure. Thus, for eleven of the twelve interim storage facilities and for all short time storage facilities, an environmental impact assessment had to be carried out. For the interim storage facilities, a common environmental impact assessment, carried out by BfS, covered as well the storage license as the building permission. Each licensing procedure, as well for the twelve interim storage facilities as for the short time storage facilities, was carried out in a formalized way, including a public participation procedure.

Finally, the twelve licenses for the interim storage facilities were granted by BfS from November 2002 until December 2003. Additionally, four licenses for short time storage facilities were granted in 2001 and 2003. The short time storage facilities were in operation for up to 6 years, until the loaded casks could be transferred into the new built interim storage facilities. After the building permission had been granted, the erection of the storage facilities took 2−3 years. The first interim storage facility, for which the building permission was granted earlier, could be put into operation in December 2002; the other facilities went into operation with the first cask emplacement from February 2006 on to June 2007.

4. ACTUAL SITUATION

Hence, there are presently 16 interim storage facilities in operation in Germany. By 31 December 2014, in total 1020 transport and storage casks were stored in the German storage facilities.

As well as for the central storage facilities TBL Gorleben, TBL Ahaus and ZLN, also for the on-site interim storage facilities the storage licenses are constantly undergoing modifications. These modifications concern especially the properties of the inventory, such as for example enhancements of the initial enrichment or the fuel burn-up, or higher numbers of MOX fuel elements per casks. Also, due to updated regulations, the transport and storage casks CASTOR V/19 and CASTOR V/52 have been modified and are applied for or, respectively,

have recently been licensed for storage. Moreover, the French cask type TN24E, manufactured by TNI, has been applied for in seven interim storage facilities for storage of spent PWR fuel elements. Furthermore, in all interim storage facilities, the cranes are projected to be upgraded to meet the highest requirements of nuclear technology regulations to avoid any drop of a storage cask. Additionally, for all interim storage facilities, the enhancement of physical protection of the storage facilities from disruptive actions and third party interventions is projected and applied for. Altogether, there are about 50 current licensing procedures in progress at BfS.

Notwithstanding the diverse modifications of the licensed interim storage, the licensing conditions specified in article 6 of the Atomic Energy Act, which has been the legal basis for interim storage of spent nuclear fuel from beginning on, have remained unaltered. Also the underlying concept for dry storage of spent nuclear fuel in transport and storage casks has persisted the last 36 years unaltered in its basic principles. Beginning with its statement dated from 1979, the RSK, and since 2008 the ESK[2], have accompanied the development of interim storage of spent fuel by various recommendations and advises. Thus, the central piece of regulations for interim storage of spent nuclear fuel is the recommendation of the ESK "Guidelines for dry cask storage of spent fuel and heat-generating waste" in its latest version from June 2013 [4]. Although the recommendation has continuously taken up recent developments, as e.g. the interim storage of vitrified highly active waste, the essential basics of the concept of dry interim storage in transport and storage casks, as pointed out in the RSK statement from 1979, can still be found in the present version of the recommendation.

Nonetheless, within in the 36 years of interim storage developments, important changes have affected the authorities, experts and operators by generation alternation. Over the years, many of the faces working in the field of interim storage have already been replaced by others. In the recent years, the people who experienced the developments of dry interim storage from its beginning on are getting rare. Thus, an essential aspect of age management is transfer of knowledge and experience, which was step by step gained over the years, to the next generation. This can be achieved best by continuous transfer of knowledge and experience in overlapping working constellations. But also this paper intends to give a well-grounded summary of interim storage history in Germany.

5. PROSPECTS

An outlook to further developments of interim storage shows that still various themes are in advance.

Present amending applications concern various themes as mentioned above. Prospectively, the storage facilities on the sites of the NPPs are especially facing matters associated with the further operation and finally the decommissioning of the NPPs. These can be, for example, the storage of fuel elements with defect fuel rods and other non-standard fuel elements, accruing before depletion of the core.

An important licensing matter is the storage of residues from reprocessing of spent fuel in France and Great Britain. While the return of high active vitrified waste (CSD-V) from the AREVA reprocessing plant in La Hague, France, has been concluded in November 2011, the

[2] In 2008, the Nuclear Waste Management Commission (Entsorgungskommission - ESK) has been founded equivalent to the RSK as advisory organization concerning radioactive waste management (www.entsorgungskommission.de).

return of CSD-V from Sellafield, Great Britain, is still matter of ongoing licensing procedures. As well, ongoing licensing procedures are related to the interim storage of CSD-B (Colis standards de déchets boues) and CSD-C (Colis standards de déchets compactés) from France, which will be returned within the next years. For the return of CSD-C from France a new cask type, TGC 27, a joint venture of GNS and ARVEA, is currently in development. It is planned to be stored in the TBL Ahaus. Due to a recent amendment of the Atomic Energy Act, the CSD-V and CSD-B have to be stored in on-site interim storage facilities.

Another licensing matter is the storage of spent fuel from research reactors. For the TBL Ahaus, an application has been filed for storage of spent fuel elements of the FRM II in Munich, which is a neutron source for research and medical therapy. These fuel elements are projected to be stored in a new cask type MTR3, which is similar to the MTR2, the casks for storage of spent fuel elements from the research reactor Dresden-Rossendorf, which is stored in the TBL Ahaus.

A future theme will be the management of the spent fuel presently stored in the German interim storage facilities after expiration of the 40 years storage period, which will be the first for the TBL Gorleben in 2034. The first cask reaches its licensed storage period in 2032, 40 years after its loading and closure in June 1992.

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LESSONS LEARNED FROM A PERIODIC SAFETY REVIEW, APPLIED TO THE DESIGN OF NEW NUCLEAR FUEL CYCLE FACILITIES

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Abstract

During the last safety reassessment of a spent fuel reprocessing plant operated by AREVA NC in La Hague (France), IRSN reviewed the experience feedback on incidents as well as the compliance of several equipment of the primary containment to their safety requirements.Further investigations are now necessary to control some risks poorly anticipated at the design stage. They give information guidance for the design of new nuclear facilities.For example, the systems for the detection of a leak of the primary containment in units dealing with uranium or plutonium involve detecting the arrival of a large amount of liquid in the secondary containment. The experience feedback showed that the related systems failed to detect leakages of radioactive materials because the leaks occurred at process steps where uranium and plutonium were in solid state, or because of a very low flow of a liquid leak on warm surfaces of the equipment. Correctives measures consist in conducting periodic observations of the secondary containment below the equipment. Such observations may be difficult because of the absence of port-holes, the impossibility to introduce easily cameras, the low level of light in the cells, etc. Furthermore specific tools for non-destructive testing methods are developed, in order to monitor the residual wall thickness of equipment, in order to confirm the corrosion kinetics. But their deployment in hot cells (containing radioactive material) is limited by the difficulties to introduce these devices into the cells and to position these devices on the most useful locations to check the equipment. The anticipation of these controls in the design stage would have made it possible to facilitate them and to better control the aging of facilities and the absence of leaks. The above mentioned lessons are already applied by IRSN for the safety assessment of new nuclear fuel cycle facilities.

Key Words: safety review, anticipation in the design stage, to control the aging of nuclear facilities, leaks detection.

1. INTRODUCTION

The life of a nuclear installation includes several phases:

- The definition of principles governing the operation of the process and risk control;
- The design and the production of documents and safety analyzes;
- The construction and the phases of tests before the commissioning;
- Operation and maintenance;
- The operating shutdown and dismantling operations.

2. SAFETY REASSESSMENT

The French nuclear licensees carry out at regular intervals (every 10 years according to the current regulations) a review of the safety of nuclear facilities; the objectives of the review are:

- To check the conformity of the facility in its requirements in connection with the safety in an independent way of normal operating procedures;
- To revisit the provisions of control of its aging;
- To assess its safety level in the light of the evolution of regulations, of the latest state of the art, of its environment, and of the best practices in the field of nuclear safety and radiation protection available at the time of the review, taking into account experience feedback of the facility, and similar facilities.

IRSN examined during five years the safety review case of the UP3-A plant of the facility at La Hague (France), operated by AREVA NC. The findings of this investigation were

presented in March 2015 to the French Nuclear Safety Authority (ASN). This plant, built mainly between 1986 and 2002, is dedicated to the treatment of nuclear spent fuel to separate and condition the re-usable materials in the fuel cycle (uranium and plutonium) as well as the waste resulting from this treatment (vitrified waste containing fission products and minor actinides, compacted waste containing "hulls" (bits of cladding) and the top and bottom nozzle of the fuel assemblies, technological wastes, etc.). Many different mechanical and chemical processes are conducted in the plant. The risk control for these processes, which might involve radioactive material in different physico-chemical forms, is based on many different principles, however they all result from the defense-in-depth principle.

Waste produced by the processing operations, in particular the vitrified and compacted waste (Fig. 1), are currently stored in the facility at La Hague awaiting their transfer to a future final waste repository.

The safety review (see Fig. 2) is in particular an opportunity to address in detail and to treat topics that have not been sufficiently anticipated in the design of the plant. Today some of the topics require new operating provisions, additional investigation or expansions, etc. Such topics, even if through these additional provisions they do not call into question the continuation of the operation in the years to come, would have led to a different design of certain units of the plant. The issues encountered during the ten yearly safety review provide important data to improve the design of new nuclear facilities in order to be even more in compliance with the safety requirements.

FIG. 1. Cutaway of a glass canister (left) and a compacting press for metallic waste (right).

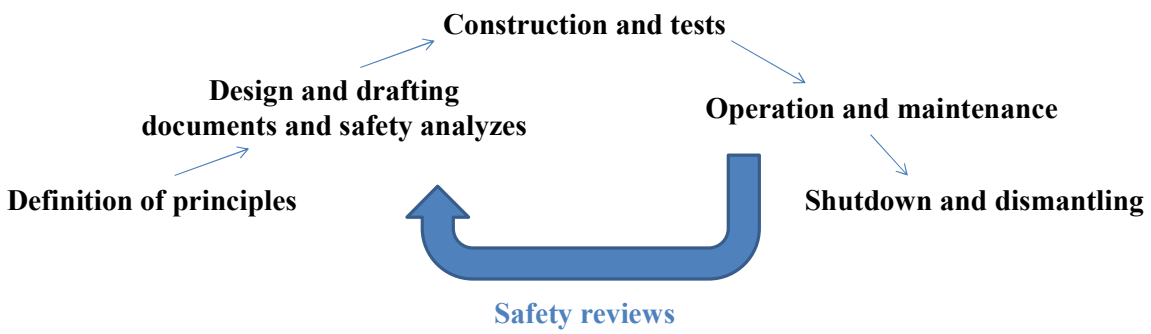

FIG. 2. Illustration how the safety review is set up for the life time of an installation enabling to improve the design of new facilities.

3. LESSONS FOR THE DESIGN OF NUCLEAR FACILITIES

3.1. Need for clear and verifiable requirements

The documents for the safety of facilities contain explicit requirements concerning the operation of processes and risk control, for example the ranges of limit values or a design with hazards such as earthquakes. Procedures are designed to ensure compliance with certain of these requirements. However, if the equipment does not require intervention or do not work at all times or if the equipment is static (anchors of some electrical equipment or handling crane, heat exchangers, etc.), the verification of compliance with certain requirements however important is not necessarily foreseen except sometimes during the ten yearly reviews.

Therefore a safety evaluation approach must be based on explicit requirements clearly identified. The way in which they will be checked must be defined at the earliest stages and concern all equipment involved in controlling risks, in normal or degraded operation, with a procedure of verification formalized and an appropriate frequency.

3.2. Document management is needed for maintenance operations

Even if the maintenance operations on the equipment is incorporated into the design, for example by implementing engines outside the hot cells (containing radioactive material), that are necessary for the operation of equipment in the hot cells, the conservation of technical features and specifications of supplying the replaceable equipment to ensure the availability of spare parts has to be anticipated at the earliest.

The information on the equipment must be preserved and the way in which the safety baseline will evolve in the light of changes or events is also to anticipate.

3.3. Need to anticipate the phenomena of aging and the consequences

The review of compliance is used to compare equipment with the guidelines at any given moment, in particular to verify compliance with the safety requirements. The consideration of aging of equipment and structures contributes to anticipate the phenomena which could query, ultimately, compliance with the safety requirements.

3.3.1. Anticipating the phenomena involved in order to have good margins in the design

In order to comply with the requirements for leaks and for rigidity of equipment and to reduce the risk of dispersal of radioactive material, the phenomena of degradation of equipment (e.g. corrosion, erosion, etc.) are considered on the basis of operating experience feedback of the facility since its commissioning, and are compared with the data selected for the design of the plant. For example, it was highlighted that certain corrosion phenomena have been potentially underestimated in the design, due to the presence of impurities in quantities greater than expected (e.g. as a result of modifications in the characteristics of substances used or modifications in the process: recycling of reagents to limit the quantity of effluent that could lead to the presence of impurities or accumulation loops, etc.) or corrosion mechanisms accelerated as a result of the presence of deposits in the equipment. It appears important to have good margins in the design especially if the phenomena involved are still poorly known.

3.3.2. Anticipating the means for monitoring the phenomena

Thickness measurements in situ monitoring of equipment may be necessary to determine with robustness the extent of degradation, to evaluate the effects on the life span of the equipment and to anticipate their replacement. However, in the spent fuel reprocessing plants, many facilities are located in closed hot cells, inaccessible to operators because of their high radiation level. Also, new tools have been developed in order to monitor the residual wall thickness of equipment, to confirm the corrosion kinetics. But their deployment in hot cells is limited by the difficulties to introduce these devices into the cells and to position these devices on the most useful locations to check the equipment. How to carry out such verification measures should be better considered in the future design of nuclear facilities.

In addition, destructive tests are sometimes necessary to ensure that the characteristics of materials meet the requirements. In most cases, it is not possible for safety reasons to take samples, necessary for destructive tests, from the existing equipment in nuclear facilities. Therefore it is recommendable during the construction phase of a nuclear facility to keep representative samples to be tested under similar aging conditions as the equipment installed. For example, samples of supporting pads present in the fuel storage pools have been kept for over thirty years in similar conditions as the real ones in order to be able to check their present state. For the same reason samples of thermal protection for concrete structures in recent vitrified waste storage facilities were retained in order to be able to monitor their aging evolution over time.

3.3.3. Anticipating the means of direct deviation detection

The review of experience feedback has highlighted that despite the means foreseen during the design phase, the possibility of leakage of process equipment and the inadequacy of detection devices provided for the design cannot be set-aside. For example, the leak detection devices in some of the hot cells in units dealing with uranium or plutonium were designed to detect the arrival of a large amount of liquid in the secondary containment located under the equipment. However, experience feedback has shown that a leak of low flow rate of solution along warm surfaces of the equipment was not detected by these devices, because the limited quantities of liquid were already evaporated before reaching the secondary containment and consequently did not trigger the alarm of the detection system.

Moreover in some steps of the process, where the radioactive materials are solid, a dispersion of radioactive material will not necessarily be detected by the above mentioned devices provided by the design. Periodic checks, in particular visual, of the state of certain hot cells have been put in place further to the deviations, in order to detect as soon as possible any accumulation of radioactive materials due to leaks. However, direct observation is not always easy, due to the configuration of the premises and the absence of port-holes (lack of lighting, hidden areas, etc.). The definition during the design phase of areas to be monitored will facilitate the implementation of these periodic observations.

3.3.4. Anticipating ways to repair

In certain cases where the equipment degradation is still poorly known, and where the opportunities are limited because of high radiation levels, in spite of the provisions foreseen in the design, it is possible that after on several decades of operation, some equipment need to be repaired or replaced. Consideration may be given to provide more space and connections available in order to mitigate as soon as possible the consequences of an unavailability.

4. CONCLUSION

The safety review of the UP3-A plant of the AREVA NC La Hague nuclear facility in France highlighted topics which must be better anticipated in the design phase of nuclear facilities containing mechanical or chemical processes, in particular concerning aging control. These lessons are already applied by IRSN for the safety assessment of new nuclear fuel cycle facilities.

This exercise to re-assess a large nuclear facility, which necessitated an analysis by IRSN spread over five years, enables to better the defense-in-depth. More generally, IRSN considers that the periodic reviews concerning the safety of fuel cycle facilities, every 10 years in France, constitute an essential process for the maintenance and continuous improvement of the safety level in nuclear facilities.

ASPECTS OF SPENT FUEL BEHAVIOR ASSESSMENT FOR TRANSPORT PACKAGES

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Abstract

Transport packages for spent nuclear fuel have to be assessed with respect to specific transport conditions which are defined in the safety regulations of the International Atomic Energy Agency. In general, gastight fuel rods constitute the first barrier of the containment system. The physical state of the spent fuel and the fuel rod cladding as well as the geometric configuration of the fuel assemblies are important inputs for the evaluation of the package safety under transport conditions. The objective of this paper is to discuss the methodologies accepted by BAM for the authority assessment of spent fuel behavior within the package design approval procedure of German package designs. In particular, cracks or failures in the fuel rod cladding can occur under regulatory transport conditions. These defects can cause the release of gas, volatiles, fuel particles or fragments into the package cavity and have to be considered properly in the safety analysis. Another issue is the transport of defective fuel rods. One concept is to use special canisters which can be handled like fuel assemblies. This concept requires additional assessment concerning drying, sealing and the mechanical and thermal design of such canisters. The package as a mechanical system is characterized by a complex set of interactions, e.g. between the fuel rods within the assembly as well as between the fuel assemblies, the basket, and the cask containment. This complexity together with the limited knowledge about the material properties and the variation of the fuel assemblies regarding cladding material, burn-up and the operation history makes an exact mechanical analysis of the fuel rods nearly impossible. The simplified approaches to consider conservatively spent fuel behavior currently accepted by BAM are presented here.

1. INTRODUCTION

The Federal Institute for Materials Research and Testing (BAM) is one of the two German competent authorities with responsibilities for mechanical, thermal and containment safety assessment within package design approval procedure. The assessment of shielding and criticality safety is incumbent on the Federal Office for Radiation Protection (BfS). But BAM also examines the geometrical and material assumptions of the fuel assemblies under the specific transport conditions which are applied as input data in the shielding and criticality safety analysis.

A spent fuel related issue with presently rising interest are special solutions for the transport of defective fuel rods, e.g. by encapsulation. Although the protection functions for these encapsulations are transferable, the assessment work is altereted and extended. In the following sections, our practical approaches with respect to spent fuel assemblies and defective fuel rods are described.

2. PROTECTION GOALS AND REQUIREMENTS

Packages for spent nuclear fuel used in Germany are dual purpose casks for interim storage and transport. The containment of these casks is closed by bolted lid systems (usually with primary and secondary lid). Each lid system (or barrier) is usually sealed by a metallic gasket to ensure long term leak tightness. A second outer elastomeric gasket is installed for testing reasons. The currently licensed designs house up to 21 PWR or 52 BWR fuel assemblies with a maximum average burn-up of 65 GWd/tU. The assessment work of BAM within the design approval process of such packages brings up two issues which involves the behavior of the fuel assemblies. In the containment analysis the activity release compliance with the regulatory limits are verified. The second issue refers to the criticality safety analysis. The potential cladding failure followed by the release of gas, volatiles, fine fuel particles or

fragments into the cask cavity is highly relevant for both safety functions. BAM assesses the physical and geometrical state of the spent fuel under transport conditions, e.g. the deformations or failure modes of the fuel rods, which can influence the correspondent analysis.

The mechanical and thermal conditions governing the assessment are given by the transport regulations [1] of the International Atomic Energy Agency (IAEA). The regulations specify requirements concerning containment, shielding and criticality safety for the severity levels:

- Routine conditions of transport (RCT);
- Normal conditions of transport (NCT);
- Accident conditions of transport (ACT).

RCT includes mechanical loads resulting from accelerations, decelerations and vibrations of the incident free transport of the package. NCT relates to minor incidents and the test sequences prescribed for ACT cover mechanical and thermal loadings of hypothetical accidents. The main mechanical loadings for NCT and ACT are defined by drop tests. With regard to the following discussion, the 0.3 m free drop onto an unyielding target (for packages with a mass > 15 000 kg) is representative for NCT. The 9 m drop onto an unyielding target and the 1 m drop onto a puncture bar have a dominant role for ACT in this context. The drop test results in acceleration/deceleration forces which are passed from the impact limiter via the cask body and the basket, to the fuel assemblies and finally to fuel rods. The loads on the fuel rods depend on complex interactions of all these components. Although there are data about the loads on different package components for transport conditions there is almost no knowledge about the loads which are transmitted to the fuel cladding. This impedes the understanding the real response of the fuel rods substantially.

The assessment of the mechanical response depends on the design of the fuel assemblies but also on their operational and storage history. This results in a wide range of material properties which are decisive for the mechanical behavior of the fuel rods such as the cladding alloys (Zircaloy-2, Zircaloy-4, ZIRLO, M5, etc.), the burn-up, the thickness of the oxide layer on the cladding, possible hydride reorientations, etc. The deterioration of the mechanical properties of the cladding materials due to burn-up makes it nearly impossible to calculate the mechanical problem precisely. In conclusion the assessment of the fuel assemblies and rods is governed by knowledge gaps, both for the definition of the load as well as for determination of covering material properties. Therefore BAM applies and accepts enveloping approaches for the containment assessment and the assumptions for the criticality safety analysis (see Ref. [4]).

The approval of special solutions for the encapsulation of defective fuel rods is getting important in Germany at the moment. The reason for that is the goal of the nuclear power plant operators to get a spent fuel free situation in their facilities as fast as possible.

3. CONTAINMENT ASSESSMENT

The IAEA transport regulations [1] define activity release criteria which have to be met by the transport package:

a) For NCT 10^{-6} A_2 per hour;
b) for ACT 10 A_2 per week for Krypton-85 and A_2 for all other radionuclides.

There are no specific limits for RCT, but pre-damage or fatigue which may occur during the incident free transport have to be considered in NCT assessment. The direct measurement or assessment of radioactive release for spent fuel packages is not feasible. Instead, a common method is to relate the admissible limits of activity release to equivalent standardized leakage rates. The radioactive content (in form of gas, liquid, solid or a combination of these) can be released through leaks in the containment system or in case of elastomeric gaskets additionally by permeation. The claddings of the fuel rods serve as the first barrier of the containment system. The second barrier is the lid system of the cask usually equipped with metallic gaskets. The design leakage rates of the cask containment are in most cases small enough to prevent any particle release to the environment. But gases and volatiles could get through a potential capillary leak in the gasket. So assumptions concerning the radioactive material in the cavity (in particular the gaseous and volatile fractions) are required for the evaluation of the radioactive release out of the package. Fission gas and volatiles can escape through breaches in the claddings. This release has to be considered for the estimation of the radioactive material in the cavity. A common method is the use of failure rates f_b for the fuel rods, which defines the amount of damaged claddings. In the absence of detailed knowledge about the mechanical response of the fuel rods during NCT and ACT the following failure rates assumptions are accepted by BAM:

a) For NCT and burn-ups below 55 GWd/tU: $f_b = 3\%$;
b) for NCT and burn-ups above 55 GWd/tU: $f_b = 100\%$;
c) for ACT: $f_b = 100\%$.

The accepted failure rates a) and c) are recommended in NUREG/CR 6487 report [3] under consideration of experiences from reactor operations (burn-up <38 GWd/tU) and in the case a) by including a safety factor of 150. For the high SNF burn-up BAM questions the transferability of the value of $f_b = 3\%$. In the absence of another reliable criterion for case b) BAM requires that all fuel rods have to be assumed as damaged for the containment analysis. For very high burn-ups, e.g. 120 GWd/tU, BAM currently requires the encapsulation of the fuel rods. In addition to the failure rates, it is required to make assumptions about the fractions of gases and volatiles which are released by a breached cladding. Currently the analysis assumes that 15% of the fission gas and 0.02% of the volatiles in the fuel rod get out. On the basis of the source term and the assumptions explained above the activity release calculation can be performed with respect to the standard leakage rates of the gaskets used [4].

4. ASSUMPTIONS FOR THE CRITICALITY SAFETY ANALYSIS

The geometrical and material assumptions for the fuel assemblies are used as input data for the shielding and criticality safety analyses. The IAEA transport regulations [1] require subcriticality of the radioactive content under RCT, NCT and ACT. With respect to the criticality safety analysis, ACT are the most severe condition. Besides the highest mechanical load, the assumption of water in the containment has a significant impact on the criticality safety. Mechanical loads under ACT include the 9 m drop test onto an unyielding target in sequence with the 1 m puncture drop test as the dominant load cases for the assessment. Depending on the orientation of the cask during the drop test, two exemplary mechanisms can occur, which affect the criticality of the radioactive content particular severe:

a) The increase of the lattice spacing of the fuel rods in the assembly causes an increased moderation ratio;
b) The breakage of the fuel rods and the release of pellet fragments can result in a cumulated mass of the fissile material in the cavity.

A possible scenario for effect a) is the buckling of the fuel rods by the load of the 9 m drop on the lid or bottom of the package. The induced inertia forces typically outrange the buckling force of the fuel rods. The deformed state of the dynamic buckling process cannot be predicted or calculated. Instead, the covering deformation state is assumed for the criticality safety analysis considering geometrical (e.g. by the head of the assembly, failure of spacers, limits of the basket region) and load conditions (e.g. load concentration in the bottom end of the fuel assembly).

The most severe accident for effect b) is the 9 m drop with the horizontally orientated package. The inertia forces of the impact induce a bending of the fuel rods which may lead to their breakage. Fissile material can be released into the cavity at the site of fracture. For the estimation of the released fissile material, it is necessary to make assumptions for the released material per fracture point and for the total number of breakage points in the fuel assembly. The total number of breakage points has to be estimated by an enveloping mechanical model. The fuel rods are assumed as continuous beams supported by spacer grids in the current approaches accepted by BAM. The bending stress in the rods under the inertia forces from the drop test has to be assessed with respect to the critical stress in the cladding which leads to failure. A possible failure of the single or multiple spacer grids has to be taken into account in different mechanical configurations. Failure of the cladding has to be assumed, along with a release of fissile material, for fuel rod locations with a higher bending stress than the critical one. The responses of fuel rod segments to transverse loadings were used to measure the amount of fuel released per fracture point in a few tests available for public use at the moment. The results range from 2 g to 10 g per fracture point [2]. Thus the total number of breakage points can be determined. In conjunction with the estimated released mass per fracture point the total mass of fissile material in the cavity can be obtained, which gives the input for the criticality safety analysis.

5. ENCAPSULATION OF DEFECTIVE FUEL RODS

Defects at fuel rods can occur during operation in the nuclear power plant. Defective fuel rods are extracted from the fuel assembly and either repaired or stored separately in the storage pool. Encapsulations can be used to ensure the transportability of these fuel rods. Such kinds of solutions are getting more in focus due to the nuclear phase-out decision in Germany. An encapsulation is used to transport defective fuel rods inside established and licensed spent fuel packages. In this case, it serves as the first barrier instead of the cladding and has to be considered in the analysis of containment and criticality. Furthermore, the licensed package design must not be affected negatively by the encapsulation. This means, it should have a similar behavior as fuel assemblies. In general, this is challenging due to the design of such encapsulations which implies e.g. a different sealing system and a higher stiffness. Nevertherless, the mechnical characteristics of these components are well-known contrary to the physical state of fuel assemblies which depend on their operational history. The behavior of the encapsulations under transport conditions can be calculated or predicted more precise and the design can be optimized. The requirements for their manufacturing can be compared with the ones for spent fuel packages. Since the closure system of the encapsulations are usually permanent (e.g. by welds), the requirements for drying, sealing and tightness testing after loading are part of the approval process.

As mentioned before, the mechanical behavior of such encapsulations under transport conditions differs from typical fuel assemblies. Primarily the higher stiffness leads to special requirements concerning the interactions between encapsulation and the cask, in particular under ACT. For instance, the 9 m drop onto the lid of the package implies high loads on the

lid system applied by the internal impact of the content. To avoid increased loads on the lid system additional damping structures are established to absorb the increased impact loads of the encapsulations. The design of such damping structures is challenging. Experimental tests are usually necessary.

6. RESEARCH AND DEVELOPMENT

The lack of knowledge about the mechanical behavior of fuel assemblies both for the containment analysis and the criticality safety assessment requires R&D activities.

The current research works of Sandia National Lab in Albuquerque, USA, can be mentioned as an example of the investigation concerning the impact of shock and vibration during the transport on the fuel assembly. A PWR assembly with surrogate fuel rods was subjected to vibration and shock loads which were applied by a shaker table. Three Zircaloy-4 fuel rods were instrumented with accelerometers and strain gauges. The results were discussed with respect to potential fatigue damage of fuel-rod cladding during RCT [5].

The research cooperation of BAM with the Institute for Transuranium Elements (ITU) in Karlsruhe, Germany, is focused on the mechanical testing fuel rod segments in the hot cell. The goal is a better understanding of the failure mechanism with respect to the gas release for loads in range of NCT. The ITU will perform 3-point bending tests on pressurized spent fuel rod segments. BAM provides boundary conditions based on the experiences of the approval work and will also contribute by numerical analysis of the test set-up and associate the results with the load conditions in the cask. An extension of this work to the specific load cases for ACT is planned.

7. CONCLUSIONS

Within the licensing procedure of spent nuclear fuel packages, BAM assesses the geometrical and material assumptions for the fuel assemblies which are applied as input data for the containment analysis and the criticality safety analysis. The potential cladding failure followed by the release of gas, volatiles, fine fuel particles or fragments into the cask cavity is highly relevant for both safety functions. The assessment is challenging due to knowledge gaps regarding the loads on the assemblies and their material properties. The enveloping approaches accepted by BAM were discussed in this paper. Defective fuel rods result from operation of the nuclear power plant. Defective fuel rods are extracted from the fuel assembly and either repaired or stored separately in the storage pool. Encapsulation can be used to ensure the transportability of these fuel rods. Such kinds of solutions are getting more in focus due to the nuclear phase-out decision in Germany. Encapsulations are used to transport defective fuel rods inside established and licensed spent fuel packages. Although the protection goals are not changed, additional aspects have to be considered in the assessment, which are addressed in this paper. The limited knowledge about the material properties of the fuel assemblies and their loading under transport conditions requires further investigation. In this context, two exemplary experimental programs were described.

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EVALUATION OF SEALING PERFORMANCE OF METAL GASKET USED IN DUAL PURPOSE METAL CASK SUBJECTED TO AN AIRCRAFT ENGINE MISSILE

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Abstract

In order to investigate of the integrity of the lid structure of the metal cask during the extreme impact loads due to aircraft crash, two impact scenarios for aircraft engine crash onto the metal cask without impact limiters are considered for both, a vertical impact onto the lid structure and a horizontal impact hitting the cask. The horizontal impact test using scale model engine of aircraft and the vertical impact onto the head of the full-scale metal cask were executed. From these experimental results, it seems that the loss of the inner pressure of the cask cavity may be avoided in the impact event with the horizontal and vertical orientation even if the severe impact load was applied on to the metal cask due to aircraft engine crush. Moreover, in order to evaluate the deformation response of the metal gasket subjected to the accidental loads measured during experiments, the impact analysis by LS-DYNA code was executed.

This paper has been removed from the proceedings as it has been published elsewhere. Interested readers should refer to the following reference:

[1] CENTRAL RESEARCH INSTITUTE OF ELECTRIC POWER INDUSTRY, “Basis of Spent Nuclear Fuel Storage”, ERC Publishing Co. Ltd., Tokyo (2015) 145–155.

FABRICATION THERMAL TEST: METHODOLOGY FOR A SAFE CASK THERMAL PERFORMANCE

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Abstract

One of the safety-related functions in a Dual Purpose Cask (DPC) is the thermal performance. Regulations only address a thermal confirmatory test to assure the cask thermal performance under accident conditions (fire test). As a DPC is to be used for both storage and transport, the thermal passive behavior of the cask under normal condition of storage/transport shall be demonstrated. Although this thermal test is not referred either in storage or transport regulations, some of the regulators require to perform this test as a confirmatory test for the normal condition. In most of the cases, the result of this test will provide with an acceptable resolution or not of the cask design licensing application. The fabrication thermal test (described here only for storage) intends to confirm the validation of the analytical model and to assure the correct thermal performance of the cask. The thermal test is divided in two phases: a) Thermal Test Finite Element Model (FEM) Simulation and b) Thermal Test.; Thermal Test FEM Simulation (phase 1) uses almost the same model used for the licensing thermal evaluation with minor changes, such us cask lid end, cask bottom end boundary condition and fuel assembly thermal axial profile. The FE model considers the same heat load per fuel assembly, and therefore the same total cask heat load as used in the licensing calculations. Modifications are on account of the difficulties to represent the normal condition of storage in the shop. Thermal Test (phase 2) uses the first cask manufactured. The cask is in final condition and the only part of the cask different from the final storage cask configuration is the cask lid system. For this purpose, a special lid is used for the test to be able to take the instrumentation wires out of the cask cavity, while keeping the same thermal behavior of the final storage cask lid configuration. The fuel assembly heat load is simulated by using electrical heaters, so specific features, such us fuel assembly thermal axial profile is quite difficult to simulate, as the electrical heaters provide a uniform heat along the length. For this and other physical test difficulties, results shall be carefully evaluated and discussed.

1. INTRODUCTION

Thermal heat dissipation is one of the safety related functions of a Spent Nuclear Dry Storage and Transportation Cask. The thermal behaviour of either storage or transport application shall be assured for both Normal and Hypothetical Accident Conditions. The thermal performance on dry storage and transportation casks are usually based on thermal passive behaviour and the decay heat from the fuel assemblies inside the cask is rejected to the outside of the cask according to the standard heat transfer mechanisms, material conduction, radiation and convection.

Traditionally, the thermal performance of a cask rely only in analytical models, using software previously accepted and validated with simple cases. Confirmatory tests are required in transport regulations only for accident conditions, and although tests are required, they are not absolutely mandatory as there are few other alternatives within the regualtions to demonstrate the thermal behaviour of a cask under accident conditions (see paragraph 701 a), b), c), and d) of IAEA SSR-6 [1]), without carriyng out thermal test (fire tests). It is well understood and for this reason accepted, due to the difficulties of these tests, that analytical evaluation is a good tool to demonstrate and justify the proper thermal behaviour of a cask under accident condition.

With regards to dry storage, it is accepted in many countries to use the same transport fire accident conditions for conservatism, unless the designer could justify other fire conditions for storage (i.e. site-specific).

In previous paragraphs only accident conditions have been mentioned, while the thermal behaviour and the possibility to validate the thermal model for normal condition has been ommited. The reason is that no confirmatory tests for normal condition of storage or transport

are required in the regulations. This is not enough reason for many regulators, which requires, as part of the licensing approval cycle, to perform a fabrication thermal test on the first unit manufactured of each new cask design, simulating the normal condition of the spent fuel cask.

Two aims are basically what it is intented to obtained with the fabrication thermal test: a) analytical model validation, and b) to assure the proper thermal performance of the new cask design.

The following sections will explain in detail the analytical model methodology, the fabrication test, results and discussion.

2. THERMAL TEST ANALYTICAL MODEL

A finite element model (FEM) of the spent fuel cask is used to simulate the fabrication thermal test. This analysis intends to simulate the real behaviour of the loaded (simulated) cask during the fabrication thermal test, in order to validate the assumptions, engineering approaches, boundary conditions, material properties that are considered within the licensing thermal calculations.

2.1. Finite element model description

The finite element model used for this validation analysis is the same as the one used for the licensing thermal evaluation. Only a few minor differences are being considered to adjust the model to the cask thermal test configuration. The differences are listed below:

- The cask inner and outer lids are replaced by a single lid. The single lid is specifically designed for this purpose to allow the instrumentation and power wires coming out from the cask cavity and basket. The thermal test lid design is made to assure that the thermal performance remains the same as from the cask configuration under normal condition, with negligible temperature variations;
- Trunnions are not modeled as they have no impact on the results. This assumption is even conservative, as trunnions should act as heat rejection fins;
- The fuel assembly equivalent material is replaced by the electrical heaters equivalent material.

The cask materials properties are taken from the ASME Code [2]. Those material properties not directly provided in the ASME Code, are derived from the equation in the ASME Section II, Part D, Table TCD [2].

As mentioned before, to simulate the fuel assembly decay heat, electrical heaters are used. These are located in every position of the basket (in this particulary case, 52 positions for the ENUN 52B spent fuel cask). Modeling the equivalent material of the electrical heaters is one of the keypoints of the analysis. For this purpose a specific methodology based on Refs [3, 4], using a detailed 2D finite element model of the electrical heaters (see Fig. 1), varying the decay heat and the boundary temperature conditions.

By using this methodology, the electrical heater equivalent material properties are obtained. These properties are:

- Radial and axial thermal conductivity;
- Density;
- Specific heat.

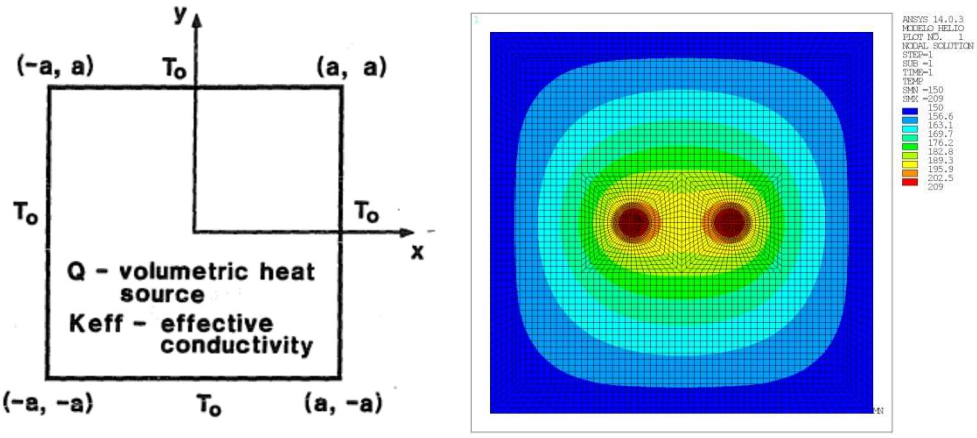

FIG. 1. Electrical heater 2D finite element model.

As it is well known, the fuel assembly axial thermal profile provides a particular shape along its active length. It is a standard practice to adjust the decay heat per fuel assembly along the length, using the proper peak factors after normalizing the axial profile in, for example, 24 nodes. This decay heat distribution is apply to the fuel assembly FE model, representing the burnup or thermal axial profile.

Unfortunately, this real shape is quite difficult to simulate in a test, as the electrical heater generates the heat uniformly along the length. The following figure (Fig. 2) shows the comparison between the real axial shape and the uniform shape provided by the electrical heater.

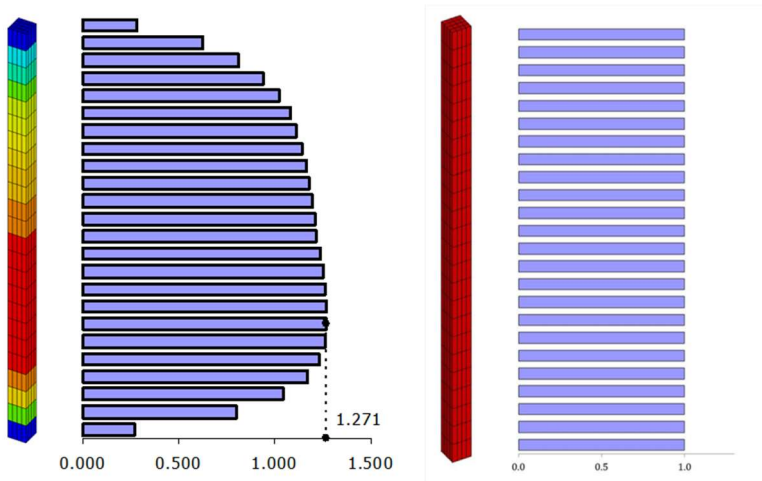

FIG. 2. Fuel assembly and electrical heater thermal axial profile.

This axial profile effect could have a minor impact on the cask top and bottom temperatures, especially those temperatures inside the basket, when comparing the results of this analysis against the licensing analysis results. As the purpose of the fabrication test is to validate the model, this minor discrepances have no effect on the licensing thermal evaluation.

Boundary conditions are also an important factor in this evalutions. Ambient temperature may vary along the fabrication test, so this could affect the time for reaching the thermal equilibrium, and even the final results. For this reason a sensitivity analysis with different ambient temperatures is performed. This will allow to compare the FE model results with the ambient temperature during the test. The licensing maximum total heat load per cask is simulated with the electrical heaters. The heat transfer mechanisim inside the cask are considered to be conduction and radiation (no credit to internal convection), and convection and radiation outside the cask. View factors for external radiation are also adjusted to simulate the working stand fabrication test conditions. Solar radiation is not considered in the model as the fabrication thermal test is to be performed inside a building.

Fig. 3 shows the FE model developed for this analysis.

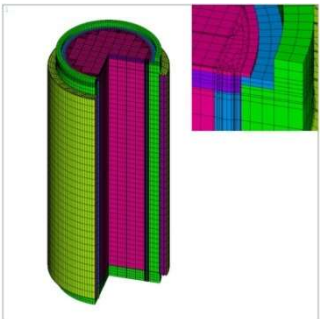

FIG. 3. Fabrication thermal test 3D FE model.

As all the results will be compared against the licensing results (quasi-static analysis), which consider a long period of storage (50 years), it is necessary to establish a criterion for determining the thermal equilibrium in the test. For this purpose, a transient analysis is also performed in order the obtain the duration of the test until the thermal equilibrium criterion is reached on every thermocouple. Based on ENSA's experience in several thermal tests performed along the last 15 years, the thermal equilibrium criterion[1] selected is:

$$T_{thermocoup} \leq {}^{1°C}/_{7h} \quad (1)$$

[1] This criterion considers the effect of the ambient temperature variation during the test.

Fig. 4 shows the thermal equilibrium in one basket loaction.

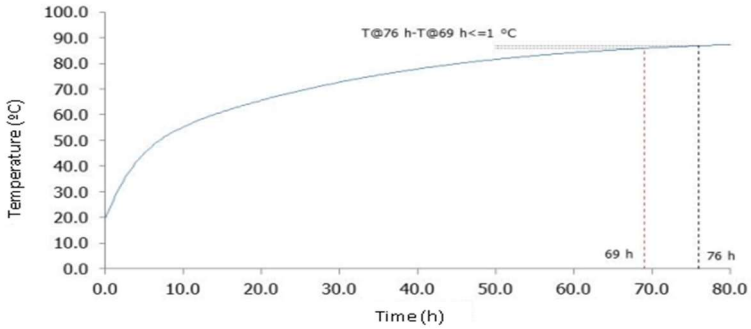

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FIG. 4. Thermal equilibrium criterion for the fabrication thermal test.

2.2. Thermal test analytical model results

The results of the quasi-static thermal test analytical evaluation provide the temperatures along the cask, selecting for the discussion those points where the instrumentation (thermocouples) are located.

Table 1 lists the temperatures in some locations at different ambient temperatures.

TABLE 1. TEMPERATURES OBTAINED FROM THE ANALYTICAL THERMAL TEST MODEL

Thermocouple location	Ambient temperature (°C)				Normal condition
	21	24	27	40.1	
Mid-plane inner (EHOS)[a]	143.6	146.0	148.4	158.9	169.9
Bottom inner cask body	58.5	61.2	63.8	75.4	78.9
Mid-plane outer cask surface	54.4	57.1	59.7	71.2	77.4
Bottom outer cask surface	49.1	51.7	54.4	66.0	71.6

[a] EHOS = Electrical Heater Outer Surface

These temperatures are required to be higher than the temperatures obtained in the thermal test. This is due to the conservative assumptions taken in the analysis. Additionally, another case was run at the maximum storage temperature, to compare the thermal test results at the maximum operation temperature with the licensing (normal condition) model results.

Fig. 5 below shows the results of the thermal test analytical model at one of the ambient temperature selected.

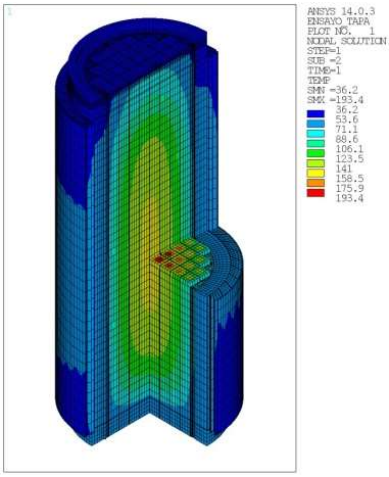

FIG. 5 Analytical thermal test results at selected ambient temperature.

3. FABRICATION THERMAL TEST

The fabrication thermal test will confirm the appropiate thermal performance of a new cask design. Theoretical analysis considers material properties from design codes and other bibliography, modelling assumption for conservatism, simplifications in the model, and many other things that may or may not have impact in the final thermal performance of the cask under operation. For this reason, it is convenient to validate those assumptions, material properties and simplifications by comparison between the analytical and test results. Having the change to do so, it is a good practice, not just for licensing purposes, but for giving the regulators, end users and others enough confident on the new design.

3.1. Cask preparation for fabrication thermal test

The fabrication thermal test should be performed on the first unit manufactured of every new cask design. The cask to be used for the test shall represent the final cask configuration under normal condition (in this case, storage). For this reason, the fabrication thermal test is one of the latest activities of the cask fabrication program.

In the case of the ENUN 52B cask (Dual Purpose Metal cask with up to 52 BWR fuel assemblies), the cask used for the test is almost ready for delivery, this means that it has been gone through all other final tests, such us the hydrostatic pressure test, trunnion loading tests, and with the exception of the helium leak test that is to be performed inmediately after the thermal test and prior delivery to end user. The cask outer surfaces are also in final condition (painted). All QA inspections confirm the compliance with all fabrication requirements (ASME Code and others) of the cask.

Fig. 6 below shows the ENUN 52B design and the first unit manufactured ready for the fabrication thermal test.

FIG. 6. ENUN 52B Cask for fabrication thermal test.

Fifty-two (52) electrical heaters simulate the maximum design thermal decay heat coming from the 52 BWR fuel assemblies. The electrical heaters, as indicated in previous sections, provide a uniform heat all along the length (no fuel assembly thermal axial shape can be represented without too many additional difficulties). The 52 electrical heaters are interconnected in four (4) groups, where each group is connected to an independent power station (four units). Fig. 7 shows the electrical heaters configuration and group connections.

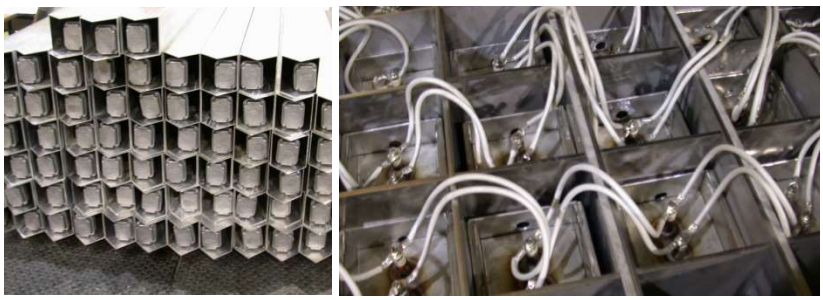

FIG.7. Electrical heaters and group connections.

The temperature in several key locations of the cask is recorded by using calibrated type 'K' thermocouples. These are to be located in different parts of the cask, such us the basket, outer surface of electrical heaters, cask body inner surface and cask external surface. Every control location has two thermocouples (one official and another as a backup in case the official one fail). Thermocouples are also located in three sections along the length of the cask, bottom, mip-plane and top. Fig. 8 shows thermocouples location in the cask.

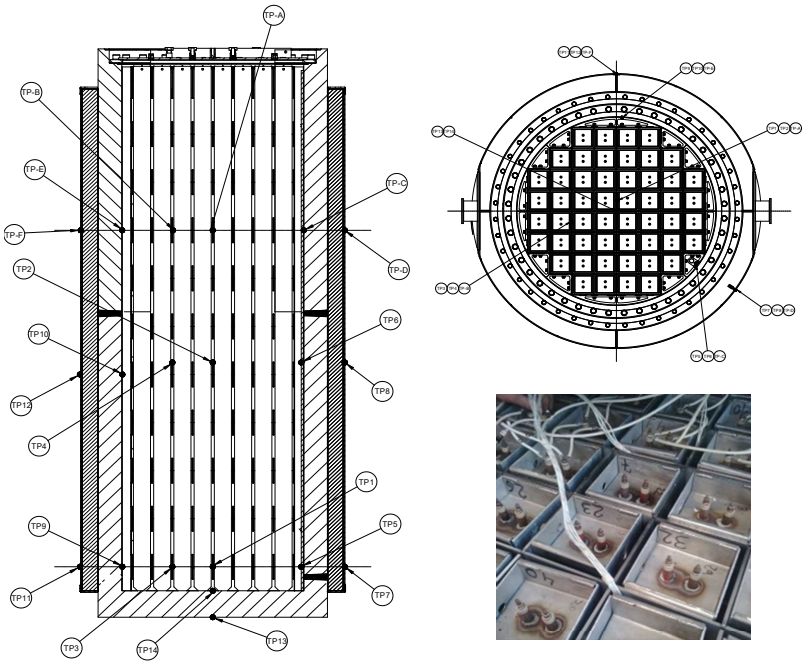

FIG. 8. Thermocouple locations for the fabrication thermal test.

One of the biggest and critical issues in a thermal test is the design of a cask test lid which will have to allow the instrumentation and power wires coming out from the inner cask cavity through several penetrations. This lid and its penetrations shall also assure enough leak tightness to guarantee a good vacuum level for subsequent inert gas filling inside the cavity. The inert gas inside the cask cavity is slightly pressurized above 1 atmosphere (abs.). Lid tightness design shall maintain the same level of pressure along the test. The following figure (Fig. 9) shows the lid design and the penetrations with the sealing device system.

FIG.9. Cask lid for fabrication thermal test.

4. RESULTS, DISCUSSION AND CONCLUSION

The thermal test temperatures reached the thermal equilibrium criterion when expected (around 80 hours). Minor leakage occurred during the thermal test so inert gas filling to maintain the pressure was required. Temperatures on every thermocouple location were continuously registered. A recorder with a special software was used to process the data.

The results are listed in Table 2, comparing temperatures between the fabrication thermal test and thermal test analytical model.

TABLE 2. THERMAL TEST AND ANALYTICAL MODEL TEMPERATURES RESULTS

Results (temperature in ºC)			
Thermocouple[a]	Fabrication thermal test	Analytical model	ΔT
TP1	76.8	93.5	16.7
TP2	114.3	146	31.7
TP3	69.9	87.1	17.2
TP4	104.3	128.8	24.5
TP5	52.9	61.2	8.3
TP6	60.4	68.3	7.9
TP7	46.5	52.6	6.1
TP8	51	57.1	6.1
TP9	56.2	64.9	8.7
TP10	68.1	76.2	8.1
TP11	45.8	51.7	5.9
TP12	51	55.2	4.2
TP13	59.5	66.4	6.9
TP14	61.9	68	6.1

[a] Thermocouple number and location per Fig. 8.

It can be seen from the Table 2 above, that the calculated temperatures in all locations are always higher than the temperatures obtained directly from the fabrication thermal test. This means that the assumptions taken in the analytical model are quite conservative and over estimate the test temperatures. It can be also noticed that temperatures in those thermocouples attached to the outer surface of the electrical heaters (TP1 to TP4) give the higher differences, and this is due to the equivalent material calculated for the electrical heaters, which indicates that the methodology for the equivalent thermal conductivity is quite conservative. The rest of the thermocouples have a difference less than 10ºC.

Fig. 10 graphically shows the above mentioned, showing the temperature trend up to the thermal equilibrium criterion is reached of several thermocouples, providing the comparison between the temperatures of the analysis and the thermal test. The Fig. shows that temperatures obtained in the analysis are always higher that those obtained in the fabrication thermal test.

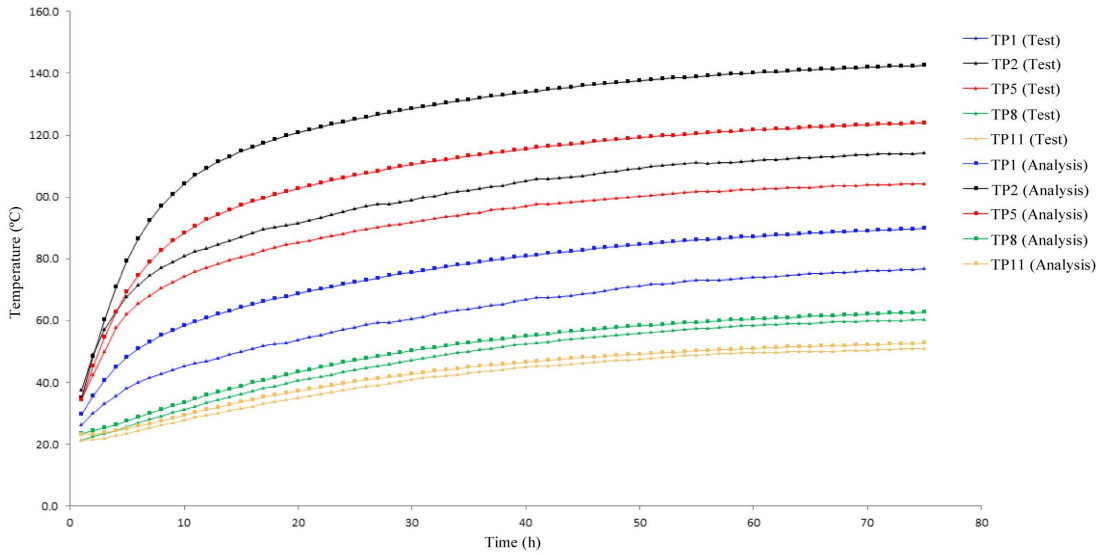

FIG. 10. Test vs. analytical temperature results.

In summary, it can be concluded that the methodology presented in this paper assure the safe ENUN 52B cask thermal performance, demonstrating that the analytical approaches are conservative and providing enough confident to rely on analytical evaluations. The two aims described in previous sections were met, as the thermal test and analytical results have an excellent correlation, so validation of the model was confirmed. In the same manner, the fabrication thermal test results also confirm the correct thermal performance of the new ENSA Dual Purpose Metal Cask design, ENUN 52B.

ACKNOWLEDGEMENTS

I would like to thank V. Gomez and A. Palacio from the Ensa's Design & Analysis Spent Fuel Cask team, L. Bilbao and L. Lopez from Ensa's Project Department, as well as the Heat Treatment and Assembly Shop Divisions, for their support and encouragement during the last few months. Special thanks to my wife, my children and my parents who have always encouraged me in the good and bad times.

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AGEING MANAGEMENT PROGRAMMES

(Session 4)

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INVITED SPEECH 05: MANAGING AGING EFFECTS ON DRY CASK STORAGE SYSTEMS FOR EXTENDED LONG-TERM STORAGE AND TRANSPORTATION OF USED FUEL

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Abstract

The report "Managing Aging Effects on Dry Cask Storage Systems for Extended Long-Term Storage and Transportation of Used Fuel, Rev. 2" is a key reference in the U.S. Nuclear Energy Institute's (NEI's) "Industry Guidance for Operations-Based Aging Management". The guidance in NEI 14-03, Rev. 0 was submitted for endorsement by the U.S. Nuclear Regulatory Commission (NRC) in September 2014, while the NRC is updating its own guidance document in the "Standard Review Plan for Renewal of Used Fuel Dry Cask Storage System License and Certificate of Compliance", NUREG-1927. The main features of the aging management report are highlighted, along with a brief discussion of guidance documents from industry and NRC, as well as the aging management needs for a Pilot Interim Storage Facility and beyond. The latter is of interest as the used fuel may need to be stored and transported multiple times before final disposal at a mined repository or geological disposal facility.

Key Words: Aging management, spent fuel dry storage casks.

1. INTRODUCTION

At present, there is no designated disposal site for used nuclear fuel in the United States, which raises the prospect of extended long-term storage (i.e., >60 years) and deferred transportation of used fuel at operating and decommissioned nuclear power plant sites. Under U.S. federal regulations contained in Title 10 of the Code of Federal Regulations 72.42,

> *"the initial license term for an independent spent fuel storage installation (ISFSI) must not exceed 40 years from the date of issuance. Licenses may be renewed by the U.S. Nuclear Regulatory Commission (NRC) at the expiration of the license term upon application by the licensee, for a period not to exceed 40 years. Applications for ISFSI license renewals must include (1) time-limited aging analyses (TLAAs) that demonstrate that structures, systems, and components (SSCs) important to safety will continue to perform their intended function for the requested period of extended operation and (2) a description of the aging management program (AMP) for management of issues associated with aging that could adversely affect SSCs important to safety. In addition, the application must include design basis information as documented in the most recently updated final safety analysis report, as required by 10 CFR 72.70. Information contained in previous applications, statements, or reports filed with the Commission under the license may be incorporated by reference, provided that these references are clear and specific"* [1].

The NRC has issued the "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance", NUREG-1927,

> *"under which NRC may renew a specific license or a Certificate of Compliance (CoC) for a term not to exceed 40 years. Both the license and the CoC renewal applications must contain revised technical requirements and operating conditions (fuel storage, surveillance and maintenance, and other*

requirements) for the ISFSI and dry cask storage system (DCSS) that address aging effects that could affect the safe storage of the used fuel. The information contained in the license and CoC renewal applications will require NRC review to verify that the aging effects on the SSCs in DCSSs/ISFSIs are adequately managed for the requested period of extended operation" [2].

Over 2000 dry casks in the United States were loaded with used fuel; four ISFSIs (Surry, H.B. Robinson, Oconee, and Calvert Cliffs) have received their renewed licenses for 40 years, and an application for license renewal for 40 years for another, Prairie Island, is pending approval by NRC. In addition, renewal of the CoCs for the VSC-24 cask and Standardized NUHOMS are pending NRC approval. The NRC expects to receive fifteen (15) renewal applications between 2015 and 2025.

2. THE ANL/DOE AGING MANAGEMENT REPORT

2.1. Description of the aging management report

As part of the U.S. Department of Energy's (DOE's) Used Fuel Disposition Campaign Research and Development program, Argonne National Laboratory (Argonne) has prepared a report titled "Managing Aging Effects on Dry Cask Storage Systems for Extended Long-Term Storage and Transportation of Used Fuel" Rev. 2, FCRD-UFD-2013-000294, ANL-13/15 [3]. This report, herein referred to as the aging management report, examines issues related to managing the aging effects on the SSCs in DCSSs and ISFSIs for extended long-term storage and subsequent transportation of used fuels, following an approach similar to that of the Generic Aging Lessons Learned (GALL) report, NUREG-1801 [4], for the aging management and license renewal of nuclear power plants. The aging management report contains five chapters and two appendices, A and B, respectively, on Quality assurance for AMPs for used-fuel dry cask storage systems and a comparison of TLAAs and AMPs in the aging management report with those in NUREG-1801, Chapters X and XI.

Among the five chapters in the report, Chapter I provides an overview of the ISFSI license renewal process based on 10 CFR 72 and the guidance provided in NUREG-1927 [2]. Chapter II contains definitions and terms for structures and components in DCSSs, materials, environments, aging effects, and aging mechanisms. Chapter III and Chapter IV contain the TLAAs and AMPs, respectively, that have been developed for managing aging effects on the SSCs important to safety in the DCSS designs described in Chapter V, which contains summary descriptions and tabulations of evaluations of AMPs and TLAAs for the SSCs that are important to safety in the DCSS designs (i.e., NUHOMS, HI-STORM/HI-STAR 100, TN metal cask, NAC International S/T storage cask, Ventilated Storage Cask [VSC-24], Westinghouse MC-10 metal dry storage cask, CASTOR V/21 and X/33 dry storage casks, and W150 FuelSolutions storage system) that have been and continue to be used by utilities across the United States for the dry storage of used fuel.

"The goal of the aging management report is to help establish the technical basis for extended long-term storage and subsequent transportation of used fuel, which may occur multiple times before final disposal at a mined repository or geological disposal facility. It has been suggested that future efforts should include (1) development of additional TLAAs and AMPs that may be deemed necessary and (2) further evaluation of the adequacy of the generic TLAAs and AMPs that may need augmentation. Industry and site-specific operating experience from the various DCSSs/ISFSIs located across

the United States should be periodically examined to (a) ascertain the potential aging effects on the SSCs in the DCSSs, thereby enabling a compilation of existing aging management activities, and (b) assess adequacy of these activities for extended long-term storage and subsequent transportation of used fuel" [3].

2.2. Resolution of stakeholder comments

An earlier version of the aging management report, Rev. 1, was released by DOE in November 2013 for stakeholder comments worldwide, via the United States Electric Power Research Institute's (EPRI's) Extended Storage Collaboration Program (ESCP), with the comment period ending 31 March 2014. Over 200 written comments were received from stakeholders representing industry, utilities, NRC, U.S. national laboratories, and Germany. The various comments focused on:

- Interpretations of 10 CFR 72 and related NRC regulatory requirements;
- Definitions of environments, aging mechanisms, and aging effects;
- Specific content in recommended AMPs and TLAAs with respect to aging management activities and analyses;
- Difficulties in performing inspections, condition monitoring, and other aging management activities in terms of accessibility and hostile environments;
- Descriptions and evaluations of the various dry cask storage systems.

At the request of industry and NRC, separate meetings were held at Argonne in May 2014 to discuss the comments and their proposed resolutions. These meetings were highly productive in that the technical issues in aging management were discussed in depth, as were the approaches reflecting the unique perspectives of the NRC and industry. Over 100 additional comments were also received in June from NRC, as well as from dry cask vendors — these comments are particularly important for verifying the accuracy of the descriptions of the DCSS designs in Chapter V of the aging management report. Highlights of the significant changes resulting from the stakeholder comments can be found in the Executive Summary of the report [3].

3. THE U.S. INDUSTRY AND NRC GUIDANCE ON AGING MANAGEMENT

The industry has incorporated the aging management report into the Nuclear Energy Institute's (NEI's) "14-03: Guidance for Operations-based Aging Management for Dry Cask Storage" [5] to augment the 10 CFR 72 license and Certificate of Compliance (CoC) renewal review guidance in NUREG-1927, Section 3.0, "Aging Management Review". Revision 0 of NEI 14-03 was submitted for the NRC's endorsement in September 2014. Note that NEI 14-03 introduced 'tollgates' into the Part 72 license and CoC renewal process.

> "*'Tollgate' is a new term created by the industry to address the fact that the applicability of potential dry cask storage aging mechanisms may not be able to be verified at the time license and CoC renewal applications are submitted. This information would enhance the current understanding of the future state of dry spent fuel and the canisters that contain it. Tollgates are part of a learning, operations-based aging management program implemented by licensees via requirements in the renewed license or CoC and associated final safety analysis report. These requirements obligate the licensees to perform periodic assessments of the aggregate state of knowledge of aging-related operational experience, research, monitoring, and inspections to ascertain the ability of in-*

scope DCSS design SSCs to continue performing their intended safety functions throughout the requested period of extended operation" [5].

In March 2015, the NRC released a draft of NUREG-1927, Rev. 1 [6], which was developed in parallel with the NEI effort to develop guidance in NEI 14-03, Rev. 0 for the industry in the preparation of applications for renewal of licenses and CoC. The draft NUREG-1927, Rev. 1 includes guidance on the continued evaluation of operating experience (see Section 3.6.1.10 of NUREG-1927, Rev. 1 [6]) and mentioned the use of 'tollgates' in NEI 14-03, Rev. 0 as a structured approach for assessing operating experience and data from applicable research and industry initiatives. The NRC staff has provided comments on NEI 14-03, Rev. 0 in January 2015 (ADAMS Accession No. ML15013A201). The NRC is continuing to work with NEI as it develops its guidance for proposed NRC endorsement. In the meantime and according to NRC, Section 3.6.1.10 of the NUREG-1927, Rev. 1 draft reflects the current thinking and guidance regarding information in NEI 14-03 that may be used or referenced by applicants for license or CoC renewals. As mentioned earlier, the aging management report recommends periodic assessment of industry and site-specific operating experience of the various DCSSs/ISFSIs, which share the same basic philosophy as that of the tollgates in NEI 14-03 and NUREG-1927.

4. AGING MANAGEMENT NEEDS FOR A PILOT INTERIM STORAGE FACILITY (ISF) AND BEYOND

The DOE established the Nuclear Fuels Storage and Transportation (NFST) Planning Project in October 2012. The mission of the NFST is to lay the groundwork for implementing interim storage, including associated transportation, and to develop a foundation for a new nuclear waste management organization [7]. The NFST activities are divided into four major areas: (1) consent-based siting; (2) storage; (3) transportation; and (4) strategic crosscuts. Descriptions of these activities can be found in reference [7]. The objective for the storage design activity is to lay the groundwork for implementation of interim storage as a planned part of the waste management system, with an initial focus on a pilot ISF accepting SNF in dual-purpose canisters from reactor sites that have been shut down. The primary focus is on developing and evaluating interim storage design options, including design concepts to support development of a Topical Safety Analysis Report (TSAR) for eventual submittal to the NRC. This is a key activity that can be performed now to help reduce uncertainties and mitigate schedule risks associated with full implementation in the future.

Assessing aging management needs for a pilot ISF and beyond is an ongoing activity that supports the storage design. Under this activity, regulatory, programmatic, and technical needs for aging management of a pilot ISF and beyond are being assessed by:

- Performing a comparison of the TLAAs and AMPs in the DOE/ANL aging management report with those in the recent and ongoing ISFSI license/CoC renewal applications to identify gaps;
- Updating the DOE/ANL aging management report, as appropriate, to ensure consistency of the generic AMPs and TLAAs for the DCSS designs with the operations-based AMPs in the NRC/industry guidance documents;
- Compiling 'lessons learned' in the present aging management activities for potential application to address aging management needs for the pilot ISF and beyond.

A database of operations-based AMPs and supporting technologies, including aging management R&D and domestic/international guidance documents, is also being established

by continuing collaboration with the EPRI's ESCP partners, such as the Federal Institute for Materials Research and Testing (Bundesanstalt fűr Materialforschung und -prűfung [BAM]) in Germany [8] and other countries.

5. SUMMARY AND CONCLUSIONS

An aging management report has been prepared to address issues related to managing the aging effects on SSCs in DCSSs/ISFSIs for extended long-term storage and subsequent transportation of used nuclear fuel. Managing aging effects on DCSSs/ISFSIs for extended long-term storage and transportation of used fuel requires knowledge and understanding of the various aging degradation mechanisms for the materials of the SSCs and their environmental exposure conditions during the extended period of operation.

> "*The operating experience involving the AMPs, including the past corrective actions resulting in program enhancements or additional programs, provides objective evidence to support a determination that the effects of aging will be adequately managed so that the intended safety functions of the SSCs will be maintained during the period of extended operation*." [3]

Operations-based or operations-focused AMPs and periodic assessment of their efficacy and effectiveness are a theme consistent in the aging management report and the guidance documents being developed by industry and the NRC. The fundamental principles and approach in aging management discussed herein are applicable to addressing the aging management needs for a pilot Interim Storage Facility and beyond.

> *"Managing aging effects on DCSSs for extended long-term storage and subsequent transportation of used fuel depends on AMPs for the prevention, mitigation, and early detection of aging effects on the SSCs through condition and/or performance monitoring. Detection of aging effects should occur before there is a loss of any structure's or component's intended functions that are important to safety. Among the important aspects of detection are the method or technique employed (i.e., visual, volumetric, or surface inspection), frequency, sample size, data collection, and timing of new/one-time inspections to ensure timely detection of aging effects. The challenges in the detection of aging effects include (1) detection of aging in regions that are inaccessible for inspection and monitoring and (2) determination of the frequency of inspection and monitoring (i.e., periodic versus continuous)*." [3]

DOE, the NRC, and industry personnel worldwide are actively engaged in various R&D and demonstration activities and meet regularly to exchange information through EPRI's Extended Storage Collaboration Program and other venues, such as the International Atomic Energy Agency Coordinated Research Programs and international conferences.

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AGEING MANAGEMENT SOLUTIONS TO ENSURE SAFETY OF EXTENDED DRY FUEL STORAGE

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Abstract

Storing used fuel at reactor sites for long period of time wasn't originally planned. But given the delays in the deployment of long term solutions such as geological repositories, license renewal beyond the initial license duration of 20 years is now necessary in some countries. Considering national revised plans to development of geological repository for UNF and HLW, we can predict that UNF will need to remain in dry storage configuration for period that could easily reach at least 100 years and potentially more. In the United States, there are 72 000 MTU of SNF discharged including 22 000 MTU in dry storage (about 2000 casks/canisters) at 63 operating dry storage facilities. In Germany, as of today about 1000 dry storage casks are stored at 16 sites including on-site and consolidated storage sites as transportation of UNF is prohibited nowadays. With reactor shutdowns by 2025, it is expected to increase to a total of 1600 dry storage casks (equivalent of 17 000 MTU). Regulators are in the process of defining a program to monitor and maintain dry storage systems on site to ensure of same level of safety and security. In this early stage of license renewals, aging management programs include mainly periodic inspections of the used fuel dry storage systems and components to ensure potential aging effects are identified and effectively managed. As our industry learns through R&D studies and demonstration and surveillance programs, aging management programs will evolve with advanced inspection technologies and industry operating experience as they become available in the future. This presentation will discuss innovative solutions being developed to monitor age-related degradation, prevent equipment failures caused by ageing and develop contingency plans as needed.

1. INTRODUCTION

Interim storage is needed in all UNF management strategies to enable the radioactivity and heat output of UNF and high level wastes (HLW) to decay for downstream processes. While wet storage provides easy access to the fuel to monitor its behavior over time, dry storage presents more challenges. In addition, with persistent delay in the development of permanent disposal capabilities in many countries, on-going studies to better understand the fuel behavior in dry storage conditions over time and also monitoring of the aging of systems initially designed for shorter durations are crucial. Some component aging mechanisms are well known, and a great deal of information is available from power plant experience. However, for some ageing mechanisms while well understood from scientific perspectives, there is limited operating data to predict their occurrence. As we learned from studies and operational site inspection experience, risk assessments will need to be updated and aging management programs upgraded accordingly. Development of ageing management programs is going to be the basis for obtaining license extension.

In any cases, extending licenses of dry storage facilities will demand continuous monitoring.

2. STATUS DRY STORAGE

The need to develop ageing management programs for interim dry storage is applicable in the countries that deployed nuclear energy in the 1970's and have implemented used fuel dry interim storage solutions while evaluating geological repositories or recycling solutions: USA, Germany, Belgium, Switzerland and Japan with the largest inventories of used fuel[1].

[1] Eastern Europe countries excluded.

In the United States, about 33% of the total spent fuel inventory representing about 2100 dry storage systems are currently loaded on around 78 different sites. Less than 10% of the fuel in dry storage is high burnup fuel. The initial license being limited to 20 years in the USA, a number of limited sites have / are renewing their license.

In Germany, dry storage was first implemented in 1992 at the storage facility TBL Ahaus after change in policy regarding recycling. There are currently more than 1000 dual purpose casks loaded at 15 different sites (12 on-site storage facilities as well as 3 centralized storage facilities) and there will be a total of 1600 casks in 2025 after all reactors have been shut down and unloaded. High burnup and MOX fuel is estimated to represent around 10 percent of the total inventory.

Dry storage was first implemented in 1982 in Switzerland and 1995 in Belgium. High burnup and MOX fuel represents about 15 percent of existing inventory. Extension of licenses will be needed. The policy in these two countries is to limit dry storage to a time period of 40 years. All casks require a type B transport license with respect to the IAEA regulations and must maintain transportability during storage.

Finally, Japan who has elected a closed fuel cycle with recycling has also deployed interim dry storage for operational flexibility of the reactors. In Japan, there are currently only 19 dual purpose casks stored on sites. First systems were deployed in 1995 at the Fukushima Daiichi Nuclear Power Station. Licenses have been granted for a period up to 50 years. However, testing of full integrity and inspection is required during this period.

3. REGULATORY STATUS

Regulators are upgrading requirements to allow extension of the current licenses. In the United States, to renew an ISFSI license following to the initial period (20 years), the licensee must submit a request to US Nuclear Regulatory Commission with all information to demonstrate that the site and the storage system can continue to operate for another license period (40 years).

NRC's guidance to prepare the request is provided in the Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificate of Compliance (NUREG-1927).

The license renewal application will include a review and evaluation of the Structures, Systems and Components important to safety as existing in the ISFSI license and / or CoC considering 'Ageing' for the extended period.

A time limited ageing analysis (TLAA) need to be performed complemented by inspections as needed. Following this TLAA and inspection, an aging management review shall be deployed in order to develop an ageing management programme. This ageing management programme must be approved by the NRC for license extension. Ageing management programme will be updated based on result of inspections and operational experience (learning ageing management).

In Europe (Germany, Switzerland, Belgium), the site storage facilities and the storage casks are licensed accordingly to sustain a period of time of 40 years (with an exception for Spain where initial license is 20 years with an option of extension of 20 years.). All storage casks require a type B transport license with respect to IAEA regulation and must maintain transportability during storage in order to enable shipping the casks from the storage facility

whenever needed or at the latest when site storage license expires. The transport license of the storage cask has to be renewed every 5 years.

In Japan, the policy is to closely integrate interim storage facility regulations and transportation regulations of the dual purpose casks. The operator must accumulate knowledge related to long term integrity of the metal casks and their contents so to ensure as much as possible transportation safety after storage. The interim storage facility is licensed for 50 years.

Only US and Spain regulations issue initial license with the ability to apply for extension after initial period. Other countries have not yet upgraded their regulations to extend license as license where issued for a longer time period that the United States.

4. MAIN CHALLENGES WITH LONG TERM STORAGE AND/OR EXTENDED DRY STORAGE

Beyond this interim storage period, used fuel was planned to be transported to a geological repository or recycling facilities. Due to persistent delays in implementing geological repositories, there is a need to extend dry storage beyond initial time period. In order to guarantee the safety of systems, two key parameters have to be monitored and/or mitigated:

- Ageing and degradation of the storage systems: The main functions of the storage system are radiation protection and containment. Ageing management programmes must ensure that all the functions of the storage systems remain intact. Integrity of the canister / cask is particularity of importance especially with respect to potential chloride induced stress corrosion cracking (CISCC);
- Evolution of fuel mechanical characteristics over time. A dry storage system is licensed for a specified content. There are more uncertainties for high burnup fuel assemblies degradation over time (US regulation define high burn fuel when more than 45 GWd/THM). Consequently, risk of deterioration of the cladding making fuel retrievability more complex needs to be further evaluated as authorized content could be altered.

The two parameters above are the most critical for developing a robust ageing management plan. In addition, over time, factors such as record keeping with retirement and availability of qualified and experienced individuals at shutdown sites must also be mitigated.

AREVA TN has deployed two types of storage systems currently:

- NUHOMS Canister type systems with concrete storage modules or overpacks;
- TN 24 Dual purpose casks.

While the first system type is mostly designed with a welded lid, the dual purpose cask is designed with a bolded lid. Consequently, different monitoring methods and mitigation plans must be deployed for those two different systems.

5. AGEING MANAGEMENT PROGRAMME AND SOLUTIONS

An ageing management programme includes three main phases:

- Inspection;
- Monitoring,

- Remediation.

Effective ageing management programmes require a technical understanding of the ageing degradation mechanisms, inspection and assessment techniques, mitigation measures, and as needed guidance on repairs or replacements. For this purpose, innovative solutions need to be developed for systems to be deployed in order to facilitate inspection and monitoring. Further, it is necessary to develop innovative solutions to mitigate risks related to aging management.

5.1. Inspection and monitoring

Inspection and monitoring of dry storage need implementation of sensors related to aging management (cracking, surface chemistry, internal gas composition, leakage, temperature, humidity). Some of them are already in operation: depending on the facility and requirement: pressure, temperature and dose are currently monitored.

5.2. Material degradation: corrosion inspection

The potential canister or cask material degradations are: Stress Corrosion Cracking (SCC), aqueous corrosion, and atmospheric corrosion.

Utilities and vendors have performed in-situ inspections of actual canisters measurement of surface temperatures, visual and surface chloride sampling. AREVA TN contributed to the development of tools to provide full canister inspection (surface chemistry) capability for stress corrosion cracking in chlorine induced SCC environments. Analysis has demonstrated the effectiveness of the sampling at Calvert Cliffs NPP. The innovative measurement technology (Fig. 1) enables the operators to determine the amount of salt on the exterior of the canister.

AREVA TN is studying and developing remote-weld inspection techniques that are more sensitive, simple to deploy and cost effective.

FIG. 1. SaltSmart™ delivry tool on canister surface.

5.3. Leaktight and temperature inspection

The containment function of the radioactive material must be maintained in the long-term.

Monitoring pressure systems operate on AREVA TN dual purpose casks since several decades and it would be incorporated on the TN NOVA™ system. This system monitors the pressure permanently; three pressure sensors continuously measure the pressure between inner lid and outer lid (in order to provide redundant information); and an alarm system alerts

the operators for decreases in pressure or for abnormal increases to the rate of pressure change.

The Fig. 2 presents the pressure monitoring system.

FIG. 2. Preparation of monitoring system on TN®24 casks (AREVA).

Temperature monitoring system could be also added on the cask in order to check the cask surface temperatures continuously or periodically. The records from the temperature measurements will allow the comparison to the calculated temperatures at different times along the storage period and provide information of the cask component behaviour.

5.4. Remediation: double walled canisters

As a back-up in case of issue, a remediation plan needs to be deployed. For example, in case of detection of potential breach of containment, a 'contingency storage module' with a pre-installed 'outer' canister would allow the loading of the potentially leaking 'inner' canister.

AREVA TN has designed several double walled canister systems providing two independent containments comprising an inner canister where the fuel is loaded and an outer canister which enclosed the inner canister. This solution has been patented in France and in US (number patent respectively FR0350775 and US 10/578 147). The fuel loading operations and the closure operations of the inner and outer canister are similar to the ones performed for classical canister (DSC) in accordance with the Technical Specifications and Requirements of the NUHOMS® license.

An alternative solution has been also designed recently which allow the storage of a canister inside a storage module (horizontal or vertical module) equipped with a liner. When the canister loaded with used nuclear fuel, is inserted inside the liner, the liner is then closed by a lid providing an additional and independent containment boundary to the fuel. The annulus between the two shells (DSC and liner) is filled with inert gas and could be easily monitored. The advantage of this solution is the compatibility of the existing canister which involves no modification the loading procedure inside the reactor building (such as lid welding) and no modification of the transportation procedure. Indeed, the liner is used only in storage providing the additional confinement. In case of failure of the liner due to potential SCC degradation for example, the DSC with the fuel could be removed from the storage module using the transfer or the transport cask and the liner could easily be removed or exchanged without damaging on the fuel placed inside the DSC. After replacing the liner, the DSC could be inserted in the same module equipped with the new liner.

The Fig. 3 presents the double walled canister comprising a DSC which contained the fuel and a liner housing the said DSC.

This solution has been patented in France under number patent FR2969362 and the patent is pending in US under number US2013261367.

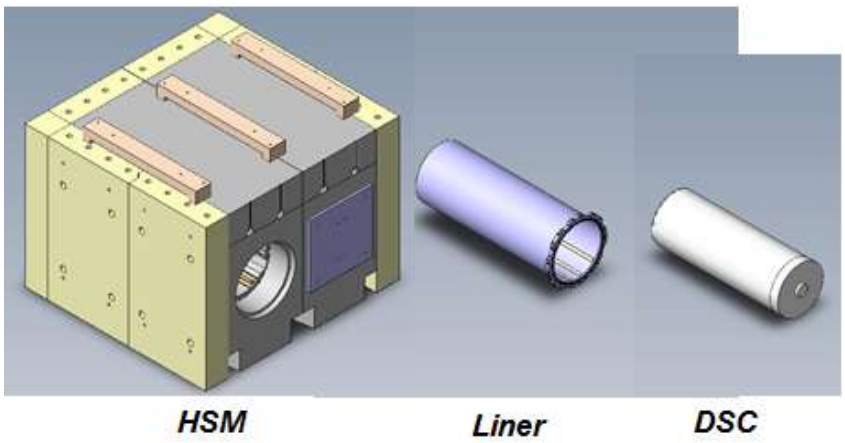

FIG. 3. Double walled canister in storage configuration.

5.5. Innovative solutions

AREVA has developed some innovative interim dry storage systems which allow the storage for long duration and mitigate the aging effects in order to maintain all of the safety functions of the systems.

5.6. SCC proof canister

As explained previously, atmospheric chloride-induced stress corrosion cracking (SCC) of stainless steels have been identified by the NRC as a potential degradation risk that can breach the canister shell by cracking when exposed to atmospheric chlorides like those found in marine environments.

AREVA has developed an alternative to its canister using Duplex SS instead of current Stainless Steel. The NRC has reviewed the licensing approach for Duplex SS in NUHOMS® EOS (pre-meeting) in October 2014 and found no impediment to licensing. The advantage of this solution is the minor design change on the original NUHOMS® license and the full compatibility with the existing storage module or transport or transfer systems.

6. CONCLUSION

The pace of the work being done to find sites for a permanent repository for spent fuel and high-level radioactive waste in various countries is considered concerning by some stakeholders. While we can acknowledge that progresses already made, one of the main objectives in the management of radioactive waste is to not impose any unreasonable burden on future generations. Countries should develop strategies and action plans to speed up the selection process and deployment of permanent repositories. Meanwhile, Interim dry storage solution is a key component of a used fuel management program. Safety and continued

robustness of the dry storage solutions will require innovative solutions to monitor and mitigate any potential deterioration of the systems. Existing technical gaps need to be addressed. International exchanges on operational lessons learned will also be an important element to support safe extended dry storage.

In parallel, permanent disposal facilities need to be actively developed in order to minimize interim storage at sites.

RENEWING DRY SPENT FUEL STORAGE CERTIFICATES OF COMPLIANCE AND SPECIFIC LICENSES

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Abstract

The United States (U.S.) Nuclear Regulatory Commission (NRC) regulations for storage of spent nuclear fuel (SF) permit initial storage terms of up to 40 years. Certificates of Compliance (CoCs) for SF storage system designs, or specific licenses to store SF, can then be renewed for up to 40 years. Renewal authorization requires time-limited aging analyses and aging management programs (AMPs) that demonstrate that structures, systems, and components important to safety will continue to perform their intended functions for the requested period of extended operation. NRC staff has recently completed the renewal review for a specific license at the Calvert Cliffs Nuclear Power Plant site, and is currently reviewing renewal applications for a second specific license and two CoCs. In addition, work is underway on a revision to the staff guidance for the safety review of specific-license and CoC renewal applications. The guidance will define an acceptable method for the NRC staff to review and determine if the applicant demonstrates that the storage system, or independent spent fuel storage installation, will continue to meet the applicable regulatory requirements during the renewal duration. In addition, the guidance will provide examples of aging management programs for: (1) localized corrosion and stress corrosion cracking of welded stainless steel dry storage canisters; (2) reinforced concrete structures; and (3) high burnup fuel performance. This paper will address the recent experience of the NRC staff in reviewing renewal applications and in developing the revised staff guidance, including discussion of example AMPs for concrete structures, and for localized corrosion and stress corrosion cracking of welded stainless steel dry storage canisters. Renewal authorization requires time-limited aging analyses and aging management programs (AMPs) that demonstrate that structures, systems, and components important to safety will continue to perform their intended functions for the requested period of extended operation.

1. BACKGROUND

The United States (U.S.) Nuclear Regulatory Commission's (NRC) regulations [1] allow for storage of spent nuclear fuel (SF) for an initial storage term of up to 40 years. Certificates of Compliance (CoCs) for SF storage system designs, or specific licenses to store SF, can then be renewed for up to 40 years. Renewal authorization requires time-limited aging analyses and aging management programs (AMPs) that demonstrate that structures, systems, and components important to safety will continue to perform their intended functions for the requested period of extended operation.

Aging management programs (AMPs) monitor and control the degradation of structures, systems, and components, within the scope of the renewal period so that aging effects will not result in a loss of intended functions during the period of extended operation. An AMP includes all activities that are credited for managing aging mechanisms or effects for specific structures, systems, and components. An effective AMP prevents, mitigates, or detects the aging effects and provides for the prediction of the extent of the effects of aging and timely corrective actions before there is a loss of intended function.

Aging management programs should be informed, and enhanced when necessary, based on an ongoing review of both site-specific and industry-wide operating experience. Operating experience provides direct confirmation of the effectiveness of an AMP, and critical feedback for the need for improvement. As new knowledge and data become available from new analyses, experiments, and operating experience, licensees and CoC holders should revise

existing AMPs (or pertinent procedures for AMP implementation) to address program improvements or aging issues.

The AMP commences at the end of the initial storage period for each dry storage system loaded under a renewed certificate of compliance. Activities in an AMP may start at different times (e.g., an AMP can be implemented before the period of extended operation to capture baseline data, and AMPs may have different frequencies of implementation). If the initial term of a certificate of compliance exceeds 20 years, the AMP may have been incorporated into the original certificate of compliance. In that case, the AMP commences at the time indicated in the certificate of compliance.

2. RENEWAL REVIEWS

The NRC has completed five site-specific license renewals. The first storage renewal application was submitted in 2002 for the Surry Power Station and completed in 2005. Following the Surry review, the NRC reviewed applications for H.B. Robinson Plant, Oconee Nuclear Station, and Fort St Vrain Nuclear Power Plant. These reviews were completed before the current draft of the revised guidance, and none of them included high burnup fuel.

Most recently, the staff completed the renewal review for the Calvert Cliffs independent spent fuel storage installation (ISFSI) renewal in 2014. The Prairie Island independent spent fuel storage installation and two certificates of compliance (VSC-24 and the AREVA TN Standardized NUHOMS® Storage System) are still under NRC review.

3. RENEWAL GUIDANCE

The renewal guidance defines an acceptable method for the NRC staff to review and determine if the applicant demonstrates that the storage system, or ISFSI, will continue to meet the applicable regulatory requirements during the renewal duration.

After the initial five site-specific reviews, the agency determined guidance was needed on how to conduct the renewal reviews. In March 2011, the NRC issued NUREG-1927 [2]. This guidance was developed from the preliminary NRC guidance of Ref. [1] and adopted the regulatory philosophy of Ref. [3], and the 2011 rulemaking to update 10 CFR Part 72 [1]. The review of these most recent applications resulted in staff requests for additional information on the planned aging management reviews, planned aging management programs, and planned lead system inspections, indicating a need for enhanced and updated guidance for renewals.

The NRC established a Storage Renewal working group in December 2013 to assess the existing storage renewal regulatory framework. The team was comprised of staff from across the Agency and incorporated lessons learned from the completed and ongoing reviews. The staff held public meetings in April and July of 2014 to inform the public about the guidance update and to receive public input and guidance on AMPs. The draft revised guidance will be published in 2015 for public comment. In September 2014, the nuclear industry (Nuclear Energy Institute/NEI) issued their own guidance [4] for the NRC's endorsement. NRC staff has reviewed NEI 14-03, but has not taken any action with respect to endorsement.

The working group identified several key topics to be updated in the guidance: (1) AMPs; (2) lead inspection program; (3) high burnup fuel performance; (4) tollgates; and (5) TLAAs.

3.1. Ageing management

One of the key topics updated in the draft revised guidance was aging management reviews (AMRs). An aging management review is an assessment conducted by the licensee or CoC holder that addresses aging mechanisms and effects that could adversely affect the ability of SSCs from performing their intended functions during the period of extended operation. In contrast, an aging management program is a program conducted by the licensee or CoC user for addressing aging effects that may include prevention, mitigation, condition monitoring, and performance monitoring. The staff has updated the process flow and applicable regulatory requirements and expanded guidance on environmental data and identification of aging effects and mechanisms. Guidance on time-limited aging analysis (TLAA) was updated to ensure consistency with regulatory requirements and for calculations and analysis not part of the design basis. The staff expanded discussion on each AMP element, provided new guidance on learning AMPs and use of operating experience. Information about high burn up fuel was also added to the guidance since it was not included in the initial revision of the guidance.

The NRC believes aging management of dry storage systems should be operations-focused, learning, proactive, and responsive. AMPs should consider and respond to operating experience and results of confirmatory research. The cornerstone of operation-based aging management approach is the collection, assessment, and dissemination of operating experience. The industry is currently evaluating ways for collecting, assessing, and disseminating experience amongst operating facilities, vendors, and storage system owners. The industry also suggested the use of a tollgate approach to provide a structured way to periodically assess operating experience to determine if modifications of AMPs and TLAAs are warranted.

In addition, the staff added a new appendix with example AMPs for localized corrosion and stress corrosion cracking, concrete structures, and high burnup fuel that the staff can reference as they conduct their reviews.

Each example AMP goes through the ten elements of an aging management program and provides a description of what each element entails: (1) scope of program; (2) preventive actions; (3) parameters monitored/inspected; (4) detection of aging effects; (5) monitoring and trending; (6) acceptance criteria; (7) corrective actions; (8) confirmation process; (9) administrative controls; and (10) operating experience.

3.1.1. Localized corrosion and stress corrosion cracking of welded stainless steel dry storage canisters

Welded stainless steel canisters are used in the majority of dry storage systems in the United States for spent nuclear fuel from commercial power reactors at both specific-licensed and general-licensed ISFSIs. The welded stainless steel canisters are the primary confinement boundary during storage. While there are no known operational occurrences of aging or localized corrosion of welded stainless steel canisters, operational experience with nuclear reactors that were located close to an open ocean, or bay, has shown that pitting corrosion, crevice corrosion, and chloride-induced stress corrosion cracking (CISCC) can occur in welded stainless steel components as a result of atmospheric deposition and deliquescence of chloride-containing salts. Laboratory and natural exposure tests suggest that CISCC can occur with sufficient surface chloride concentrations and that crack propagation rates can be of engineering significance for welded stainless steel canisters during the period of extended operation.

Based on reactor operating experience as well as laboratory and field testing, localized corrosion and CISCC are potential aging mechanisms for welded stainless steel canisters. Environments where chloride-containing salts may be deposited on welded stainless steel canisters include coastal locations near salt water, and locations that are close to cooling towers or roads that are salted. To address these potential aging effects, an AMP for welded stainless steel canisters used in dry storage systems was developed by the NRC that relies on guidance from consensus codes and standards to perform in-service inspection of canisters to detect localized corrosion and CISCC. The recommended parameters for monitoring and/or inspection include:

- Visual evidence of discontinuities and imperfections such as localized corrosion, including pitting corrosion, and crevice corrosion and stress corrosion cracking of the canister welds and weld heat affected zones;
- Size and location of localized corrosion and stress corrosion cracks;
- Appearance and location of deposits on the canister surfaces.

These inspection and monitoring parameters are one part of the full example AMP which includes all ten elements of an aging management program listed above.

3.1.2. Reinforced concrete structures AMP

The AMP for reinforced concrete structures consists of condition monitoring, performance monitoring, mitigation and prevention activities. The program includes periodic visual inspections by personnel qualified to monitor reinforced concrete for applicable aging effects, such as those described in the American Concrete Institute standards (ACI) and American National Standards Institute / American Society of Civil Engineers standard (ANSI/ASCE). Identified aging effects are evaluated against acceptance criteria derived from the design bases or industry codes and standards, including ACI standards and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

This AMP also includes periodic sampling and testing of groundwater and the need to assess the impact of any changes in its chemistry on below-grade concrete structures. Additional activities include radiation surveys to ensure the shielding functions of the concrete structure are maintained and daily inspections to ensure the air convection vents are not blocked (per the requirements of the approved design bases). The AMP also includes provisions where modifications may be appropriate for specific-license renewals.

3.1.3. High burnup fuel performance AMP

An example of a high burnup (HBU) fuel (i.e., fuel assemblies with discharge burnup greater than 45 gigawatt-days per metric ton of uranium [GWd/MTU]) monitoring program has been added to the guidance. The AMR is not expected to identify any aging mechanisms and effects that could lead to a loss of intended function (primarily: retrievability; secondary: criticality, shielding, and confinement), as long as the HBU fuel assemblies are stored in a dry inert environment and temperature limits are maintained. Short-term testing (i.e., laboratory scale testing up to a few months) and scientific analyses examining the performance of HBU fuel have provided a sound foundation for the technical basis that long term storage of HBU fuel (i.e., greater than 20 years) may be performed safely and in compliance with regulations. However, scientific analysis is not a complete substitute for confirmatory operating experience, and there has been relatively little operating experience, to date, with dry storage of HBU fuel. Therefore, the purpose of the HBU fuel monitoring program is to provide

confirmatory data regarding the continuation of the HBU fuel assemblies' intended function(s) during the period of extended operation. Although this program is a confirmatory program, it uses each attribute of an effective AMP for the renewal of a specific license to the extent possible.

3.2. Lead system inspections

The staff has added new guidance for reviewing lead system inspections in renewal applications. The NRC staff expects a lead system inspection to be performed prior to the submittal of a renewal application. The lead system inspection is an important element in an operations-focused aging management approach and provides valuable operating experience. At this point, due to insufficient operating experience, the use of surrogates for a specific license is not justified. The NRC staff believes that baseline inspections should be completed at each site prior to entering the period of extended operation.

Dry storage system materials are selected to be resistant to environmentally-induced degradation during the initial storage period. However, because inspections of the dry storage systems are not typically conducted, or are limited in scope during the initial storage period, it is important for an applicant to determine the condition of the system and demonstrate that structures, systems, and components within the scope of the renewal application have not undergone unanticipated aging and degradation prior to entering the period of extended operation. If the system inspection reveals unanticipated aging or degradation, then the applicant would need to address the condition of the structures, systems, and components in the renewal application. A lead system inspection is an NRC staff-accepted approach to verify system condition at an independent spent fuel storage installation.
This inspection may provide data to confirm a TLAA prediction of the system's condition at the end of the initial storage period or to revise TLAAs for the period of extended operation. The results from the lead system inspection will be used by the applicant to develop detailed AMPs, including inspection and maintenance frequencies, and provide a baseline for the monitoring and trending element of AMPs in the period of extended operation.

If multiple storage systems or designs are in use at the independent spent fuel storage installation, then the lead system inspection should address the different storage systems or designs that are used. The inspection should include all SSCs within the scope of the renewal application.

The selection of system(s) for inspection should consider the results of the aging management review and should be based on the degradation mechanism(s) of concern for the structures, systems, and components. Structures, systems, and components for inspection should be selected based on site-specific conditions, system design, material combinations, and operating parameters that may contribute to aging and degradation.

3.3. High burnup fuel performance

The draft guidance provides an appendix that gives NRC staff a basis for reviewing if a demonstration program has the necessary properties to qualify as one method that an applicant might use in license and CoC applications to confirm the integrity of HBU fuel during continued storage.

The experimental confirmatory basis that low burnup fuel (≤45 GWd/MTU) will maintain its integrity in dry cask storage over extended time periods has been provided in a demonstration test [5, 6]. A similar confirmation test, which includes information over a similar length of the

time available for low burnup fuel, does not exist for other light water reactor (LWR) fuels, HBU fuel and mixed oxide fuels.

There is no evidence to suggest that HBU fuel cannot similarly be stored safely and then retrieved for time periods beyond 20 years, but the supporting experimental data is not extensive. Therefore, confirmatory data or a commitment to obtain data on HBU fuel and taking appropriate steps in an AMP will provide further information that will be useful in evaluating the safe handling of individual assemblies of HBU fuel for extended durations. The draft guidance provides potential acceptable methods for demonstrating compliance.

3.4. Tollgates

One of the principles introduced in NEI 14-03 is the use of 'tollgates' as a structured approach for assessing operating experience and data from applicable research and industry initiatives. NEI 14-03 defines 'tollgates' as periodic points within the period of extended operation when licensees would be required to document an aggregate safety assessment. Tollgates are described as an additional set of in-service assessments beyond the normal continual assessment of operating experience, research, monitoring, and inspections on dry storage system component performance that is part of normal ISFSI operations for licensees during the initial storage period as well as the period of extended operation. Staff has provided recommendations in the draft guidance for assessing applicant plans for use of tollgates.

3.5. Time-limited ageing analyses

Time-limited aging analyses are licensee or CoC holder calculations or analyses used to demonstrate that systems, structures and components designated important to safety maintain their intended function throughout an explicitly stated period of extended operation. Staff has provided recommendations in the draft guidance for assessing applicant plans for use of tollgates.

4. CONCLUSIONS

NRC staff is revising its guidance for renewal reviews for dry storage systems and ISFSIs based on completed and ongoing renewal reviews, and incorporating industry input. The draft revised guidance contains information on aging management reviews, aging management programs, time-limited aging analyses, and other tools and criteria. Example AMPs have been developed for stress corrosion cracking, high burnup fuel, and concrete structures. The draft revised guidance will be published for public comment in 2015.

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DEMONSTRATION TEST PROGRAM FOR LONG TERM DRY STORAGE OF PWR SPENT FUEL

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Abstract

In Japan, the first interim spent fuel storage facility away-from-reactor (AFR) will start its operation. This facility can store BWR / PWR spent fuel assemblies using dry metal dual purpose casks (storage / transport) which will be transported to their destinations after the interim storage for decades. This facility is not equipped with a hot-cell for opening the primary lid of the cask because one of the basic concepts the facility is a simple operation not to handle a radioactive material directly, that reduces radiation exposure of workers and a risk of contamination troubles. Visual inspections of spent fuel assembles cannot be carried out in the facility, although a visual inspection of spent fuel assemblies is usually carried out before spent fuel transportation. Therefore, we are preparing for the demonstration test designed to confirm the spent fuel integrity during a long-term dry storage by use of the test container which is able to reproduce storing the PWR spent fuels in the similar environment to actual casks. This paper describes specifications of the test container and the heat transfer test results. In addition the preparation works including the thermal analysis for estimation of the stored fuels and the fuel inspection results before storage are discussed.

1. INTRODUCTION

Recyclable Fuel Storage Center in Mutsu City, Aomori Prefecture, which is the first interim spent fuel storage facility in Japan, is preparing for maximum 50 year storage of spent fuels in dry metal casks for both transportation and storage. Spent fuels whose integrity was confirmed will be loaded and transported to the facility, and stored over the long term, maintaining its integrity based on research achievements in Japan and overseas. In addition, to reduce risk of exposure to workers and waste materials, the interim storage facility has no hot cell, and the spent fuels will be confirmed for their integrity indirectly by monitoring cask during storage and transported after the storage without opening the cask lid.

Currently, spent fuel storage in dry metal casks is a proven method mainly in Europe and the United States based on numerous tests including demonstration tests. In Japan, long-term integrity of PWR spent fuels is investigated and interim storage is planned under the conditions maintaining fuel integrity. On the other hand, Nuclear Regulation Authority (Formerly Nuclear Safety Commission of Japan) required the utilities to accumulate knowledge and experience on fuel integrity during dry storage, considering experience of spent fuel dry storage in Japan and the above facility features. Therefore, BWR fuels that are stored using dry casks in Fukushima Daiichi and Tokai Daini reactor sites have been confirmed several times for their integrity during storage.

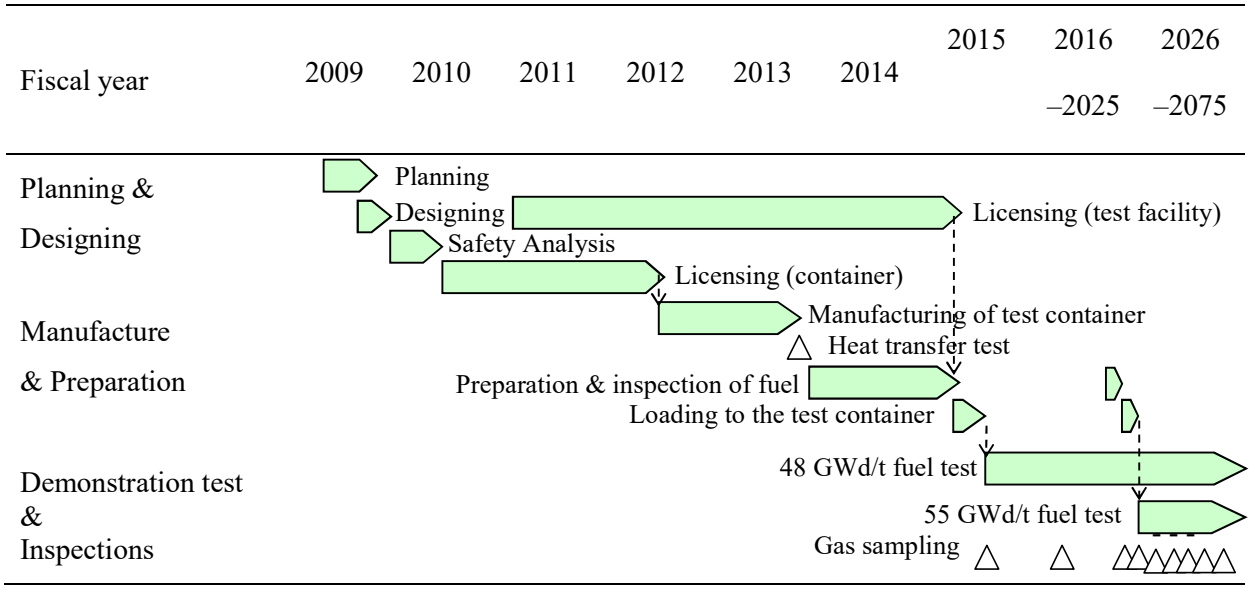
Under the circumstances, the utilities plan a demonstration test for fuel integrity conducted in the domestic research facility (Nuclear Development Corporation (hereinafter called NDC)) to accumulate knowledge and experience on long-term integrity of PWR spent fuels during dry storage.

2. DEMONSTRATION TEST PROGRAM

2.1. Test overview and process

Integrity of 48 GWd/t type spent fuel and 55 GWd/t type spent fuel are planned to be investigated in this demonstration test. The design of test container was already licensed and the test container was manufactured in fiscal 2013. Now the final licensing procedure for the test facility, NDC, is being performed. The demonstration test will start at the middle of fiscal 2015. For the first 10 years, only one 48 GWd/t type fuel will be stored. Another 55 GWd/t type fuel will be additionally loaded in fiscal 2025. The test is planned to continue for 60 years. In the demonstration test, it will be confirmed that appropriate test temperature, atmosphere in the container and no significant external force are maintained by monitoring temperature, visual appearance of the test container and lid sealing performance. It will be also confirmed that the spent fuel integrities are maintained by conducting internal gas sampling and Kr-85 analyses periodically (Table 1).

TABLE 1. TIME SCHEDULE OF DEMONSTRATION TEST OF PWR SPENT FUEL STORAGE



2.2. Fuel specification for demonstration test

Fuel specification for the demonstration test is shown in Table 2. In September 1993, one of 48 GWd/t type fuels unloaded from Kansai Electric Takahama Unit 3 was transported to the hot laboratory in Tokaimura (NDC). Post irradiated examinations (hereinafter called PIE) of some fuel rods were conducted and now the fuel is being stored in the pool of the facility. 55 GWd/t type fuel is only assumed and a proper spent fuel will be prepared for the test in the future.

TABLE 2. FUEL SPECIFICATION FOR DEMONSTRATION TEST

Fuels for test	48 GWd/t Type fuel	55 GWd/t Type fuel (assumption)
Type	17×17	17×17
Burnup (MWd/t)	42 800 (past record)	≤55 000
Cooling period (years)	21 (as of June 2015)	10 or more (as of October 2025)
Cladding material of fuel rods	Zircalloy–4	MDA or ZIRLO
Remarks	15 empty fuel rods*	Non

*Fuel rods used for PIE are not reinserted into the fuel assembly.

2.3. Outline of container

To meet the test purpose, the test container is designed to store two of Type 17 × 17 PWR spent fuels for 60 years. Also, it is designed to be able to simulate temperatures, pressures and atmosphere of an actual metal cask. Furthermore, maintenance of sealing, subcriticality and shielding performance is required to implement the test safely. Outline of the test container is shown in Fig. 1.

A basket spacer (part of a basket) is made of boron containing aluminium alloy to maintain subcriticality. A container body mainly consists of an inner cylinder as a containment boundary, a thick mid-body for gamma-ray shielding and resin for neutron shielding. A thermal insulator is placed between the inner cylinder and the mid-body as a temperature control component simulating actual cask temperature, since a design heat load of two fuels stored in the test container is much lower than the assumed heat load of an actural cask. For the first 10 years, only one 48 GWd/t type fuel will be stored with lower heat load, so that another insulator is placed outside of the container to adjust its initial temperature.

The container has a single lid with double metal gaskets. A performance of containment boundary at lid will be monitored by monitoring pressure between the gaskets.

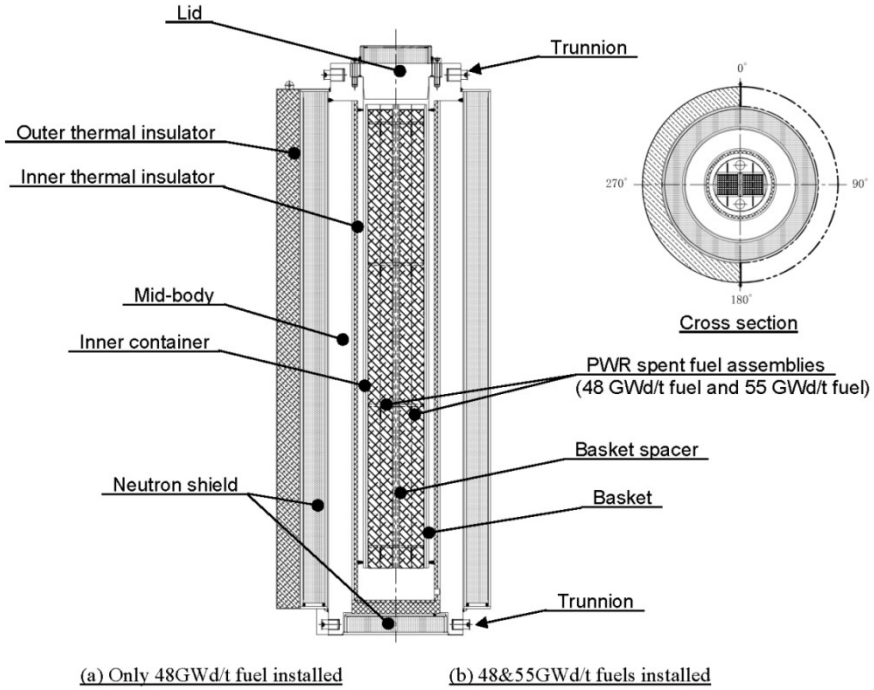


FIG. 2. Test container profile.

2.4. Verification method of fuel integrity

Fig. 2 shows a flow diagram of the demonstration test. The following inspections to confirm integrity of fuels stored are conducted before, at the beginning of, and during the demonstration test.

2.4.1. Confirmation before demonstration test

The spent fuel assembly used for the demonstration test is inspected visually in the pool of the facility before the demonstration test. The integrity of the 48 GWd/t type fuel was confirmed by observing visual appearance of 4 outermost surfaces of the fuel assembly with an underwater camera last year.

2.4.2. Confirmation at beginning of demonstration test

After a loading operation of 48 GWd/t type fuel into the test container and also after an additional loading operation of 55 GWd/t fuel, it is confirmed that there is no leak of the fuel rods by conducting gas sampling and analyses of Krypton-85 and compositions.

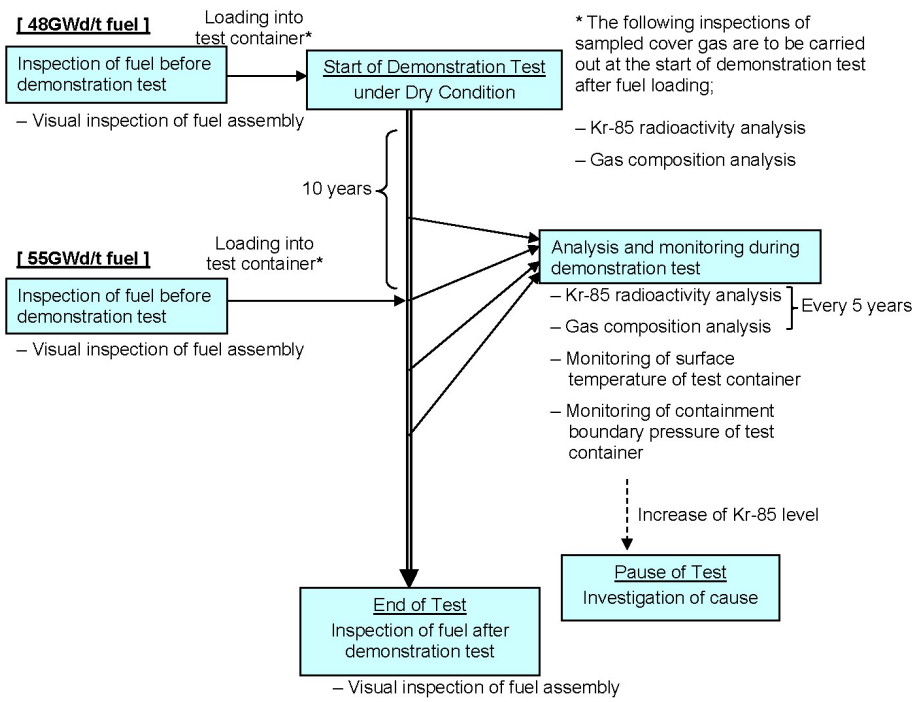


FIG. 3. Flow diagram of test programme.

2.4.3. Confirmation during demonstration test

The initial test conditions are confirmed to be maintained by periodical sampling and analyses of gas from inside of the test container and monitoring of its temperature and pressure from outside during storage until the completion of the test.

2.4.3.1. Gas sampling

For the purpose of detecting fuel rod failure, gas samples inside of the test container are taken to conduct gas analyses periodically every 5 years. The gas sampling is conducted by connecting the previously-evacuated sampling pod to the lid port leading to inside of the test container through a valve. The sampled gas is analyzed for radioactive gas (detection of ^{85}Kr) with a Germanium semiconductor detector and components with a mass spectrometer. If any problems like significant increase of ^{85}Kr level are detected, the test will be paused to evaluate effect on safety and investigate the cause. When examination of fuel integrity shows a possibility of fuel failure, the fuel might be investigated in detail.

2.4.3.2. Monitoring of containment

Monitoring of the following items is performed determining its frequency during the test:

a) Temperature monitoring:

With the aim of estimating temperature history of spent fuels, thermocouples are installed on the outer surface in the middle area of the test container to monitor temperature. Temperature of spent fuels is estimated based on the outer surface temperature and external temperature by using a previously-verified assessment method for temperature of the test container and fuel assemblies.

b) Confirmation of integrity of the test container:

To confirm containment integrity of the test container, pressure monitoring is performed using pressure gauges installed to a buffer tank leading to gap between double metal gaskets, where is pressurized with helium gas (Fig. 3). Gas temperature is also measured. Also, to confirm that there is no problem on the test container, visual inspections of surface of the test container and its fixing conditions are conducted.

c) Inspection of measuring equipment:

Thermocouples, temperature indicators, pressure gauges, etc. are calibrated or replaced with calibrated equipment to provide quality control of measuring equipment.

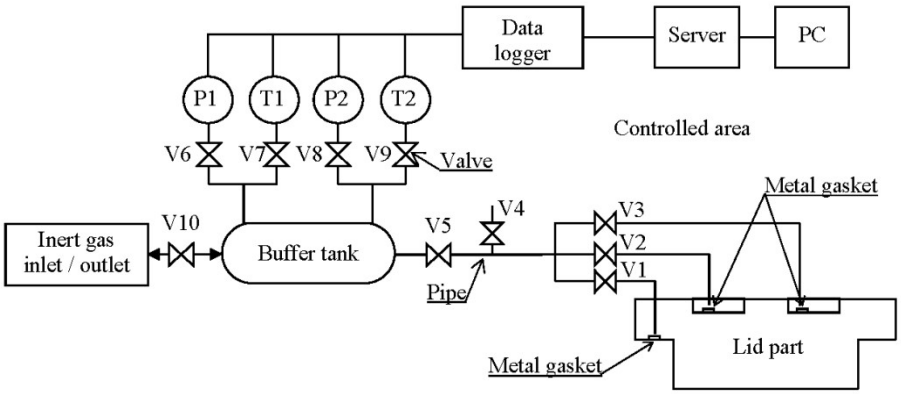


FIG. 4. Schematic drawing of pressure monitoring.

3. TEST CONTAINER

3.1. Manufacturing of test container

The test container used for the demonstration test was manufactured from fiscal 2012 to 2013. The photos of the test container are shown in Fig. 4. The material list and the main specification of the component of the test container are shown in Table 3 and Table 4.

(Top of the test container)

(Full view of the test container)

FIG. 5. Test container (photos).

TABLE 3. MATERIAL LIST OF THE COMPONENT OF THE TEST CONTAINER

Component	Material	Standard
Inner container/Flange/Bottom plate	Stainless steel	JIS G 3214 SUSF304L
Mid-body	Carbon steel	ASTM A350M LF5 Class1
Outer shell	Stainless steel	JIS G 4304 or JIS G 4305 SUS304
Trunnion	Stainless steel	JIS G 4303 SUS630-H1150
Inner thermal insulator	Ceramic fibre	
Neutron shield	Resin	
Basket	Stainless steel	JIS G 4304 or JIS G 4305 SUS304
Basket spacer	Boron contained aluminium	
Lid	Stainless steel	JIS G 3214 SUSF304

TABLE 4. MAIN SPECIFICATION OF THE TEST CONTAINER

Item	Specification	Remark
Dimensions	φ1680mm × 5220mm	
Weight	Around 50 tonne	
Maximum number of fuels loaded into the test container	Two (2)	(48 GWd/t and 55 GWd/t)

3.2. Heat transfer test

To evaluate a thermal performance of the test container, heat-transfer tests were carried out using the test container. Outline of the heat-transfer tests is shown in Fig. 5. Two mock-ups of fuel with heaters which simulate the decay heat of spent fuel assemblies were stored into the test container. Temperature of each component of the test container was measured during the heat-transfer test.

Two heat-transfer tests were conducted. Test conditions of each test are shown in Table 5. In the first heat-transfer test (case 1), only one of the mock-ups of fuel which simulate the decay heat of the 48 GWd/t type fuel was heated under the vacuum condition of the cavity. In the second test (case 2), both mock-ups of fuel were heated under the condition with helium gas in the cavity. Temperatures at each point when the mock-up of fuel reached to the maximum temperature are listed in Table 6.

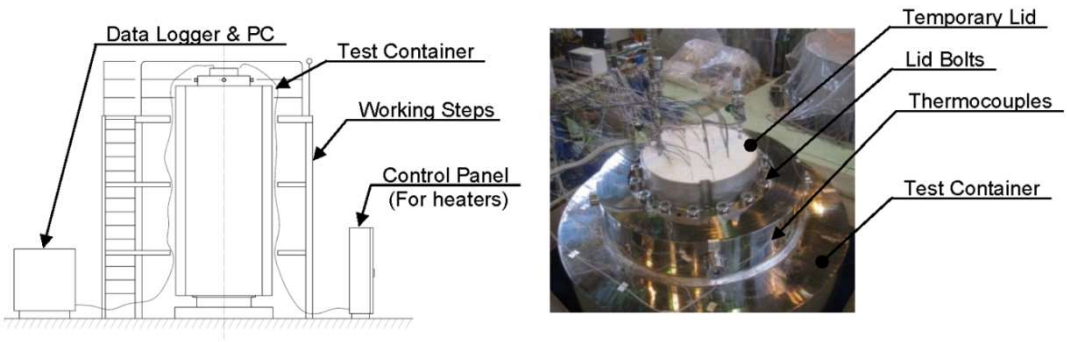


FIG. 6. Outline of heat-transfer test.

TABLE 5. TEST CONDITIONS OF HEAT-TRANSFER TESTS

Test case	Case 1: One heater (simulated 48 GWd/t fuel)	Case 2: Two heaters (simulated 48 and 55 GWd/t fuels)
Heat load	513 W (513 W / -)	1397 W (428 W / 969 W)
Filling gas into the cavity	Vacuum	Helium gas
Outer thermal insulator	Attached	Detached

TABLE 6. TEMPERATURES OF EACH COMPONENT WHEN MOCK-UP OF FUEL REACHED TO MAXIMUM TEMPERATURE

Component	Case 1: One heater	Case 2: Two heaters
Outer shell	22.5˚C	18.5˚C
Inner body	79.7˚C	158.4˚C
Basket	137.1˚C	171.1˚C
Mock-up of 48 GWd/t fuel	206.2˚C	186.5˚C
Mock-up of 55 GWd/t fuel	—	192.4˚C
Lid (inner surface)	24.3˚C	53.0˚C
Bottom plate (inner/outer surface)	74.9˚C / 15.9˚C	133.9˚C / 22.3˚C
Ambient temperature	9.2˚C	10.8˚C

4. PREPARATION FOR DEMONSTRATION TEST

4.1. Thermal analysis

4.1.1. Simulation analysis for the heat-transfer test

Thermal analysis was carried out for the purpose of the simulation of the heat-transfer test to estimate the temperature of spent fuels in the demonstration test. The simulation analysis for case 1 was conducted by using ABAQUS code and the simulation analysis for case 2 was conducted by using FLUENT code in order to consider a convection of Helium gas in the cavity.

Temperature distributions of the test container in case 1 and case 2 are shown in Fig. 6 and Fig. 7 respectively. Comparing the analysis results with the heat-transfer test results, the temperatures of the simulation analysis are well agreed with the heat-transfer test results as shown in Table 7, so that the temperature evaluation method was verified.

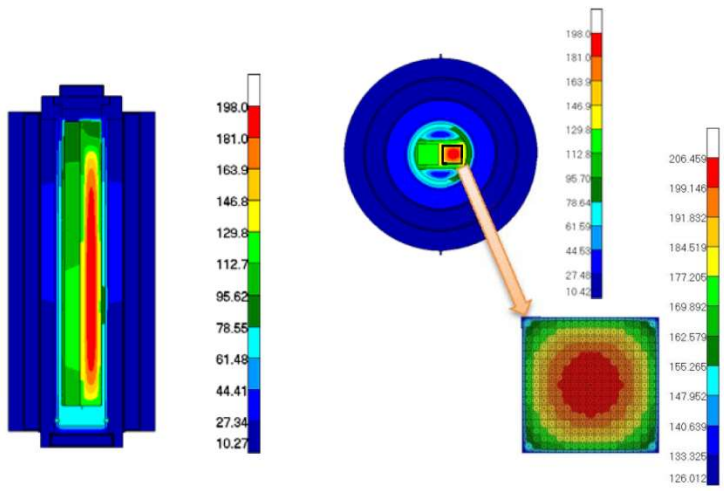


FIG. 7. Temperature distribution at the maximum temperature of heater (case 1).

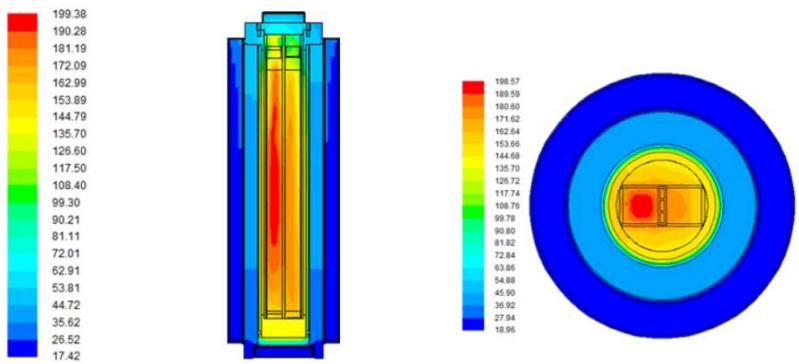


FIG. 8. Temperature distribution at the maximum temperature of heater (case 2).

TABLE 7. COMPARISON OF TEMPERATURE BETWEEN HEAT-TRANSFER TEST AND SIMULATION ANALYSIS

Component	Case 1: One heater		Case 2: Two heaters	
	Test	Analysis	Test	Analysis
Outer Shell (Outer Surface)	22.5˚C	25.6˚C	18.5˚C	19.0˚C
Mock-up of 48 GWd/t fuel	206.2˚C	205.9˚C	186.5˚C	179.8˚C
Mock-up of 55 GWd/t fuel	—	—	192.4˚C	198.6˚C

4.1.2. Evaluation of temperature during demonstration test

Initial temperature of the fuels at the installation into the test container was evaluated by using the verified method mentioned above. The evaluated fuel temperatures are shown in Table 8. As shown in Table 8, evaluated fuel temperatures are almost satisfied the target temperature

which is set according to the results of thermal analysis in the report of PWR cask [1] so that the test container has satisfying thermal performance for simulating actual cask temperature.

TABLE 8. COMPARISON BETWEEN ESTIMATED FUEL TEMPERATURE AND TARGET TEMPERATURE

Case		Evaluated temperature	Target temperature*
One fuel loaded:	48 GWd/t fuel	216˚C	200˚C
Two fuels loaded:	48 GWd/t fuel	172˚C	-
	55 GWd/t fuel	190˚C	200˚C

Design heat load including a safety margin of 5% and an ambient temperature of 10 ˚C are taken into account.

4.2. Fuel inspection

Before loading 48 GWd/t type fuel into the test container, visual inspections of the fuel were carried out as shown in Fig. 8. It was confirmed that the fuel has no significant crack, no deformation and no adhesion of foreign substances. Some of fuel rods were extracted from the fuel and the visual inspections, the dimensional inspections and the oxide film thickness measurement for these fuel rods were carried out. These inspection results could be helpful in case that any problems like a leakage of fuel rod were occurred.

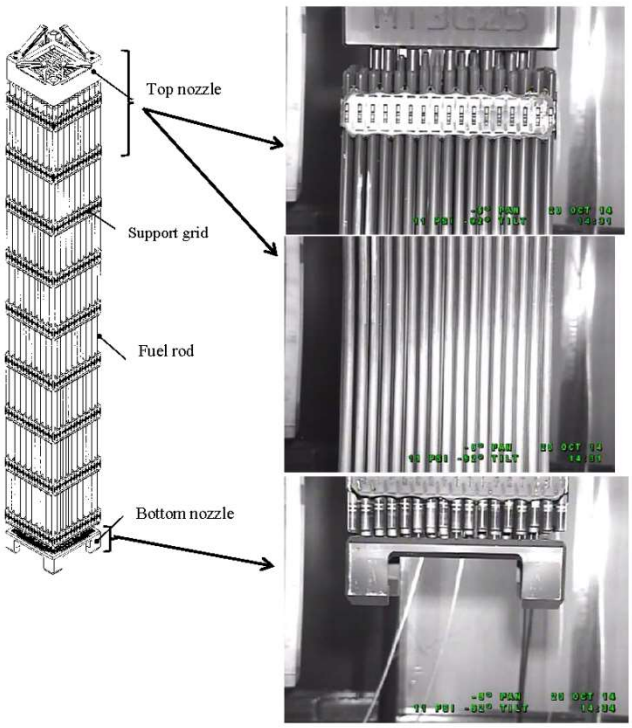


FIG. 9. External surface of 48GWd/t type fuel.

5. SUMMARY

Some Japanese utilities are planning to conduct the demonstration test for up to 60 years to accumulate knowledge and experience on long-term integrity of PWR spent fuel during dry storage.

The design of the container was licensed and the test container was manufactured from fiscal 2012 to 2013. Thermal design of the test container is important in order to simulate temperature of an actual cask. To evaluate a thermal performance of the test container, heat-transfer tests were carried out using the test container.

Fuel temperatures at the installation into the test container were evaluated based on the heat-transfer test results and it was confirmed that the container has satisfying thermal performance for simulating actual cask temperature. Moreover, the visual inspections for 48 GWd/t type fuel have been carried out before loading and its integrity was confirmed.

Now final licensing procedure for the test facility is being performed, and then the demonstration test of 48 GWd/t fuel will start at the middle of fiscal 2015.

ACKNOWLEDGEMENTS

We thank Nuclear Regulation Authority (Formerly Japan Nuclear Energy Safety Organization) for incorporating in the heat-transfer tests.

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THE HIGH BURNUP CONFIRMATORY DATA PROJECT

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Abstract

Plans to design, license, construct and operate a commercial spent nuclear fuel repository in the United States have been delayed. As a result, storage times of spent fuel in licensed storage systems will need to be extended; in some cases, beyond their allowable license extension periods. While it is expected that the spent fuel in licensed storage systems will remain safe and secure for many decades, an effort is underway to provide confirmatory data to validate this understanding. This expectation is based largely on an earlier storage demonstration project for low burnup fuels that took place in the U.S. in the late 1980's (Kessler 2002, McKinnon 1997). In the past several decades however, reactor operations have improved and become much more efficient, resulting in higher fuel burnups before discharge from the reactor. Higher fuel burnups impact discharged fuel component properties (e.g., higher hydrogen contents in the cladding, differences in pellet-clad interactions, higher internal rod pressures, etc.). Therefore, confirmation of fuel performance for extended periods of time for high burnup fuel (e.g., >45 GWd/MTU) is desirable. To address high burnup fuel performance during extended storage, the U.S. Department of Energy (DOE) entered into a contractual relationship with the Electric Power Research Institute (EPRI) and its partners (Dominion Power and AREVA) to stage a demonstration project to generate data and assess real time performance of HBU fuel in an NRC-licensed dry storage cask for 10 years.

1. BACKGROUND

Two independent events led to the realization of the Confirmatory Data Project (CDP). First, improvements in reactor operations resulted in discharge of higher burnup fuels. It was recognized that the positive results from the earlier dry storage demonstration project was limited to spent fuel burnups <35 GWd/MTU. With fuel being routinely discharged at burnups >45 GWd/MTU, a current understanding of spent fuel degradation mechanisms in long term storage needed to be extended to these higher burnup fuels. To help address this gap, the U.S. Nuclear Regulatory Commission (NRC) issued Interim Staff Guidance, ISG-11, that provided a licensee guidance for evaluating high burnup spent fuel cladding characteristics under storage and transportation conditions [1].

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The Electric Power Research Institute (EPRI) then published a report on the feasibility of conducting a high burnup storage demonstration program [2]. This work was conducted under the Nuclear Energy Plant Optimization (NEPO) program and three options for conducting a high burnup dry storage demonstration program;

- Option A – Augment existing examination program;
- Option B – Utility independent spent fuel storage installation (ISFIS) followed by laboratory examination;
- Option C – Laboratory storage and examination.

Evaluation of these options included estimated duration of project, evaluation of potential storage systems, assessment of periodic monitoring and inspections while in storage, post-storage fuel examination, and post-test disposal and clean-up. Time and cost for these options ranged from 5 years and $5M for Option A to 12 years and ~$20M for Options B and C. The conclusions of this report stated that there is a technical need due to the gap in fuel data for spent fuel with burnups >45 GWd/MTU; there is interest from industry, the regulator, and the DOE; and there are resources available.

Second, the cessation of licensing activities in 2009 for the planned Yucca Mountain spent fuel repository immediately cast into question the ability to extend the licensing of existing storage systems. Without a firm date for beginning repository operations, it was reasonable to consider having the need to store spent fuel in their current systems for periods that exceeded their licensing terms. Without confirmatory data for these high burnup fuels in extended storage, it becomes problematic to quantitatively demonstrate fuel integrity for many decades.

In 2009, the DOE initiated an R&D program that was focused on supporting the development of the technical basis to support extended storage and subsequent transportation of high burnup spent fuel. The DOE national laboratory system provides the technical support to this program along with a strong collaborative effort with industry and the NRC. The first task under this program was to identify and prioritize the technical gaps needed to be addressed to build the technical data base that will be used to support future storage extension and transportation licensing activities. A comprehensive assessment of technical gaps and priorities for the DOE program are provided in Refs. [3, 4]. Areas of high importance included hydrogen effects on cladding; thermal and stress profiles; monitoring; stainless steel canister corrosion and stress corrosion cracking; general corrosion of bolts and seals for bolted closure systems; and cask drying after loading of the spent fuel.

Other organizations also provided independent gap assessments, including the Nuclear Waste Technical Review Board [5], the Nuclear Regulatory Commission [6], the Electric Power Research Organization [7], and the International Atomic Energy Agency. In addition, EPRI sponsored a comprehensive international gap assessment through their Extended Storage Collaboration Program (ESCP) [8]. While there are differences in identified gaps and associated priorities in these studies, there is consistency across the reports in identifying high priority issues; in particular, hydrogen effects on cladding and corrosion issues associated with stainless steel canisters.

The NRC subsequently issued draft Interim Staff Guidance, [9], for public comment that provided guidelines to conducting a high burnup dry storage demonstration program. This draft ISG provided guidance for gathering high burnup spent fuel data in extended storage conditions over an extended timeframe.

The work discussed in this section provided the confidence across multiple organizations that conduct of a dry storage demonstration program would provide the necessary confirmatory data to support extended storage licensing applications.

2. THE HIGH BURNUP CONFIRMATORY DATA PROJECT: DETAILS

In April 2013, the DOE entered into a contract with EPRI to conduct the High Burnup Confirmatory Data Project. This project aligns quite closely to Option B of the 2003 EPRI feasibility report [2] on options for conducting a demonstration project for high burnup fuel.

For this project, EPRI is teaming with Dominion Power and Areva (specifically, Areva Federal Services, Areva Transuclear, and Areva Fuels). From the DOE side, technical support is provided by several of the DOE national laboratories.

In general, the objectives of this project are to monitor and inspect high burnup spent fuel that is placed in an NRC-licensed dry storage cask for a period of approximately 10 years. This will provide the confirmatory data needed to support extended licensing considerations. The industry team that was chosen is especially well suited to address these project objectives. Important technical aspects of this project include:

- The high burnup fuel will be selected from the Dominion North Anna spent fuel pool. All the fuel will be PWR and will include a range of high burnups and cooling times, as well as the major cladding types; Zr-4, Zirlo, and M5;
- The cask that will be used is an NRC-licensed TN-32B. Fig. 1 is a picture of a TN-32 cask;
- Thermal lances with integrated thermocouples will be a permanent part of the storage cask. These lances will require a lid penetration and will trace cask internal temperature readings on a continuous basis;
- Gas samples will be taken soon after the cask has been loaded and completely dried as well as three times over the 10 year period that the cask is operating on the storage pad. This sampling will specifically look for any indication of water (e.g., incomplete drying) and fission gasses (e.g., potential fuel rod leaks);
- Individual rods will be pulled from similar (sister) assemblies. These rods will be sent to a DOE laboratory to undergo separate effects testing to determine initial material property and physical conditions of the rods. These initial conditions will then be compared with testing results on the stored fuel after the extended drying period. The transport of the sister rods and subsequent testing are not a part of this contract;
- A license amendment will be prepared for NRC review and approval to cover the lid penetration as well as the specific collection of high burnup fuel assemblies;
- After approximately 10 years, the cask will be shipped to a DOE laboratory to inspect and conduct testing on the spent fuel.

FIG. 1. TN-32 Dry storage cask.

The development of the test plan, selection of fuel assemblies and sister rods, and the identification of monitoring and inspection techniques and processes, are focused on gathering the necessary data discussed in the preceding section to build the technical data base to support extended licensing of dry storage systems in the U.S. This assessment must balance the desire to obtain as much detailed information and data possible to support the science aspect of the project with the operational and licensing constraints inherent in a licensed, operating nuclear facility.

3. THE HIGH BURNUP CONFIRMATORY DATA PROJECT: STATUS

The CDP planning has progressed remarkably well during the two years the contract has been in place. The first important document to be completed is the Final Test Plan [10]. Before being finalized, the draft was issued for public comment. A total of 189 comments were received and were addressed before the final document was issued.

Placement of the thermal lances and resultant location of the thermocouples within the lances and their position relative to the spent fuel and location within the basket was an early activity. This activity was important to locate the thermocouples in strategic positions to optimize the thermal data that will be collected. Seven lances will penetrate the closure lid. Nine thermocouples will be threaded into each lance, providing 63 temperature collection locations. These locations will collect data from spent fuel assemblies of interest, as well as provide a comprehensive data set of temperature characteristics within the basket and internal cavity of the cask as a function of time.

One of the major tasks during this period was the selection of the specific fuel assemblies for the CDP. As stated in the Introduction, the primary parameters were that the spent fuel should be high burnup (i.e., >45 GWd/MTU), all the various PWR cladding types should be included (i.e., Zr-4, low tin Zr-4, Zirlo, and M5), and the cladding temperature during loading, drying, and storage should reach as close as possible the NRC limit of 400°C. This temperature criterion is important to drive as much as the incumbent hydrogen in the cladding back into solution so that when the rod cools and the hydrogen precipitates back into a solid, the maximum possibility of hydride re-orientation will occur.

Initial selection of the fuel was based on these criteria. Following the selection, detailed thermal analyses were conducted to estimate peak cladding temperatures. Initial results were surprising in that, even with high burnup fuel, peak cladding temperatures only reached ~300°C. This is due to the design of the TN-32 and its ability to shed heat very effectively through conduction pathways. After several iterations, the final assembly identification and configuration inside the TN-32B basket was decided. High burnup fuel assemblies with shorter in-pool cooling times were substituted for longer cooled high burnup fuel. This resulted in maximum cladding temperatures of about 340°C [11]. While the maximum cladding temperature is still somewhat lower that the desired 400°C, this was the maximum that could be attained as this configuration resulted in a limiting temperature of 149°C for the neutron shield on the external surface of the cask. The total heat load generated by the spent fuel in the cask is 37 kW.

Recent work that has just been completed is the identification of specific sister rods that will be pulled from assemblies that have like characteristics to the assemblies that will be loaded into the TN-32B. A total of 25 rods were selected to conduct the separate effects tests. This inventory of sister rods is made up of 9 Areva M5 rods, 9 Westinghouse Zirlo rods, 4 Westinghouse low-tin Zr-4 rods, and 3 Westinghouse Zr-4 rods. The Areva M5 rods were

pulled from their assemblies at the North Anna Site in January 2015. The schedule for the Westinghouse rods is to pull them in June 2015.

Under the contract, a Design and Licensing Basis Document (DLBD) has been written to support the up-coming license amendment request (LAR) [11]. These documents are necessary to process the NRC review and approval for amending the current TN-32 storage license. Aspects of the amendment include the lid design to account for the thermal lance penetrations, a new criticality analysis, a new thermal analysis to account for the higher heat loads, and a new radiological analysis to account for the higher burnup fuel.

4. THE HIGH BURNUP CONFIRMATORY DATA PROJECT: NEAR-TERM SCHEDULE

The schedule of recent and up-coming important milestones for the CDP includes [11]:

- December 2014: Completion of the Design Basis Licensing Document
- January 2015: Areva M5 sister rods pulled
- June 2015: Westinghouse low-tin Zr-4, Zr-4, and Zirlo rods pulled
- July 2015: Submit License Amendment Request to the NRC
- First Quarter 2016: Ship sister rods to DOE laboratory
- January 2017: NRC review complete
- March 2017: TN-32B cask delivered to North Anna site
- June 2017: Dry run and functional tests complete
- July 2017: Cask loading complete – begin initial monitoring
- August 2017: Cask moved to the storage pad – begin long term monitoring

The scheduling of this project was also subjected to the need of balancing the desire to accelerate the data collection efforts to support the science with the operational and licensing requirements of the North Anna site. These dates were chosen as earliest opportunities to conduct the site work without conflicting with the normal operational requirements at a nuclear facility.

5. CONCLUSIONS

The CDP is designed to provide the necessary confirmatory data to support extended storage licensing for dry storage systems containing high burnup fuel. The data needed has been identified through a decade's long process that included an early storage demonstration for low burnup fuels, succeeding R&D associated with high burnup fuels, and regulatory guidance developed as a result of the R&D. This portfolio of work provides the confidence that the data collected from the CDP, coupled with the separate effects tests on the sister rods, will confirm our understanding of fuel behavior over long storage periods and justify the technical basis for license extension for storage.

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PERFORMANCE OF ELASTOMETER SEALS IN TRANSPORT AND STORAGE CASKS

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Abstract

Elastomer seals are widely used as barrier seals in containers for low and intermediate level radioactive waste and for spent fuel transportation casks. In addition, they are also used for spent fuel storage and transportation casks (dual purpose casks [DPC]) as auxiliary seals to allow leakage rate measurements of metal barrier seals for demonstration of their proper assembling conditions. Depending on the area of use, the rubber materials have to demonstrate proper sealing performance with regard to mechanical, thermal, and environmental conditions as well as irradiation during the entire operation period. Concerning DPC, degradation effects should be limited in a way that, for example, effects from potentially released decomposition elements may not harm e. g. metal barrier seals. Leakage rate measurements should be possible also after long interim storage periods prior to subsequent transportation. Because of the complex requirements resulting from the various applications of containers for radioactive waste and spent nuclear fuel, BAM has initiated several test programmes for investigating the behaviour of elastomer seals. In this contribution the current status is described and first results are discussed.

1. INTRODUCTION

Concerning the back end of the fuel cycle and the ageing management of spent fuel storage, metal seals are of high importance for dual purpose casks (DPCs) used for transport and storage. But in addition the ageing behaviour of elastomer seals is of interest as they are also used in spent fuel transport casks and as auxiliary seals in DPC to allow leakage rate measurements of metal seals. Additionally elastomer seals are employed as primary seal in containers for low and intermediate level radioactive waste and are therefore contributing to the enclosure of radioactive waste generated from operation and decommissioning of nuclear power reactors.

The performance requirements resulting from application conditions are manifold compared with other application areas as automobile, aviation or process engineering. Depending on the terms of use of the specific container these requirements include:

- Long-term use up to several decades;
- Radiation effects resulting from the inventory;
- Operation as well at elevated temperatures and at possible low temperatures during transport or within the storage facility;
- Mainly under pure static conditions but potentially highly dynamic events in case of accidents.

In many conventional applications where elastomer seals are commonly used due to their easy handling and special material properties a seal exchange at defined intervals is typical [1]. For transport and storage casks this is not possible or at least hard to perform as minimisation principles of radiation exposure prohibits unnecessary cask handling.

Under this prerequisite we started to investigate the behaviour of elastomer seals in detail focussing on several aspects relevant for application in transport and storage casks namely:

- Low temperature behaviour and sealing performance:
  - Under static conditions;
  - After partial loss of compression.
- Ageing of elastomer seals:
  - Under thermos-oxidative long-term conditions;
  - Property changes caused by γ-irradiation.
- Development of a numerical description of seal deformation during operation.

In this contribution the aims of these research areas are described and an overview of actual results is given.

2. LOW TEMPERATURE BEHAVIOUR AND SEALING PERFORMANCE

The low temperature behaviour is especially important during transport of casks as the transport regulations require full performance of the cask even at -40°C. Therefore the seal is also required to be reliable at this temperature. With this motivation an investigation programme was launched which focussed on the dependence of seal performance on temperature and the correlation with material changes occuring at low temperatures. As an extension of these investigations which were focussed on sealing under purely static force conditions some current work is performed concerning the behaviour of seals at low temperatures after partial loss of compression.

It is known that material properties of rubbers are strongly temperature dependent. At low temperatures this is caused by the glass-rubber transition (abbr. glass transition). During continuous cooling, the material changes from rubber-like entropy-elastic to stiff energy-elastic behaviour, that allows nearly no strain or retraction due to the glass transition. Hence rubbers are normally used above their glass transition but the minimum operational temperature limit is not defined precisely.

For the investigations, fluorocarbon (FKM) and ethylene-propylene-diene (EPDM) rubbers were selected as they are often used in radioactive waste containers. The materials were purchased from a commercial seal producer.

We started to investigate the behaviour of seals respectively seal material at low temperatures with regard to potential leak-tightness changes [2, 3, 4]. In the course of this investigation also a new method for characterization of elastomer seal materials has been developed [5, 6] that emulates the compression set measurement by using dynamic mechanic analysis (DMA). It can be performed semi-automatically and much faster than standardized procedures according to ISO 815-2. The measurement of time-dependent recovery of a compressed elastomer at low temperatures expressed as compression set with a DMA worked very well. A comparison of different materials is shown in Fig. 1.

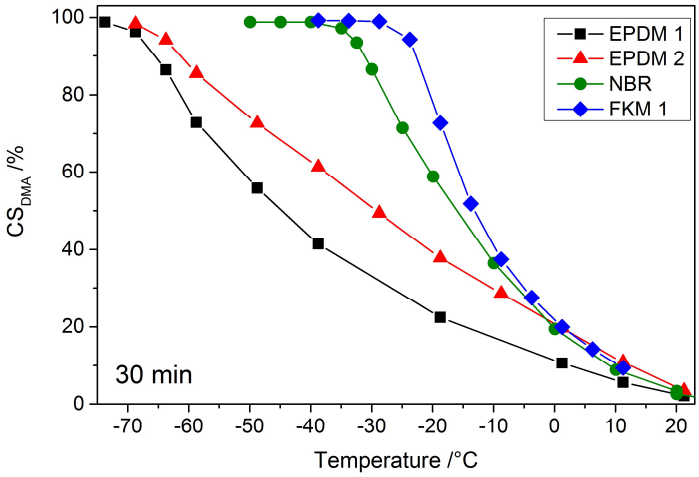

FIG. 1. Temperature dependent compression set of different materials measured 30 min after sample release [6].

The compression set measurement proved to be able to give valuable information to differentiate the material behaviour of various compounds.

An example of component tests under static test conditions is given in Fig. 2. The strong increase in leakage rate occurs at a considerably lower temperature compared with glass transition temperatures from differential scanning calorimetry (DSC) or DMA measurements.

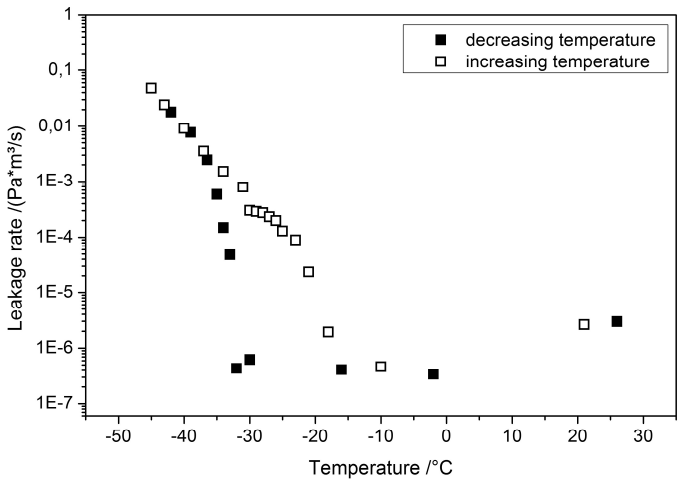

FIG. 2. Leakage rate values of an FKM material over temperature [7]. Measurements performed on one seal in two parts: first part decreasing temperature steps, second part increasing temperature steps.

The temperature at which the sealing performance of the seal during heating reaches a value comparable to the 'good' sealing performance at room temperature is easily explained by the necessity of molecular mobility, associated with a temperature above the glass-rubber transition region. The much lower failure temperature of the seal during cooling led to further questions but could be explained by consideration of the thermal expansion behaviour of seal and surrounding flange. The results help to characterize the performance of a seal

manufactured from these materials. A seal can only work properly at low temperatures and in case of a dynamic event if the seal material is able to react fast enough to changes of the dimensions of the sealing system. If the reaction of the material is slower than the changes in the dimensions of the sealing groove a leakage path is formed. This path could be closed again by a sufficient (but time- and temperature-dependent) amount of recovery of the material.

Starting from the results of the purely static leakage rate measurements shown in Figure 2 the question arised how a seal performs at low temperatures when deformations of the sealing system takes place caused e.g. by accidents. To investigate the seal behaviour beyond purely static force conditions a measurement setup was developed allowing a small but quick partial release of the compressed seal at low temperatures. Here, a small release means that the seal compression is reduced from initially 25% to about 23% and a quick release is a change in the compression within a timeframe of one second.

First results [8] show that such a small deformation of the sealing system at -20°C is already sufficient to cause an enormous leakage rate increase for a material similar to the FKM used for the measurements shown in Fig. 2. The temperature of -20°C was chosen as it is at the low temperature limit of the glass-rubber transition process observed by DMA measurements. After retightening of the sealing system and a new evacuation an increased leakage rate remains. Further investigations to quantify this behaviour are running at the moment.

3. AGEING OF ELASTOMER SEALS

Besides examining the low-temperature behaviour of elastomer seals, it is our goal to evaluate the service lifetime of the seals with regard to the requirements for long-term safety of containers.

Ageing describes a multitude of different processes which lead to deterioration of material properties. With ageing, elastomers can gradually lose their elasticity and their ability for recovery, which might result in a leakage above the tolerable level. It is important to know the rate of degradation and to determine which property can be used as a practical and easily measurable end-of-lifetime criterion. For this reason, we started an ageing programme with selected rubbers for monitoring the change of properties. In one sub-project the focus is put on the seal behaviour over long time periods under thermooxidative conditions, in another sub-project the influence of γ-irradiation on material properties is investigated.

3.1. Long-term influence of thermooxidative conditions

In this part of our ageing programme EPDM and FKM seals are oven-aged at four different temperatures (75°C, 100°C, 125°C, and 150°C) in order to extrapolate the data to service temperature using a suitable model. They are examined after 1 d, 3 d, 10 d, 30 d, 100 d, 0.5 a, 1 a, 1.5 a, 2 a, 2.5 a, 3 a, 3.67 a, 4.33 a, and 5 a. In order to be able to compare between compressed and relaxed rubber, the samples are aged in their undeformed O-ring state as well as compressed between plates with a deformation of about 25% corresponding to the actual compression during service. Furthermore, samples are aged in flanges that allow leakage rate measurements. Schematic representations are given in Fig. 3.

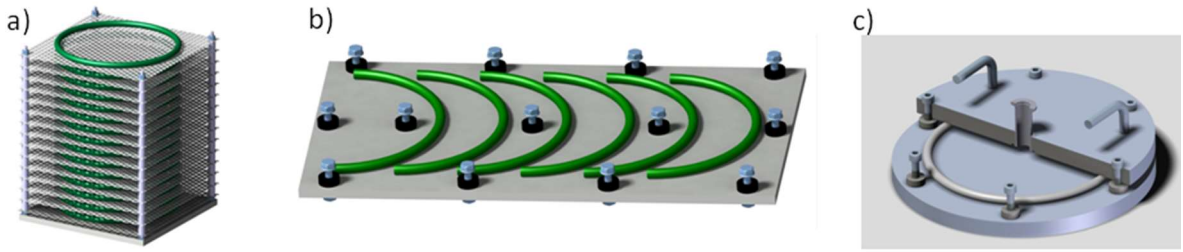

FIG. 3. Different ageing conditions: a) uncompressed in metal rack, b) compressed by 25% between plates; c) compressed by 25% in metal flanges for component tests.

Used methods for investigation include hardness as a quick indicator, compression set and compression stress relaxation as measures reflecting the actual properties of a compressed seal. Additionally, we are applying classical polymer analysis methods like dynamic mechanic analysis and thermogravimetric analysis which show changes in the polymeric structure due to chain scission and crosslinking. Furthermore, we are testing the leakage rate of the O-rings in order to correlate the changes in physical and chemical properties to the actual leak-tightness of the seal.

The ageing programme was started in May 2014. As an example pictures of aged seals and compression set values from samples stored for 100 d between metal plates with 25% compression are shown in Fig. 4. In contrast to ISO 815-1 the values are calculated from the O-ring height instead of the height of a standard sample. The measurements were performed several days after seal release at room temperature.

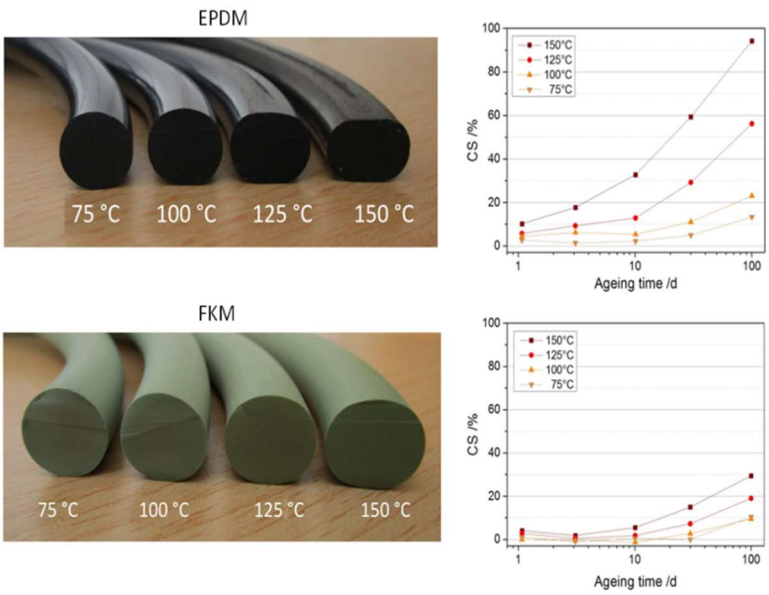

FIG. 4. Pictures of aged seals and compression set values of samples stored with 25% compression for up to 100 d.

Both materials show an increase of residual deformation with storage temperature and storage time. EPDM shows a distinct deformed cross section at 125°C and 150°C. At 150°C the

compression set values are greater than 90% and therefore the resilience is rather low. FKM shows much lower compression set values which indicate higher durability. Further results are described in Refs. [9, 10].

The aims of the investigations are used to increase the understanding of occuring ageing processes, test existing lifetime estimation methods such as the Arrhenius approach and verify their applicability. Furthermore, appropriate methods shall be chosen to ensure the long-term performance respectively determine the end of lifetime of elastomer seals used in transport and storage casks.

3.2. Property changes caused by γ-irradiation

FKM seals and plate material were irradiated with γ-radiation doses of 50 kGy, 100 kGy, 200 kGy, 400 kGy, and 600 kGy. The samples were analysed with standard methods as e.g. hardness and compression set measurements. Additionally, the glass-rubber transition behaviour was tested by DMA. The results are described in detail in Refs. [11, 12].

The irradiation of fluorocarbon rubbers leads to substantial changes in the material properties, especially on the glass-rubber transition. Due to predominant cross-linking reactions, the glass-rubber transition is shifted to higher temperatures [13]. This effect is not detectable by a standard hardness measurement, but can be shown clearly by thermoanalytical methods.

The height recovery of a sample after compression is emulated by the compression set test, which is therefore an important value for sealing applications. An example of the influence of irradiation on the recovery behaviour at -15°C is shown in Fig. 5.

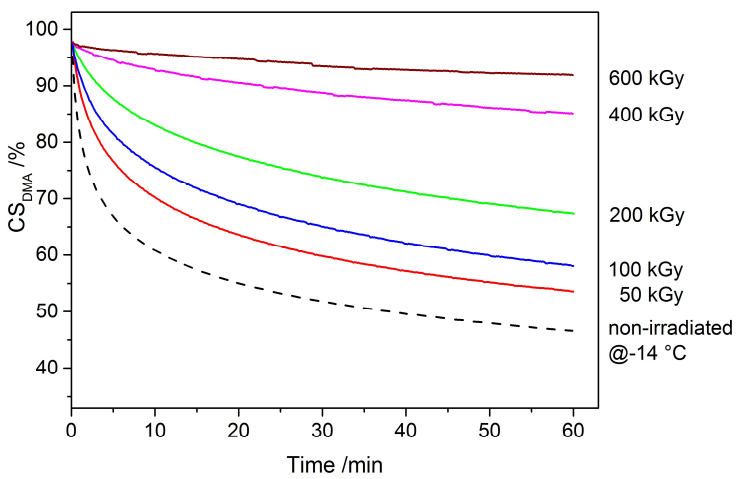

FIG. 5. Compression set data for FKM with varying γ-irradiation dose measured at -15°C over time [11]

The compression set values at -15°C show a strong increase with increasing irradiation dose. At low temperatures the γ-irradiation caused higher compression set values with higher doses. The behaviour at room temperature was not as strongly affected. An explanation could be that the additional cross-links that were induced in the material in an undeformed condition also increase the restoring force of the sample. In the case of continuous irradiation during a seal's

service, additional cross-linking would stabilize the deformed condition and therefore hinder recovery of the initial geometry.

4. DEVELOPMENT OF A NUMERICAL DESCRIPTION OF THE SEAL DEFORMATION DURING OPERATION

Beside the experimental investigation of material properties and component behaviour, numerical analysis is an important tool to simulate thermo-mechanical seal behaviour. Especially for safety evaluations of containers where the safety case is based on numerical calculations instead of experiments performed with prototypes, a sufficient description of the seal is necessary. The finite element method is preferred and hence an adequate finite element model is required. Due to inherent polymer properties, elastomers show non-linear stress-strain behaviour with a pronounced strain rate and temperature dependence. For technical elastomers a dominantly viscoelastic behaviour is combined with additional effects which derive from the incorporated filler particles interacting with the polymer molecules and with each other [13].

These prerequisites of the complex material behaviour require a sophisticated material model to allow a sufficient description. Various material models are available in typical finite element analysis software packages such as ABAQUS. Some of these models are phenomenological and others are based on physical approaches.

In a first step results of basic compression and tensile tests were used to identify parameters for hyperelastic material models already implemented in the computer code to describe the hyperelastic behaviour of the investigated rubberlike materials. Therefore experiments with different strain rates and additionally stress relaxation experiments where performed. Fig. 6 shows the stress distribution in a compression test specimen in z-direction. The exemplary results were generated using the Ogden strain energy potential. A friction coefficient of 0.1 and a Poisson Ratio of 0.495 were assumed.

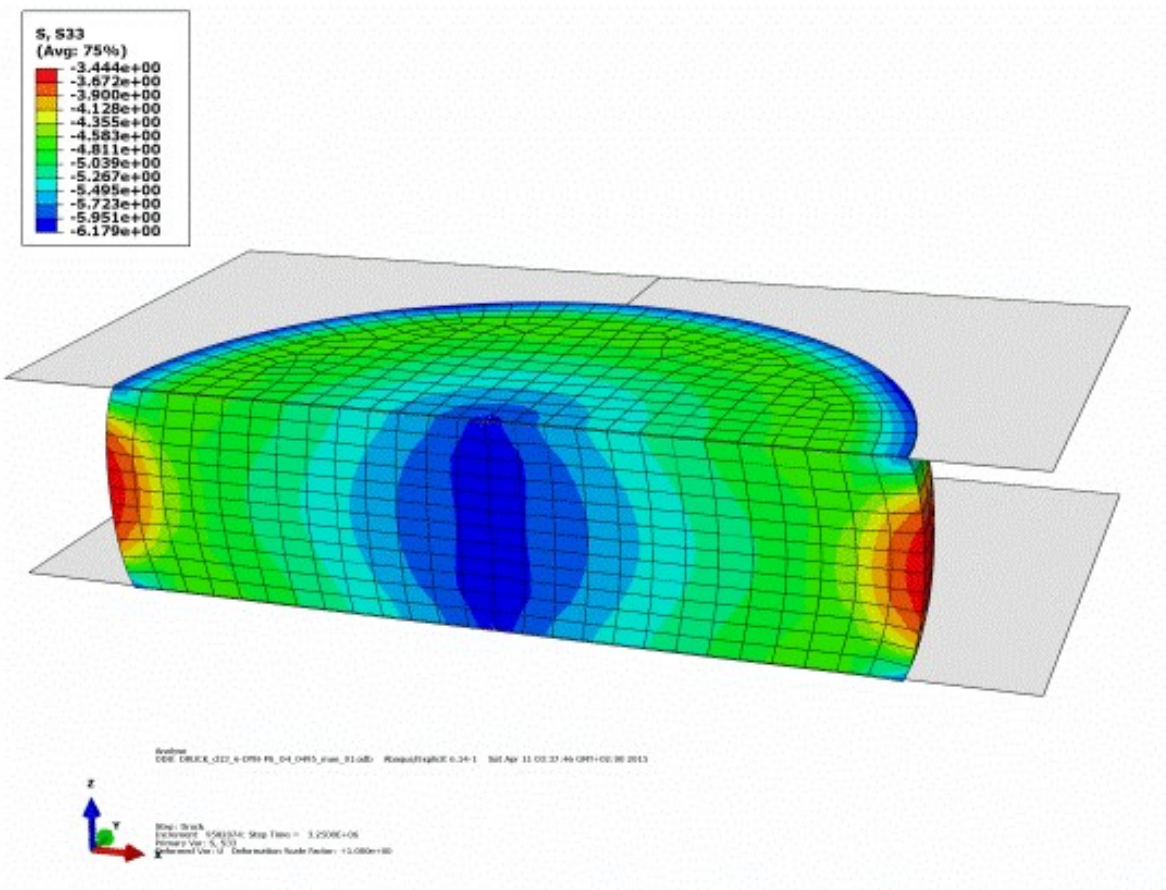

FIG. 6. Stress distribution in z-direction in a 25% compressed specimen.

The now available models are able to describe simple tension and compression tests sufficiently. Further model development and verification are planned using the model also for the description of more complex tests as e.g. compression set and finally whole components.

5. SUMMARY

The use of elastomer seals in casks for radioactive waste can be a suitable way of enclosure but several aspects of this application require an indepth investigation. Therefore at BAM several projects are running to increase the knowledge about material properties and performance of elastomer seals at low temperature, during long-term use and under γ-irradiation. Additionally the numerical description of containment systems becomes increasingly important for the safety case. Therefore a material model of the used seal material is necessary to describe the mechanical seal behaviour correctly. The shown investigations are focussed on representative materials and try to increase the general understanding of the behaviour of elastomer seals in casks. They are not intended to replace the necessary investigations for individual safety evaluations.

ACKNOWLEDGEMENTS

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MECHANICAL PROPERTIES OF ALUMINIUM ALLOYS FOR TRANSPORT AND STORAGE CASK AFTER LONG TERM STORAGE

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Abstract

In recent dry transport and storage dual-purpose casks for spent nuclear fuel, various kinds of aluminium alloy are widely used for basket material as structural members. Basket for dual-purpose cask is designed to maintain prescribed geometrical arrangements of spent fuel assemblies under accidents during transport and storage condition, and also designed to have thermal performance for removal of decay heat from spent fuel assemblies. Aluminium alloys, which have a high heat-transfer performance and material workability as well as a high specific strength, are suitable for basket material. Because cask may be transported to a reprocessing plant after long term storage at an interim storage plant, it is important to maintain integrity of basket structure under transport accidents after long term storage. However, mechanical properties of these alloys are possibly changed by thermal ageing during storage since cask service temperature may affect microstructure of aluminium alloys over the long time period. It is therefore essential in cask design qualification to identify degradation of mechanical strength of basket material both for normal storage and transport accident after long term storage. MHI has conducted comprehensive evaluation of mechanical properties considering thermal ageing of the aluminium alloys A6N01 containing boron carbide (B_4C) and commercial A3004 without boron for the basket, which resulted in construction of a data base for assessing structural integrity of the basket [1, 2, 3]. In this paper, effects of thermal ageing on mechanical properties of these materials are presented based on the data, and the microstructural strengthening mechanism which play important roles in the evolution of mechanical properties are discussed.

1. INTRODUCTION

Spent fuel transport / storage casks are one of the components necessary for the nuclear fuel cycle. The casks are required to undergo safety assessment with the assumption that they will be transported from a plant to an interim storage facility, stored for 60 years at the longest, and finally transported to a reprocessing facility or a final disposal site after the storage.

The safety functions of the cask are subcriticality, shielding, heat removal and sealing, and the cask is required to have a structural strength necessary to achieve these functions. Its basket is a member that is mainly required to have subcriticality and heat removal functions, and have structural strength that ensures them. For the subcriticality, heat removal and structure functions. This paper covers A6N01 containing B_4C, and A3004. A6N01 containing B_4C is a material that has all the three functions: subcriticality, heat removal and structure, while A3004 can be used as a material that has two functions: heat removal and structure.

2. BASIS OF EVALUATION OF BASKET MATERIALS

2.1. Thermal and mechanical load on the cask

Casks are used for transport of spent fuels to a storage facility inside or outside the site of a nuclear power plant after being loaded with them at the plant. After the spent fuel transport, the cask is provided with a required treatment at the storage facility, and serves for storage for 60 years at the longest. After the storage, it serves for transport to a spent fuel reprocessing facility or a final disposal site. Fig. 1 shows an overview of typical structure of the MSF-type

cask. The basket in the MSF-type cask is assembly of hollow tubes made of aluminium alloys.

The cask is required to have a safety function as a Type B package, and maintain its safety function in an environment intended for storage for 60 years at the longest. Fig. 2 shows a schematic diagram of the cask's thermal and mechanical load in its operating environment. Thermal loading on cask is represented by a gradual temperature decrease from approximately 200°C to approximately 100°C lasting for 60 years in terms of heat.

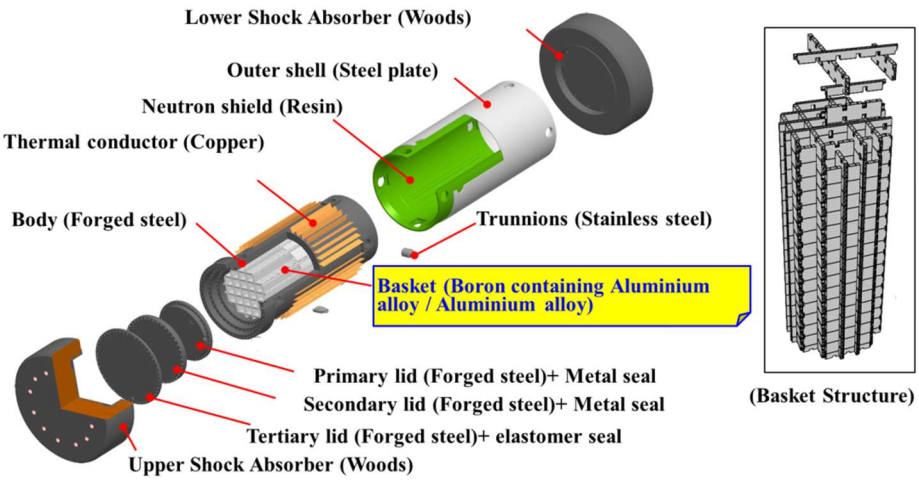

FIG. 1. Schematic illustration for structure of cask.

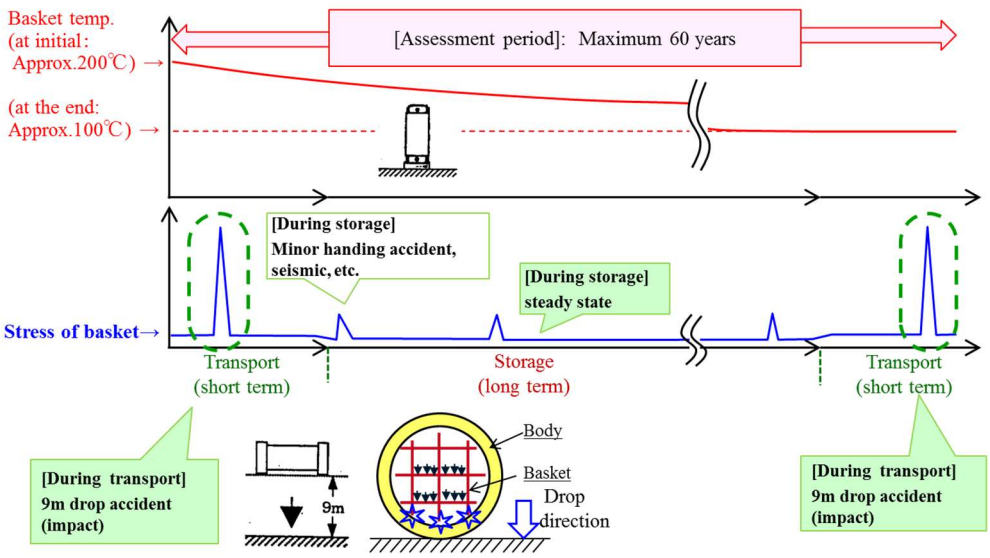

FIG. 2. Schematic illustration for history of thermal and mechanical load on cask.

As for mechanical loading, there is very low level of stress in the basket under normal operating condition of the casks during storage. However, an assumed largest mechanical load

is that for accident conditions of transport (drop of cask from 9 meters high) during transport and post-storage transport in terms of structure. The basket is required to have mechanical properties enough for keeping structural integrity against this stress level, even after thermal deterioration due to the long-term exposure at approximately 200°C.

There are also functional requirements such as sub-criticality and heat removal characteristic. However, this paper focuses on the thermal impact on mechanical properties.

2.2. Basket materials

The following two materials are used for baskets of the MSF-type cask:

(1) A6N01 material containing B_4C: This is a composite material consists of particles of B_4C and matrix base metal of A6N01, an Al-Mg-Si alloy designated in Japanese Industrial Standards (JIS). This is produced through powder metallurgy process and finally shaped in tube through hot extrusion process (hereafter called 'BC-A6N01');
(2) A3004: This is a commercial Al-Mn alloy designated as JIS standard A3004 produced by hot extrusion (hereafter called 'A3004').

It should be noted that none of the materials have been heat treated after hot extrusion. This state is referred as 'as extruded'.

3. THERMAL DETERIORATION OF MECHANICAL PROPERTIES

3.1. Strengthening mechanisms

Generally, metallurgical factors which contribute to mechanical strength (strengthening mechanisms) of aluminium alloys are: (1) solid solution strengthening; (2) precipitation strengthening; (3) strain hardening; and (4) strengthening from grain boundaries. All of these strengthening mechanisms are effective through decreasing mobility of dislocations in a crystal in its each way, which leads to enhance necessary stress to cause yielding and slip of crystals. These mechanisms are basically the same for all metallic materials including ferrous materials. However, some special considerations should be done for those in aluminium alloys for basket, because time-dependent changes in the strengthening mechanisms have to be taken into account even at operation temperature explained in the section 2. This difference between aluminium alloys and ferrous materials essentially comes from the difference in thermal stability of the matrices, for example, as seen in difference in their melting points (1811 K for iron, 933 K for aluminium). In this paper, the change in mechanical properties of basket materials in above context is referred as 'thermal deterioration'. Therefore, for the reasonable design and safety assessment of casks, it is necessary to assess the material properties taking the thermal deterioration brought by the cask operating environment into account.

Followings are general explanation of each strengthening mechanism together with its specific contribution in the two materials for basket.

(1) Solid solution strengthening:

The introduction of solute atoms into solid solution in the solvent matrix invariably produces an alloy which has higher mechanical strength than pure metal. This is achieved by that fact that strain field created around the solute atom by difference in atomic size between that in

matrix decreases the mobility of dislocation. Magnitude of the solution strengthening depends on concentration and atomic size of the solute.

Typical aluminium alloy strengthened by this mechanism is Al-Mg alloys. In the case of basket materials BC-6N01, there is solid solution strengthening by solute elements like Mg and Si, but the contribution is very small because contents of these alloying elements are as low as about 1 mass% in total. This is similar for A3004, and solubility of its main alloying element Mn in aluminium matrix is practically zero in the equilibrium state.

(2) Precipitation strengthening:

In this mechanism, the dislocation mobility is reduced by fine precipitates or strain field created by the precipitates. In the aluminium alloys, this is achieved by solution heat treatment followed by quenching to generate supersaturated solid solution, and then ageing at room temperature to about 200℃. Through this process, solute atom clusters or fine metastable precipitates appear. It should be noted that with excess temperature or time of the ageing, the precipitates grows and its number density decreases. This causes loss of precipitation strengthening and called as “over aging”. Among the commercial aluminium alloys, Al-Cu, Al-Cu-Mg, Al-Mg-Si, Al-Zn and Al-Li alloys have significant hardenability by precipitation strengthening.

BC-A6N01 does have precipitation hardenability by Al-Mg-Si precipitates. However, for the application for the cask basket, no intentional heat treatments for the precipitation hardening are applied during it manufacturing process. Nevertheless, it should be remembered that there would be small precipitation strengthening due to slight supersaturation during cooling from hot extrusion, in spite of lack of intentional heat treatments. In any cases, the precipitation strengthening cannot be expected as the principal strengthening mechanism at the final stage of storage, because the precipitates grows to lose its contribution to strengthening.

In hot extruded A3004, there are particles of Al-Mn second phases dispersed in the matrix. But these second phases are not considered as strengthening factors because of its large size and low number density [4]. Although there are some reports of precipitation hardening of A3004 caused by Al-Cu-Mg precipitates [5] A3004 generally does not have significant precipitation hardenability.

(3) Strain hardening:

Strain hardening occurs in metallic material by introducing cold working (plastic deformation at relatively low temperature, typically at ambient temperature) in it. Any cold works lead to increment in dislocation density, which causes interaction between dislocations and therefore impede migration of dislocations.

Industrially applied cold working is cold rolling, cold forging, cold forming and tensile deformation, but no intentional strain hardening is introduced in the two basket materials. However, small degree of work hardening may be introduced during it manufacturing process. Strain hardening disappears by exposure to elevated temperature, because of migration of dislocation and extinction accelerated by thermal diffusion (recovery). Strain hardening is assumed to be far from dominating factor in the mechanical strength of the basket materials, however, it has to be taken into account as a potential factor in the thermal deterioration.

(4) Strengthening from grain boundaries:

In this strength mechanism, grain boundaries are barriers to migration of the dislocations. At the temperature level where recovery of dislocation structure occurs, interaction between dislocations creates new grain boundaries or causes grain boundary migration. This comes to decrease of strength from grain boundaries.

BC-A6N01 consists of fine crystal grains created through the powder metallurgy process, and its coarsening may decrease mechanical strength. Al-Mn alloys including A3004, Al-Mn have high resistance to grain coarsening since the particles of Al-Mn second phases suppresses grain growth. Hence the two basket materials are considered to have some extent of strength from grain boundaries, and it should be evaluated as one of potential factor in the thermal deterioration.

3.2. Material property evaluation for thermal deterioration

It is necessary to evaluate the properties of a thermally deteriorated material in the light of the cask operating environment described in 2.1 and material characteristics of the basket materials described in 3.1. The following shows the method for reproducing the thermally deteriorated material.

3.2.1. Sample materials

The following two materials were used as sample materials:

(1) BC-A6N01: Hot extrusion product of BC-A6N01 was selected as sample materials. The extrusion profile is hollow tube with thickness of 5.5 mm and outer dimension of 150 mm × 150 mm for a cross section vertical to extrusion direction;
(2) A3004: Hot extrusion product of A3004 was selected as sample materials. This is also hollow tube profile, and its thickness is 16 mm, outer dimension is 60 mm × 200 mm for a cross section vertical to extrusion direction.

None of the materials have been heat treated after hot extrusion and before the heat treatments to be described in 3.2.2. The chemical compositions of the two sample materials are described in Table 1.

TABLE 1. CHEMICAL COMPOSITIONS OF SAMPLE MATERIALS (MASS%)

	Si	Fe	Cu	Mn	Mg	Cr	Zn	Ti	Al	B_4C
BC-A6N01	0.7	0.16	0.02	0.03	0.5	<0.01	<0.01	<0.01	Bal.	5.0
A3004	0.20	0.5	0.19	1.1	1.0	0.01	0.05	0.02	Bal.	-

3.2.2. Heat treatment process for evaluation of thermal deterioration

In this study, two kinds of approach were taken to obtain a state of thermal deterioration which is reasonable as a basis of defining mechanical strength for design. One is long-term

overaging heat treatment. This is proposed to simulate thermal deterioration state at 60 years of usage of basket, by small temperature acceleration of the thermal deterioration.

The other approach is application of slow cooling from high-temperature annealing. The aim of this process is complete removal of unequilibrium excess solid solution strengthening and precipitation strengthening.

The details of the two heat treatment processes are given here.

(1) Long-term over aging heat treatment:

The difficulty in this method is validation of appropriateness of heat treatment conditions. As a method for estimating necessary temperature and time for a heat treatment which cause thermal deterioration equivalent to that for 60 years of usage at basket temperature, temperature acceleration based on extrapolation parameters like Larson-Miller parameter [6] assuming the similar thermal activation processes for thermal deterioration as those in creep, has been proposed. However, in contrast to the creep, this method is not necessarily widely accepted as an extrapolation method for changes in mechanical properties caused by thermal ageing due to lack of fundamental data.

At the risk of this difficulty, the long-term aging conditions were determined, so that the conditions are equivalent to thermal exposure far above 60 years at 200ºC using the extrapolating parameters. It is necessary to study the treatment temperature and the treatment time according to the material, and the required constants for the extrapolating parameters are defined by creep rupture tests result of the same materials.

(2) Slow cooling from high-temperature annealing:

This method is proposed for full removal of precipitation strengthening and unequilibrium excess solid solution strengthening by holding at high temperature, and following cooling with cooling rate small enough to cause any supersaturated solution. It is important to determine holding temperature, and cooling rate enough to achieve above quasi-static condition.

For the two materials addressed in this study, holding temperature was selected as 520ºC. In a preliminary study, it was confirmed that this is the highest temperature which does not significantly affect grain size. As for the cooling rate necessary for removal of unequilibrium excess solid solution strengthening, each material has its own sensitivity to cooling rate depending on the diffusion rate of solute atoms in the matrix, and it should be determined by experimental approach.

Table 2 shows the heat treatment conditions for the sample material.

TABLE 2. HEAT TREATMENT CONDITIONS

Material		BC-A6N01			A3004		
Type of heat treatment / Parameter		Temperature (ºC)	Hold time (hr)	Cooling rate (ºC/hr)	Temperature (ºC)	Hold time (hr)	Cooling rate (ºC/hr)
Long-term overaging	LA1	200	20 000				
	LA2			Furnace cooling	250	10 000	Furnace cooling
	LA3	300	20 000		300	10 000	
Slow cooling	SC1	520	10	>600 (air cooling)	520	10	>600 (air cooling)
	SC2						10
	SC3						5
	SC4			2			2
	SC5			1			1
	SC6			0.5			0.5

3.2.3. Characterization of materials

The material that has been provided with long-term overaging treatment and slow cooling from high-temperature annealing is evaluated with the following methods.

(1) Electrical conductivity measurement:

Electrical conductivity of a metallic material decreases by existence of any metallurgical factors which can reduce periodicity of crystals. Hence solute atoms, dislocations or precipitates decrease electrical conductivity. For this reason, electrical conductivity is widely used as index of solute atom concentration for aluminium alloys with no significant strain hardening and precipitation hardening. Therefore, in this study, electrical conductivity was measured at various condition of thermal history of materials to know the degree of solid solution strengthening.

Electrical conductivity was measure by eddy current method on polished surface of samples at room temperature. The unit of electrical conductivity here is %IASC, the International Annealed Copper Standard. This is an expression of electrical conductivity as a relative value to that of annealed commercial pure copper in percent (%).

(2) Tensile test:

Tensile tests were carried out for evaluating mechanical properties after various heat treatment process using round bar specimen with diameter of 4 mm and gauge length of 20 mm. Specimens are cut from the materials so that the tensile direction is parallel to extrusion direction.

3.2.4. Result and discussion BC-A6N01

Fig. 3 and Fig. 4 show effect of cooling rate after high temperature annealing on tensile strength and electrical conductivity, respectively. Tensile strength decreases with decreasing cooling rate, but with the cooling rate smaller than 2ºC/hr, the tensile strength is stable. Similarly, electrical conductivity does not change with change in cooling rate below 2ºC/hr (conditions SC4, SC5 and SC6).

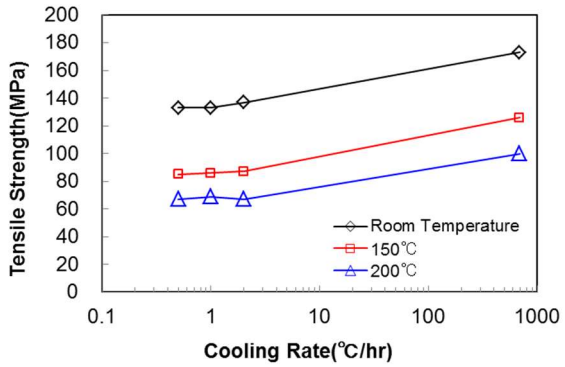

FIG. 3. Effect of cooling rate after high temperature annealing on tensile strength of BC-A6N01.

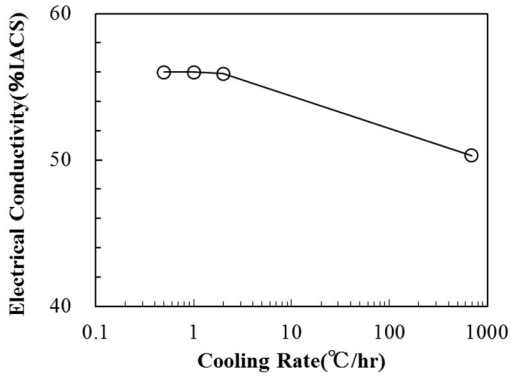

FIG. 4. Effect of cooling rate after high temperature annealing on electrical conductivity of BC-A6N01.

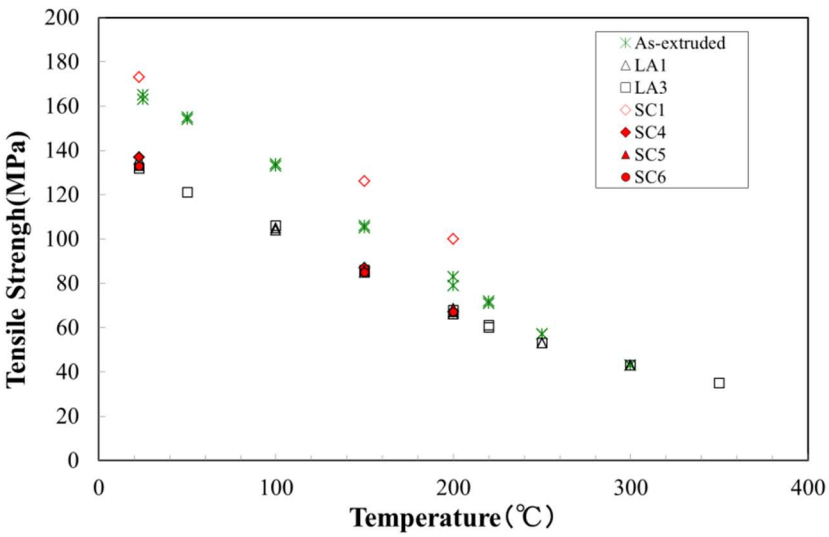

FIG. 5. Temperature dependence of tensile strength of BC-A6N01heat treated at various conditions.

These facts clearly tell that, after solution treatment at 520ºC, the cooling rate below 2ºC/hr is quasi-static enough to realize an equilibrium strengthening state without solid solution hardening by excluding solute atoms during cooling, even if there was supersaturated solid solution before the cooling.

Fig. 5 shows comparison of tensile strength of the long-term aged materials (LA1 and LA2) and high temperature annealed materials. The tensile strength of the long-term aged materials are in good agreement with that of high temperature annealed material with cooling rate below 2ºC/hr. For this reason, it is suggested that the long-term aged material and the high temperature annealed material have same magnitude of solid solution strengthening, which is considered to be very close to equilibrium state.

(1) A3004:

Fig. 6 shows effect of cooling rate after high temperature annealing on tensile strength of A3004. Similarly, Fig. 7 is effect of cooling rate on electrical conductivity and Fig. 8 the change in tensile strength in each treatment. Tensile strength decreases and electrical conductivity increases with decreasing cooling rate. However, in contrast to the behavior of BC-6N01, the changes in tensile strength and electrical conductivity do not seem to become saturated even at a cooling rate as low as 0.5°C/hr.

Considering the fact that in annealed Al-Mn alloy there are neither fine precipitates nor high density of dislocations both of which are effective to decrease electrical conductivity, the above change in electrical resistivity can be attributed to change in concentration of solute elements in the solid solution. More specifically, it suggests that there exist solid solution which cannot be phase decomposed during cooling after being formed at 520°C. Mn in aluminium matrix is practically zero in the equilibrium state. However, it is known that Mn is relatively easily supersaturated into aluminum matrix depending its thermal history, for example, rapid cooling from molten metal [7]. This unequilibrium excess Mn causes significant solid solution strengthening. Furthermore, it is also reported that once formed, the Mn-supersaturated solid solution does not easily decompose by heat treatments. Although no special heat treatment for Mn-supersaturation is applied to A3004 as a basket material, the slight effect of it cannot be completely denied according to the result, and further careful investigated will be required.

Although the high temperature annealed material with cooling rate down to 0.5°C/hr does not reach a stable state, the values of strength property and electrical conductivity in the case with the cooling rate of 0.5°C/hr of cooling from high-temperature annealing became almost equivalent to the values of the long-term overaged material (LA3). The two methods for obtaining thermal deterioration are comparative from the viewpoint of removing solid solution strengthening.

As these results indicate, different aluminium alloys requires different heat treatment conditions for achieving appropriate stable state for evaluating mechanical properties after thermal deterioration, since sensitivity of change in each strengthening mechanism to thermal history differs according to its alloying elements. The evaluation of materials for cask requires further studies ranging from microstructural mechanistic consideration to data collection of macroscopic mechanical behaviors.

As a result of assessing these heat treatment conditions and the materials, it seems necessary to provide an appropriate treatment for each material, since the sensitivity under the heat treatment conditions varies with the materials.

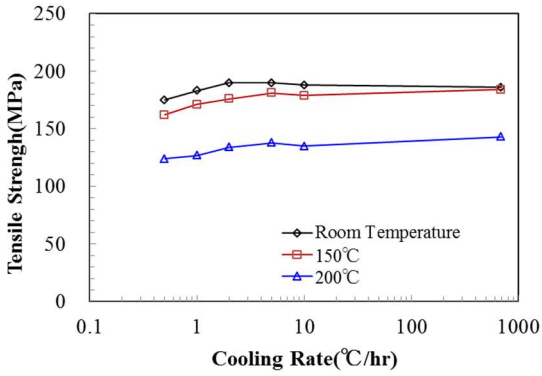

FIG. 6 Effect of cooling rate after high temperature annealing on tensile strength of A3004.

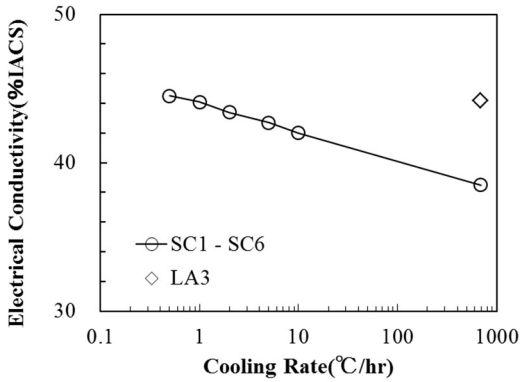

FIG. 7 Effect on cooling rate after high temperature annealing on electrical conductivity of A3004.

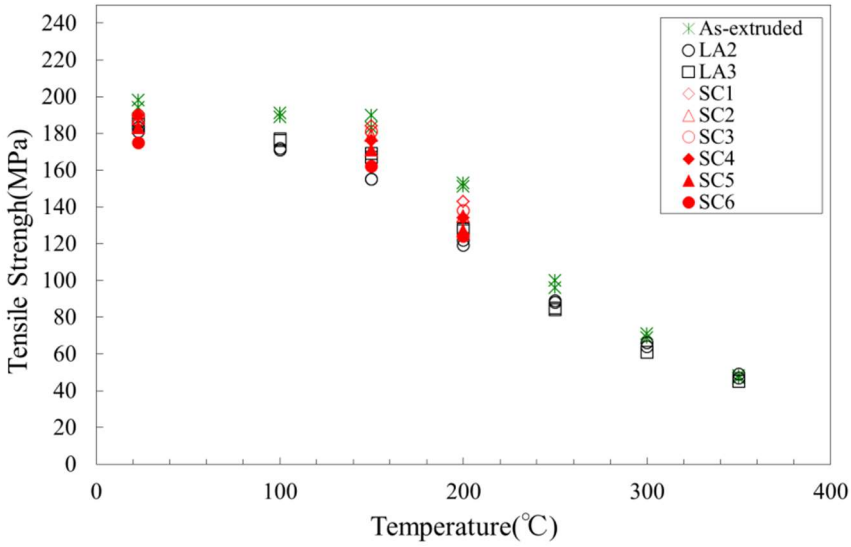

FIG. 8. Temperature dependence of tensile strength of A3004 heat treated at various conditions.

4. CONCLUSION

Two kinds of approach to assess a material thermally deteriorated after 60 years were proposed. The result is summarized as follows:

(1) Tensile strength of BC-6N01 and A3004 aluminium alloy decreases when the materials had long-term ageing or annealing followed by cooling at various cooling rates, although they are not intentionally precipitation strengthened nor heat treated for solid solution strengthened by supersaturation in the manufacturing process;

(2) Long-term ageing at 200°C and 300°C for 20 000 hr and slow cooling from 520°C at cooling rate of 2°C /hr or smaller both make BC-6N01 stable state of thermal deterioration;

(3) Tensile strength of A3004 aluminium alloy become similar value by long-term ageing at 250°C and 300°C for 10 000 hr and slow cooling from 520°C at cooling rate of 0.5°C/hr.

The result also suggests that it is necessary to study methods for obtaining stable state for evaluation of mechanical properties thermal deterioration individually for each material with consideration of its strengthening mechanisms.

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A UNIFIED SPENT NUCLEAR FUEL DATABASE AND ANALYSIS SYSTEM

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Abstract

A fully integrated spent nuclear fuel (SNF) management system involves managing SNF from the time it is discharged from a reactor to the time it is disposed of in a geologic repository. To support the US Department of Energy Office of Nuclear Energy (DOE-NE) Nuclear Fuels Storage and Transportation Planning Project (NFST) activities to lay the groundwork for implementing interim storage, including associated transportation (per the Administration's Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste), an integrated data and analysis tool has been developed. The tool is called the Used Nuclear Fuel-Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS). UNF-ST&DARDS provides a controlled source of technical data integrated with key analysis capabilities to characterize the inputs to the overall US waste management system from reactor power production through ultimate disposition. This system is a new capability / resource that enables automated assembly-specific and cask-specific evaluations for assessing issues and uncertainties related to the extended storage and transportability of loaded canisters, supporting safety confidence and research and development prioritization, and providing a foundational data and analysis capability resource for the future. This paper provides an overview of the UNF-ST&DARDS architecture and automated analysis capabilities.

This paper has been removed from the proceedings due to significant textual overlap. Interested readers should refer to the following references:

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THE CRP ON DEMONSTRATING PERFORMANCE OF SPENT FUEL AND RELATED STORAGE SYSTEM COMPONENTS DURING VERY LONG TERM STORAGE (CRP T13014)

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Abstract

The purpose of the Coordinated Research Project (CRP) on Demonstrating Performance of Spent Fuel and Related Storage System Components during Very Long Term Storage (CRP T13014) — hereinafter CRP on Demo — started in June 2012 is to support and share improvements in the nuclear power community's technical basis for the renewal of water reactor spent fuel dry storage licenses as durations extend. Results from the CRP on Demo are also expected to facilitate subsequent transport and disposal of spent fuel. Since the proposal of the CRP was drafted in August 2011 it was acknowledged that the Extended Storage Collaboration Program (ESCP) initiated by EPRI in November 2009 provided a broad context for the CRP on Demo which in turn could make important contributions by targeting specific needs. Accordingly, experts examined on-going gap analyses (gap between anticipated technical needs and existing technical data) and identified ten specific research objectives and corresponding specific activities of the CRP. The CRP on Demo also aims at increase the coordination with the efforts of the Long Term Interim Storage activities performed by the Nuclear Energy Agency (NEA) of the Organization for Economic Co-operation and Development (OECD). Consequently, the CRP contributes to increase the coordination among such efforts e.g. through joint IAEA-EPRI-NEA meetings so that they yield increased joint benefits for all participants. An indication of support for the CRP on Demo is that extra-budgetary funding is provided through the US Peaceful Uses Initiative (PUI). As agreed at the First Research Coordination Meeting (RCM) held in Argentina in April 2013, those ten specific research objectives originally identified were synthesized according to the topics covered by the signed Research Contracts and Research Agreements into the following six specific research objectives: Stress Corrosion Cracking Mechanisms and Monitoring, Rod Behavior, Concrete Systems, Bolted Closure Systems, Neutron Shielding and System Demo. The paper presents how the above mentioned six specific research objectives are being addressed by the participants form Argentina, the European Commission, France, Germany, Japan, Lithuania, Pakistan, Poland, the Russian Federation, Slovenia, Spain, the United Kingdom and the United States of America by highlighting the main results provided during the First and Second RCMs held in Argentina in April 2013 and Japan in November 2014, respectively.

SAFETY CONSIDERATIONS FOR THE INTERIM STORAGE SOLUTION

(Session 5)

Chairpersons

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SAFETY CONSIDERATIONS FOR THE INTERIM STORAGE SOLUTION

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Abstract

For more than 50 years, AREVA TN, part of the AREVA Group, has offered a complete range of transport and interim storage solutions for radioactive materials throughout the entire nuclear fuel cycle. Interim storage of used nuclear fuel is a reliable intermediate solution while waiting for a decision concerning disposal sites or recycling. Intermediate storage is safe as shown by important industrial feedback and the operational records. AREVA has developed different Used Nuclear Fuel dry storage solutions worldwide. Since the '80s AREVA TN has developed an entire set of transport and storage casks, the TN®24 family. Composed of 20 types of casks, this family has a wide range of capacities from 21 PWR to 97 BWR used fuel elements. These casks are currently in operation in Europe, the United States and Japan to safely provide interim storage of used fuel elements. In parallel, another type of interim storage system, the NUHOMS®, has been developed mainly for the US market. To date NUHOMS® is the dry storage solution of choice of more than 50% of U.S. nuclear facilities. Up to now more than 1000 dry storage systems designed by AREVA have been loaded throughout the world, representing a significant track record. The purpose of this paper is to present an overview of the solutions designed by AREVA for used nuclear fuel management, and specifically the different systems currently in operation in several Interim Storage Facilities throughout the world. This paper broaches as well some considerations related to the Safety Evaluation and Management program.

1. INTRODUCTION

Interim Dry Storage Systems must provide important functions to nuclear safety including thermal performance, radiological protection, containment, sub-criticality control, and retrievability. In addition, interim dry storage systems must be designed to withstand all accident conditions in storage configurations.

In a dry storage system, used nuclear fuel (UNF) that has already been cooled in the fuel pool is surrounded by inert gas inside a container called a cask or a canister. The container is typically a steel cylinder that is either welded or bolted closed. The steel cylinder provides leak-tight containment of the UNF. Each cylinder is surrounded by additional shielding materials, such as steel, concrete, or other material to provide radiation protection to workers and the public. Some containers can be used for both storage and transportation.

AREVA has developed different UNF dry storage solutions worldwide [1].

2. INTERIM DRY STORAGE: AREVA CANISTER SOLUTIONS

2.1. General description

The NUHOMS® system, which consists of the storage of canisters in concrete modules, was initially developed for 20-year interim storage at reactor sites. The used nuclear fuel is placed in leak-tight welded canisters or dry shielded canisters (DSC) filled with inert gas, inside a massive concrete bed or horizontal storage module (HSM) on- or off-site. The first barrier for the containment of used nuclear fuel assemblies is the dry shielded canister. In the future, DSCs will have to be transferred into a transportation cask for transport to centralized storage, reprocessing or final disposal.

The NUHOMS® system provides containment, shielding, criticality control and passive heat removal independent of any other facility structures or components.

The first NUHOMS® storage system was loaded in 1989 in the US. The NUHOMS® system is currently the leading technology used in the US. Up to now more than 800 NUHOMS® systems are in operation. NUHOMS® systems accommodate BWR, PWR and VVER used nuclear fuel and also some waste. A new generation of NUHOMS® systems is currently under development by AREVA with higher performance (especially higher fuel burnup).

The NUHOMS® system is shown in Fig.1.

FIG. 1. NUHOMS® system (USA).

The TN NOVA™ system is another canister system designed by AREVA. It is a system with a metal storage overpack which can be tilted vertically for on-site or centralized storage.

The TN NOVA™ system is shown in Fig. 2.

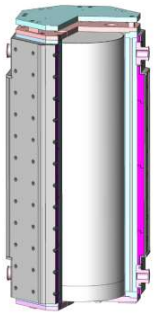


FIG.2. TN NOVA™ System.

The canister, for both the NUHOMS® and the TN NOVA™ systems, is licensed for both storage and transportation. The advantages of these solutions are modularity, passive cooling and low upfront costs.

2.2. NUHOMS® next generation: NUHOMS® extended optimized storage (EOS)

The NUHOMS® EOS system, offers the highest capacity and heat load in the industry plus significant advancements in aging management.

"*This system includes increased heat rejection, increased thickness of basket conductors for improved emissivity, the ability to store damaged fuel assemblies, options for an inspection port built into the EOS concrete module for ease and long-term aging management, and an option with a Duplex Stainless Steel canister which ensures that the system is corrosion-proof for plants in marine environments*" [2].

The EOS system is capable of storing 37 PWR or 89 BWR used fuel assemblies. The loading configurations allow for high thermal power of up to 50 kW and a high radioactive inventory resulting from used fuel with a burnup of up to 62 GWd/tHM, maximum allowable enrichment as high as 5% for PWR fuel and 4.8% for BWR fuel and a minimum cooling time as low as 3 years. The EOS 37PTH DSC and EOS 89BTH DSC are shown in Fig. 3.

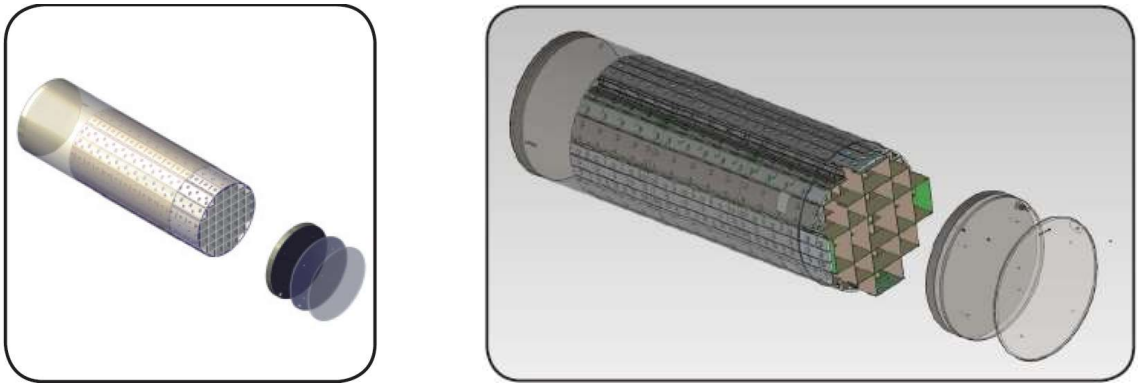


FIG.3. EOS 37PTH DSC and EOS 89BTH DSC.

The EOS 37PTH DSC and EOS 89BTH DSC are optimized designs for plants with minimum crane capacity of 125 tons. A 108 ton option is available. They are transferred to a new fully shielded NUHOMS® EOS Transfer Cask. The EOS 37PTH DSC and EOS 89BTH DSC are stored in a horizontal configuration using the NUHOMS® EOS HSM concrete modules.

The EOS HSM is a new and improved concrete module, with redesigned vents for a higher capacity heat load while maintaining the same overall footprint.

3. INTERIM DRY STORAGE: AREVA DUAL PURPOSE CASKS

3.1. General description

The TN®24 Dual Purpose Casks are metallic casks which can be used for both storage and transportation to and from a storage facility to avoid handling the fuel assemblies twice. They must comply with transport regulation requirements [3].

In Switzerland, Germany, Belgium, as well as in Japan the casks are stored in a specific building. In the US, casks are stored outside. The TN®24 casks are shown in Fig. 4.

FIG. 4. TN®24 casks in Belgium and in US.

AREVA has developed the versatile TN®24 family. These casks have been adapted to the specific needs of a variety of storage conditions and fuel. The cask capacities reach 40 assemblies for PWRs or 97 for BWRs, and maximum 65 to 70 GWd/t burnup. Cooling times of 5 to 7 years for the contents are accepted in these storage designs. Containment and dose are monitored during the interim storage.

More than 300 dual purpose casks have been in operation in the USA, Japan, Belgium, Switzerland, and Germany. The first TN®24 product was loaded in the US in 1984 for testing; and the first commercial loading was subsequently executed in 1988. For Europe, the first cask was the TN®24D which was loaded in 1994 in Belgium. Japan started the storage of used fuel at Fukushima-Daiichi in 1995 using a cask jointly designed and manufactured by AREVA TN and Kobe Steel.

The advantages are modularity, passive cooling, and dual purpose storage and transport capability.

3.2. TN®24E: Dual purpose casks for the German market

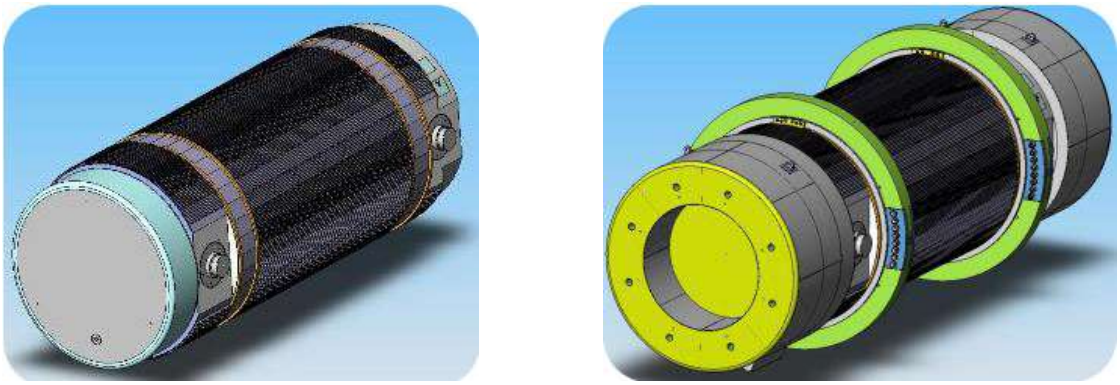
The TN®24E cask is a dual-purpose cask, designed for meeting the transportation and storage requirements of the German market. This cask is capable of storing, over a period of 40 years, 21 PWR used fuel assemblies which can be of type UO_2 or MOX, with a maximum of 17 MOX fuel assemblies per cask. The loading configurations allow for high thermal power of up to 39 kW and a high radioactive inventory including high gamma and neutron source terms resulting from used fuel with a burnup up of to 65 GWd/tHM, maximum allowable enrichment as high as 4.65% and a minimum cooling time as low as 2 years [4].

The TN®24E dual-purpose cask is mainly constituted of:

- A cask body consisting of a heavy-walled forged shell and a welded bottom forged cover;
- A basket placed inside the cask body;
- A primary lid and a secondary lid, both equipped with metallic gaskets.

In transport configuration, in order to withstand the accident transport conditions defined by the IAEA 2012, the cask is equipped with two shock absorbing aluminum rings in addition to the shock absorbing covers. In storage configuration, the cask is equipped with a protective cover which is designed to protect the flask in the accidental cases defined for storage

conditions (consequences of an airplane crash or missile impact on the storage building). The TN®24E cask is shown in Fig. 5.



Storage configuration. *Transport configuration.*

FIG. 5. TN®24E cask in storage and transport configuration.

The safety is mainly ensured by the mechanical properties of the cask body equipped with its shock absorbing covers. The cask body also ensures the shielding and the transfer of the decay heat. All forged components are made with high performance steel for its resistance to brittle fracture: A 508 Grade 4N class 3 according to ASTM standards.

For increased thermal performance the exterior of the cask body is equipped with 36 copper heat conductors and a surrounding outer shell equipped with heat transfer fins. Between the copper heat conductors is a patented epoxy vinylester resin compound, called TN®VYAL-B™, which gives the cask a high shielding performance.

4. SAFETY OF THE INTERIM DRY STORAGE SYSTEMS

The purpose of the storage system is to protect people, property and the environment from the effects of the storage of radioactive material. Therefore, storage systems are designed to ensure the safety of used fuel during the storage period and subsequent transportation after storage. They have been designed to meet type B package requirements of the transport regulations and the national requirements of storage regulations.

4.1. Proven records

The safety of UNF storage systems has been demonstrated globally. For the different systems manufactured by various companies worldwide there is good feedback from visual inspections and safety parameter monitoring including temperature, dose rate, and containment through pressure checking. The different operators have been sharing their knowledge on their dry storage systems for decades. Specific open-cask investigations have been conducted as well.

Some behavioral investigations have been performed in the United States. In particular, one demonstration test was done by the Idaho National Engineering Laboratory on the AREVA TN-24P cask and NUHOMS® system. The results were satisfactory [5]. Moreover, Post Irradiation Examinations of UNF assemblies stored 15 years in a metal cask showed no

significant degradation to the cask and, more significantly, no significant degradation to the UNF assemblies [6].

Japanese utilities of BWR NPPs, performed inspection campaigns related to the integrity of BWR used nuclear fuel. Also, integrity inspections of TN®24 casks stored at the Fukushima-Daiichi Storage Building were done after 5 and 10 years of storage of 52 BWR assemblies (~ 30 GWd/tU). No release of inner gas (^{85}Kr) and no defects were observed on the used nuclear fuel assemblies (visual checking) [7].

Finally, AREVA has been developing different Used Nuclear Fuel dry storage solutions worldwide for more than 40 years. Indeed, more than 1100 dry storage systems designed by AREVA have been loaded and are currently in operation. No operational or safety issues have been observed.

4.2. Storage safety assessment

The safety of the storage facility is based on the dry storage systems, i.e. the dual purpose casks or the canister systems. Indeed, the storage systems have been designed to withstand all normal, routine and accident conditions without the necessity of complementary protection systems such as an additional structure in case of storage inside a building.

The system has been designed to ensure the safety of the used fuel during the storage period and transportation.

The following minimal functions for nuclear safety have to be taken into account in the storage system design:

- Maintain sub criticality;
- Prevent release of radioactive material above acceptable amounts;
- Ensure radiation rates and doses so as not to exceed acceptable levels;
- Maintain retrievability of stored radioactive materials throughout the life of the dry system.

4.3. Storage safety in severe conditions

AREVA has performed several aircraft crash tests on scale models at the French CEA facility using missiles to simulate the impact of fighter planes. The objective of the impact test is to demonstrate that the cask remains leak tight after the impact.

Aircraft crash tests have been performed on two TN®24 cask models with different loads. Recently in 2010, the TN NOVA™ system, the AREVA canister-based storage system, withstood the impact of a missile simulating a fighter plane. A leak tightness test — mandatory for the test validation — had excellent results. Both tests were witnessed by ENSI/SVTI, the Swiss Nuclear Safety Authority representatives. The aircraft crash test on TN NOVA™ is presented in Fig. 6.

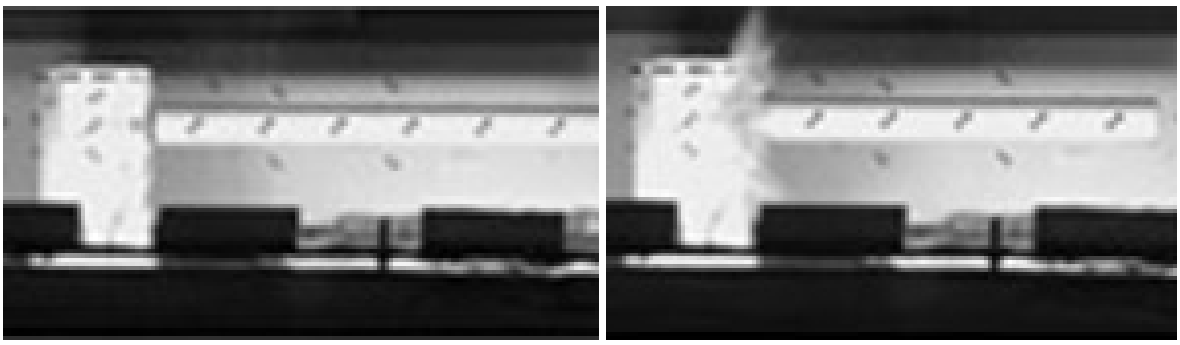


FIG. 6. Aircraft crash test on TN NOVA™.

The AREVA storage systems have demonstrated their ability to withstand severe accidents. In particular, the 9 TN®24 casks stored at the Fukushima-Daiichi Storage Building (with 408 BWR used nuclear fuel assemblies) endured the tsunami in March 2011. These casks have withstood large amounts of sea water, sand and rubble gushing into the building when it was hit by the tsunami [8]. Investigations in 2013 on the 9 dry storage casks showed no damage to either the casks or to the used nuclear fuel (in particular, 3 UNF assemblies were removed from the first cask for a visual inspection) [9].

4.4. Storage extension

Interim dry storage systems and interim storage facilities have been designed and licensed for a specific duration which is typically 20 to 60 years. It appears that storage duration may have to be extended much longer, potentially beyond one century, necessitating a reevaluation of safety throughout the service life of the storage: on-site interim storage, transfer, and transportation operations to the final disposal site or recycling facility.

The storage system needs to be designed to ensure the necessary safety functions during storage and subsequent transportation after storage. Consideration needs to be given to potential ageing deterioration of component materials that may occur during operation of the storage system.

Effective aging management programs require a technical understanding of the aging degradation mechanism, inspection and assessment techniques, prevention and mitigation measures (to retard the effects of aging) and, as needed, guidance on repairs or replacements for each component [10].

The main objectives of the aging management program are [11]:

- To ensure that the interim storage safety functions are maintained in the long term;
- To study the degradation phenomenon and mitigate the degradation;
- To address safety issues based on R&D on storage systems, components and used fuel.

And finally, special attention shall be paid on transportability after long term storage. The used nuclear fuel will have to be transported from the interim storage facility to somewhere else.

> "*It can be noted that the US Nuclear Regulatory Commission (NRC) approved, in November 2014, a 40-year license renewal for the NUHOMS® dry-cask independent used fuel storage installation at Exelon Generation's Calvert Cliffs Nuclear Power Plant. ....The approval includes an aging*

management program with periodic inspections of the used fuel dry storage systems and components to ensure potential aging effects are identified and effectively managed" [2].

This NUHOMS® system is the only nuclear fuel dry storage system with a US NRC-approved Aging Management Program addressing Chloride-Induced Stress Corrosion Cracking and canisters with high burn-up fuel.

5. CONCLUSION

Vast industrial experience and numerous safety studies have shown the ability of interim dry storage systems to protect the public and manage safely and economically used fuel from nuclear power reactors in the long term.

The interim dry storage solution is a key component of a used fuel management program while reprocessing or the implementation of final disposal is put in place. AREVA demonstrates continually its ability to provide a high level of safety and continued robustness of its dry storage solutions. And AREVA is developing innovative storage systems as well as innovative solutions to monitor and mitigate any potential deterioration of the systems.

NOMENCLATURE

NUHOMS® is a trademark of AREVA Inc. registered in the USA and in other countries.

TN® is a trademark of TN International (AREVA Group) registered in France, the USA and in other countries.

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EXPERIENCE OF CASK TECHNOLOGY FOR SNF MANAGEMENT

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Abstract

A new technology of spent nuclear fuel (SNF) management at the back-end of the fuel cycle has been developed over the last twenty years. This technology is based on the concept of the shielded cask ensuring containment of its contents (SNF) and compliance with all other requirements for SNF storage and transport. Radiation protection and activity containment are ensured by physical barriers, viz. an all-metal or composite body, body linings, internal baskets for irradiated / spent fuel assemblies (SFAs) and lids with sealing systems. SFA residual heat is released to the environment by natural irradiation and natural convection of air around the cask. This report considers the key issues associated with the creation of a family of dual purpose casks for SNF from naval and nuclear ice-breaker fleet activities, RBMK-1000 and BN-350 reactors as well as their structural peculiarities. The new development on cask for transportation of SNF VVER-1000 presents also.

1. INTRODUCTION

One of the overarching aims of Russian nuclear power industry development is SNF and radiation wastes management [1]. Adopted in the Russian Federation concept of SNF management involves its reprocessing at the final stage of nuclear fuel cycle. One fifth of SNF unloaded from reactors is subject to reprocessing. Remaining SNF volumes are to be stored pending reprocessing. Shortage of storage containers called for new safe and efficient solutions.

Approximately 10 years ago development of new SNF management technology was started. The technology principle is dual-purpose shipping package for SNF storage and transportation based on a metal-and-concrete cask (MCC) [2, 3].

2. FAMILY OF METAL AND CONCRETE DUAL PURPOSE CASKS FOR SNF

At the beginning of 2000 the most acute situation with SNF unloading was observed at decommissioned Russian NPSs. It was caused by incapability of the existing transport infrastructure and processing facility to meet the demand for shipment and handling of SNF from both NPSs being decommissioned and NPS reactors pending decommissioning.

The United States of America and Norway, together with the Russian Federation, took part in a trilateral joint Project 1.1 aimed at creating a pilot model of a cask for temporary storage and transportation of SNF from decommissioned NPSs. The Project was part of Arctic Military Environmental Cooperation (AMEC). In December, 2000 the Project was successfully completed by issuing the certificate-permission for NPS SNF construction and transportation. This was the first certified dual-purpose shipping package from MCC family with the index assigned TUK-108/1.

At the moment the development and licensing of dual-purpose metal-and-concrete casks family have been finished [4]:

- TUK-108/1 for NPS SNF transportation and storage;
- TUK-120 for storage and transportation of SNF from nuclear icebreakers;
- TUK-104 and TUK-109 for RBMK-1000 SNF transportation and storage;
- TUK-123 for BN-350 SNF transportation and storage (Aktau, Kazakhstan);

- TUK-121 for transportation and storage of high level waste.

The main stage of TUK-108/1 commissioning, as far as SNF transportation is concerned, was completed in 2003. Over 100 TUK-108/1 casks were manufactured and delivered to 'PO Mayak', Far-East Plant 'Zvezda', Ship Repair Center 'Zvyozdochka' and DalRAO.

Accumulation sites for TUK-108/1 temporary storage were built and have been operated in the terminals of SNF unloading from NPS reactors and SNF loading into casks such as Ship Repair Center 'Zvyozdochka', Far-East Plant 'Zvezda' and DalRAO.

According to the valid licenses the SNF technological storage in TUK-108/1 at accumulation sites cannot exceed 2 years. Initially, this term was limited to 6 months. In consequence of successful practices of TUK-108/1 operation in short-term mode it was decided to extend the SNF storage period to 2 years, without taking any extra measures for TUK-108/1 further safety.

TUK-108/1 has been actively used for over 15 years in order to ensure safe cask management of SNF from decommissioned NPSs.

Annexes for SFA cask storage to existing at Leningrad and Kursk NPP SNF storages were commissioned in 2012-2013. This helped solve SNF storage problem and continue NPP operation by utilizing dual-purpose casks TUK-109 for SNF storage and transportation.

Phase 1 of the chamber-type centralized dry storage was commissioned at Mining and Chemical Combine (Krasnoyarsk region) in 2013. SNF from Leningrad and Kursk NPPs is transported there in licensed shipping packages TUK-109. Over 100 UKKH-109 casks for SNF loading and temporary storage at NPP sites were manufactured and delivered to NPPs with RBMK-1000 reactors.

In order to move out UKKH-109 a removable energy absorption container (EAC) being a part of TUK-109 is used. EACs were produced only for transportation in number corresponding to special-purpose car quantity of a train.

Several years ago, they completed the delivery of manufactured in Russia TUK-123 for long-term storage and transportation of SNF from reactor facility BN-350 (Aktau, Kazakhstan).

Thus, at this point we have on industrial scale applied technology of cask storage and transportation of SNF with low level afterheat, namely from reactor facilities of RBMK-1000 and BN-350 type, as well as NPS transport power facilities.

3. ISSUES OF VVER-1000/1200/TOI SNF HANDLING

After a three–five year process storage at the NPP SNF from VVER-1000 reactors being operated in Russia is transported in TUK-10 and TUK-13 casks to the centralized pool-type storage facility at Mining and Chemical Combine. The technology of storing SNF in water requires significant operating costs that increase over time. There are also restrictions in terms of SFA wet storage. The design life time of this storage facility is extended up to 2025. At the same time, in spite of the activities carried out to increase the capacity of uranium, its capacity will be exhausted due to the placement of all SNF from VVER-1000 nuclear power plants operating in Russia, Ukraine and Bulgaria. In 2015-2016 the chamber-type centralized dry storage for SNF from VVER-1000 will be commissioned in order to provide SNF removal from the wet spent fuel storage facility the life time of which expires in 2015. In addition, it is

important to note that SNF can be received for dry storage only after 20 years of decay in water.

In order to ensure the ongoing operation of NPP with VVER-1000 it is planned to construct a cask storage area for dual purpose shipping packages at Mining and Chemical Combine to be commissioned by 2018-2020. Also, while designing the SNF cask storage area it will be taken into account that at the moment new types of NPP are being dynamically constructed in Russia and abroad. Therefore, VVER-1200 and VVER-TOI reactors of the new generation will generate SNF with a high burn-up of up to 70 MW/day per 1 ton of uranium. In the interest of safe transportation and storage of SNF with a high level of residual heat and radioactivity a new type of TUK that meets modern safety requirements is being created.

4. MAIN REQUIREMENTS TO MODERN DUAL-PURPOSE TUK FOR SNF FROM A NEW GENERATION REACTORS VVER-1200 AND VVER-TOI

In order to ensure safety requirements for storage of SNF with high burn-up and radioactivity the following criteria should be met:

- Guaranteed term of storage must be as high as possible, at least not less than the life time of NPP (in this case the requirements of foreign NPP customers that are committed to the deferred concept of SNF management shall be taken into account);
- By the end of this period it should be possible either to upgrade the storage facility with respect to aging of the appropriate equipment and new safety requirements gradually and without major cost, or gradually move to the SNF disposal and / or reprocessing, if that is preferred.

When choosing the best option for long term storage these requirements should be supplemented with the following provisions:

- Storage should be monitored, since currently there is no reliable data on the change in the state of SNF being stored during an extended period of time (over 60 years);
- In the process and at the end of the storage term it should be possible to retrieve SNF for reprocessing, disposal or transfer to more efficient technical and economic conditions of storage;
- Storage should be provided in a dry environment to prevent generation of additional RW.

Process technology of SNF storage in the cask storage areas using dual-purpose shipping packages (for storage and transportation) allows for optimally ensure all the conditions of SNF long-term storage with a maximum guarantee of safety.

The long-term storage is provided by the design of casks. Structural reliability and materials allow for storing SNF in casks from 60 to 100 years or more.

Monitoring of SNF status is provided within the test casks and SNF test samples at the mock-up benches simulating storage conditions in the casks. The storage environment that can influence the state of spent nuclear fuel is provided as well as the casks state with the possibility of replacement, if necessary. The casks with SNF can be promptly delivered to the places of study.

Possibility to retrieve SNF is provided by the design of the casks. Casks can be transported to the place of SNF retrieval in order to monitor the state of the fuel as well as to have it disposed or reprocessed.

With that sort of storage all the parameters of dry storage environment with the possibility to have it altered are provided and monitored.

If necessary, gradual update of individual elements and casks in general is possible.

The gradual transition to a disposal or reprocessing is provided by a modular type of construction of the building for cask storage, by constructing an extension, if necessary.

Nuclear safety during cask storage is achieved by using spacers and absorbing elements inside the casks as well as putting SFA into over sleeves, thus providing their isolation from each other regardless of the time and almost any design and beyond design basis external influences. Contact between SNF from different casks in case of natural or man-made emergencies is ultimately impossible.

Radiation safety in relation to the radiation from SNF is provided by a cask shielding and a multi-barrier sealing system ensuring virtually zero output activity. It is possible to monitor the level of radioactivity in the cask storage facility and identify the individual cask with a loss of sealing.

The heat removal of SNF is carried out without the use of forced cooling systems due to natural ventilation.

It requires little or no supporting systems to maintain safe conditions of storage and safety monitoring.

The most important feature of the process technology is guarantees of safety. They are provided by the following:

- The whole system is immune to natural disasters, including floods, earthquakes, etc.;
- Transport accident conditions with respect to which the casks are designed exceed the emergency exposure during storage;
- Transportation requirements to the cask design are unified around the world and strictly categorized, their performance is confirmed by conventional methods of calculation and testing, including full-scale crash tests, when it is virtually impossible for other types of storage;
- Safety justification is almost independent of the specific operating conditions, which allows to use all the acquired national and international experience in designing casks and make the best use of formal procedures for independent certification;
- Possibility to have defective structural elements monitored and replaced, including individual defective casks or gradually all the series in case of errors in the design decisions and the need to update in line with changes in safety requirements;
- Factory environment during the fabrication of casks that perform a major role in safety of the storage allow ensuring the quality of casks.

5. EXAMPLE OF A DUAL PURPOSE CASK FOR VVER-1000 SNF

Currently a lot of TUK-141O casks are being manufactured under FCNRS investment project 'Renewal of Shipping Packages Fleet for NPP SFAs Transportation to Centralized Handling Facilities'. TUK-141 is certified as package of B(U)F type. New casks are to be produced by Petrozavodsk branch of 'AEM-Technologies' in 2016. Table 1 provides the specifications for the TUK-141O.

6. CONCLUSION

The apparent simplicity of implementing the process technology of the cask storage is attractive to many owners of spent fuel. However, one should take a reasonable care not to use the method of the dry cask storage indiscriminately for a whole variety of types and characteristics of the nuclear fuel. Tough regulatory requirements are implemented at all stages of development, fabrication, testing and commissioning. This is evidenced by the experience of creating and using the family of metal-and-concrete casks for storage and transportation of SNF with a low level of residual heat.

Obviously, in the course of gaining the experience in the SNF cask storage the regulatory framework governing it will be improving as well.

Creating a new generation of dual casks for SNF from a new generation of VVER-1200 and VVER-TOI reactors is a true to life necessity. Commissioning of such TUK is planned by 2020.

TABLE 1. TUK-141O SPECIFICATIONS

Main specifications	Value
Sum SFA maximum after power in TUK / for 1 Uranium SFA, kW	40* / 2.25
SFA cooling period, years	from 5 years
SFA holding capacity, items	18
TUK life time, years	60
Storage term, years	60
Temperature range at TUK operation, °C	from - 40°C to +38°C (specified range)*
TUK loaded weight (with SFAs) with (without) dampers, tons	up to 115
Empty TUK weight with (without) dampers, tons	96 500
Max. height with (without) dampers, mm	7120 (5600)
Max. diameter with (without) dampers, mm	2770

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SPENT FUEL DRY STORAGE: CHALLENGES AND LESSONS LEARNED FROM RECENT PROJECT EXPERIENCE AT SHUTDOWN NUCLEAR PLANTS IN THE U.S.

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Abstract

Decommissioning of nuclear power plants is on the rise with planned retirements of older plants, and premature shutdowns due to post-Fukushima energy policy shifts and lessening economic viability for some plants in deregulated markets. Safely and securely storing the spent nuclear fuel is vital for the period between plant shutdown and dismantling. In the U.S., dry storage is viewed as a key enabler to successfully implement decommissioning at shutdown sites. Many challenges emerge when dry storing fuel at a decommissioning site. This paper will provide an update on the U.S. decommissioning landscape and will review the key drivers for dry storage in decommissioning projects. Furthermore, we will review the challenges and lessons learned from two NAC decommissioning projects.

1. INTRODUCTION

Dry storage of spent fuel is broadly accepted in the U.S. as a safe and economic alternative to wet storage at sites running out of in-pool capacity. Currently, about 95% of operating reactors in the U.S. have implemented dry storage, and it is anticipated that all U.S. operating nuclear plants will require dry storage at some point before 2025 [1]. In addition to the ongoing implementation and continued operation of dry storage facilities at operating plant sites, various nuclear power plants have permanently ceased operation and have been completely defueled relying on robustly designed and rigorously analyzed casks as the means to safely dry store the spent fuel onsite. These systems offer significant cost savings by reducing operations and security staffing for the site once the spent fuel pool has been emptied of nuclear fuel. In fact, eight of these sites have already been fully decommissioned and their Independent Spent Fuel Storage Installation (ISFSI) are the only facilities remaining onsite. With recent market trends challenging the profitability of nuclear reactors, such as low gas prices, renewable subsidies, anticipated costs to address security and plant modifications to extend nuclear plant licenses, and other political factors, it is very likely that more reactors could shut down in the near term, requiring these sites to assess their decommissioning options, which most likely will include consideration of dry storage as a key component of a successful decommissioning strategy.

This paper provides recent information regarding spent fuel storage in the U.S. for decommissioning sites. In particular, we provide an overview of two important NAC projects (the Zion Nuclear Plant and Kewaunee Power Station decommissioning projects) where the implementation of dry storage is a critical component in the decommissioning scope. Both of these projects rely on NAC's MAGNASTOR® dry storage systems to provide safe and efficient defueling operations, while meeting unique customer requirements, including fuel inventories containing damaged and underburned (high reactivity) fuel and in support of aggressive loading schedules. For instance, one of these projects requires 3½-year cooled fuel to be transferred to dry storage, which accelerates spent fuel pool closure and provides rapid reduction in operations and security expenses for spent fuel pool operations. Finally, this paper shares some of NAC's key lessons-learned from its relevant experience. Key elements

for which lessons have been learned include safety, site and operational dose performance, quicker pool defueling, damaged fuel, lightly-burned fuel, GTCC waste, non-invasive draining and drying operations, seismic requirements, large-scale fabrication delivery needs and schedule demands. In summary, NAC is providing significant capabilities and insights to support successful dry storage projects for sites that have permanently ceased operation.

2. U.S. PLANT SHUTDOWN AND DECOMMISSIONING STATUS

There are currently 99 operating commercial nuclear reactors in U.S., a moderate decrease from just three years ago when there were 104 units operating in the country [2]. Since 2013, five reactors have been shut down prematurely for a number of reasons and twelve (12) other reactors continue to be at risk of closure due to market uncertainties, such as low power prices (driven by low natural gas prices and subsidized renewables), high costs of major plant modifications, regulatory issues, and local concerns with safety and reliability [3]. Recent shutdowns include: 1) Duke Energy's Crystal River (PWR – 860 MW(e)) in Florida, which is slated for decommissioning due to high cost to repair extensive damage to the containment building structure sustained during a steam generator replacement project in 2010. After various attempts of repair and significant cost overruns, the plant owner decided to prematurely shutdown the reactor in February 2013; 2) Dominion Energy's Kewaunee (PWR – 566 MW(e)) in Wisconsin, was permanently shut down in May 2013, after almost 39 years in operation. Dominion cited economic reasons in a highly competitive deregulated market; 3) In June 2013, Southern California Edison decided to permanently shut down two San Onofre reactors (PWR−1080 MW(e) each) in California due to regulatory delays and uncertainty related to design issues and damage in newly installed steam generators. SCE also cited economic reasons given the uncertainties to address the steam generator issues needed to bring the plant back on line; and 4) Finally, in December 2014, Entergy permanently shut down its Vermont Yankee reactor (BWR - 635 MW(e)) in Vermont due to economic reasons.

The tally of U.S. shutdown nuclear reactor sites is provided in Table 1. There are twenty (20) shutdown commercial reactors at seventeen (17) sites whose fuel inventory remain stored onsite. Three of these reactors (Dresden 1, Indian Point 1 and Millstone 1) are at sites that have other operating reactors and their fuel disposition and decommissioning is currently being deferred until other reactors on site are also shut down and decommissioned. The remaining 17 shutdown reactors are located at fourteen (14) sites with no other operating reactors. Of these fourteen (14) sites, ten (10) have moved all the spent fuel into dry storage and four (4) sites, Kewaunee, Crystal River, San Onofre and Vermont Yankee are planning to defuel the spent fuel pools within the next five years. Table 1 also provides the current decommissioning status of each of the shutdown sites. It is important to note that of all the sites that have ISFSIs, nine (9) have already completed or are in the process of dismantling and decommissioning, leaving only the spent fuel storage facility onsite, thus termed “orphan sites” by many in the U.S. An important trend that can be deduced from Table 1 is that spent fuel is promptly moving into dry storage at shutdown site, whether decommissioning will occur promptly or be deferred. This decision to dry store at the plant site is made in part because there are no other alternatives absent a national repository or a centralized interim storage facility, but the decision to promptly load the spent fuel into dry storage instead of retaining it in the pool until active decommissioning begins is driven by considerations of safety, security, costs and stakeholder input.

TABLE 1. U.S. SHUTDOWN REACTOR SITES WITH EXISTING SPENT FUEL INVENTORY

Reactor name	BWR	MW(th)	Location	License issued	Shutdown date	Status
Big Rock Point	BWR	240	Charlevoix, MI	5/1/1964	8/29/1997	DECON Completed **ISFSI on Site**
Fort St. Vrain	HTG	842	Platteville, CO	12/21/1973	8/18/1989	
Haddam Neck	PWR	1825	Meriden, CT	12/27/1974	12/5/1996	
Maine Yankee	PWR	2700	Wiscasset, ME	6/29/1973	12/6/1996	
Trojan	PWR	3411	Ranier, OR	11/21/1975	11/9/1992	
Rancho Seco	PWR	2772	Herald, CA	8/16/1974	6/7/1989	
Yankee Rowe	PWR	600	Franklin Co., MA	12/24/1963	10/1/1991	
Humboldt Bay 3	BWR	200	Eureka, CA	8/28/1962	7/2/1976	DECON in Progress **ISFSI on Site**
Zion 1	PWR	3250	Zion, IL	10/19/1973	2/21/1997	
Zion 2	PWR	3250	Zion, IL	11/14/1973	9/19/1996	
San Onofre 1[a]	PWR	1347	San Clemente, CA	3/27/1967	11/30/1992	
LaCrosse	BWR	165	Genoa, WI	7/3/1967	4/30/1987	SAFSTOR with **ISFSI on Site**
Crystal River 3	PWR	2609	Crystal River, FL	12/3/1976	2/20/2013	SAFSTOR **ISFSI pending**
Kewaunee	PWR	1772	Kewaunee, WI	12/21/1973	5/7/2013	
San Onofre 2	PWR	3438	San Clemente, CA	2/16/1982	6/7/2013	
San Onofre 3	PWR	3438	San Clemente, CA	11/15/1982	6/7/2013	
Vermont Yankee	BWR	1912	Vernon, VT	3/21/1972	12/29/2014	
Dresden 1	BWR	700	Morris, IL	9/28/1959	10/31/1978	SAFSTOR (other operating reactors)
Indian Point 1	PWR	615	Buchanan, NY	3/26/1962	10/31/1974	
Millstone 1	BWR	2011	Waterford, CT	10/31/1986	7/21/1998	

[a] San Onofre 1 is in advanced stage of decommissioning with spent fuel already in dry storage. There is an ISFSI on site accommodating fuel from all three units on site. Units 2 & 3 are in SAFSTOR pending Decommissioning and complete defueling into a new ISFSI pad.

2.1. U.S. decommissioning options and drivers

U.S. regulations require decommission to be complete within sixty (60) years after permanent cessation of operation. There are three acceptable decommissioning methods permitted by U.S. regulations:

- DECON: The equipment, structures, and portions of the facility containing radioactive contaminants are removed or decontaminated to a level that permits release of the property and termination of the NRC license;
- SAFSTOR: The facility is placed in a safe stable condition and maintained in that state until it is subsequently decontaminated and dismantled to levels that permit license termination. This approach is often referred as "deferred dismantling or delayed DECON". This method uses the time elapsed as a decontaminating means, allowing a substantial amount of the radioactive elements in components to decay to stable elements. In addition, the funding level for the decommissioning fund may also increase as it earns interest over time;
- ENTOMB: Consists on encapsulating or encasing radioactive contaminants on site in structurally sound material such as grout or concrete. The facility is then maintained and monitored until the radioactivity decays to a level permitting restricted release of the property many decades later. To date, no NRC-licensed facilities have requested this option. The only ENTOMB projects in the U.S. were pursued on various prototype reactors that were shutdown prior to the formation of the U.S. NRC in 1975.

A plant licensee may also choose to adopt a combination of the first two options in which some portions of the facility are dismantled or decontaminated while other parts of the facility are left in SAFSTOR. The decision may be based on factors besides radioactive decay, such as availability of waste disposal sites, status of decommissioning funds, available resources, cost of security, etc. These factors become key drivers when considering the removal of spent nuclear fuel and irradiated hardware from the plant—accounting for over 99 percent of the radioactive inventory. Thus, decommissioning of a spent fuel pool, lowers the radiation level in the facility and significantly reduces the potential exposure to workers during subsequent decommissioning operations among other safety and security benefits covered later [4].

Table 1 also provides the decommissioning status for all the shutdown reactors grouped into distinct categories. Seven (7) shutdown reactors have already completed DECON having only the spent fuel ISFSI onsite; NAC technology was utilized for dry storage at three (3) of these sites. Of the remaining 13 reactors, only 4 are in DECON process with the remaining 9 opting for the SAFSTOR until DECON is initiated. Nevertheless, regardless of the option pursued, all of the sites, except those with operating reactors onsite, have opted to defuel spent fuel pools into dry storage.

2.2. Drivers for dry storage in decommissioning projects

Reactors that are permanently shut down and have no fuel in the reactor vessel present a significantly lower risk to public health and safety than operating reactors. The U.S. Nuclear Regulatory Commission recognizes this and permits shutdown plants to seek exemptions from certain requirements that enable a reduced staffing at the site. Removal of all of the nuclear fuel from the spent fuel pool into dry storage is another key milestone in decommissioning that lowers risk and enables some reduction in plant staffing. Prompt defueling of the spent fuel pool at a shutdown site paired with a robust, proven and efficient dry storage technology offers significant benefits in each of the key areas that drives decommissioning success:

(1) Safety;
(2) Security;
(3) Economics;
(4) Stakeholder involvement.

Dry Cask Safety

Dry cask storage systems have been touted by many as inherently safe technology. This assertion was confirmed after 9/11 by the U.S. National Academies of Sciences (NAS) at the request of the U.S. congress to provide independent scientific and technical advice on the safety and security of commercial spent nuclear fuel storage. The NAS report concluded that dry storage offers a more secure and passive mean to store spent fuel than in-pool storage [5]. In the NAS report, safety refers to measures that protected spent nuclear fuel storage facilities against failure, damage, human error, or other accidents that would disperse radioactivity to the environment. What is important to note is that these particular measures are typically already considered by design in modern dry cask storage systems without jeopardizing efficiencies. It is precisely these particular efficiencies that produce measurable safety benefits when loading a large amount of systems in a single decommissioning dry storage campaign. Key technology attributes that support this claim are:

- *Optimum Dry Storage System Capacity:* Dry cask loading evolutions present challenges in nuclear safety (fuel handling), radiological safety (personnel exposure during cask processing, including neutron radiation), and industrial safety (heavy lifts of the empty and loaded system, placement of lids, working on elevated surfaces, and operation of heavy equipment such as the vertical cask transporter and automated welding system). Reduction in the overall number of loading evolutions for the same population of fuel decreases the risk of safety events. This can be achieved with high capacity systems that don't challenge a plant infrastructure;
- *Efficient System Packaging and Shielding Designs*: Increasing the density of used fuel stored results in improved off-site dose characteristics as the fuel is shielded by its neighbouring assemblies. NAC's experience has shown that robust high density designs like the MAGNASTOR system can result in virtually undetectable doses from spent fuel at site boundaries, thus supporting protection of the public. This attribute is very important when considering that at decommissioning all the spent fuel pool is offloaded to the ISFSI pad in a single campaign, including the most recently discharged fuel;
- *Streamlined Operations with Safety in Mind*: Simple, proven operations minimize worker exposure. Streamlined procedures result in improved ALARA, a key safety element, through less worker activity and less time around the radiation environment. These factors are important in large decommissioning campaigns. Even the smallest details have a value added effect when considering a large number of systems loaded in one campaign. A key example, the NAC MAGNASTOR system uses a single shielding/structural closure lid for confinement, along with a light (approximately fifty pounds) closure ring for confinement redundancy. Many dry storage systems require two lids to be installed. Lid placement is a critical lift evolution (weighing in excess of five thousand pounds). The single lid design constitutes a design enhancement with major positive safety implications that reduces work activities during loading;
- *Optimized Ancillary Equipment:* Minimizing the need for external system connections to perform every aspect of equipment processing offers significant ALARA and industrial benefits. For example, one NAC customer uses an integrated vacuum drying system that requires only a single set of connections to the canister for all of its processing functions (i.e., one set of boom lines to the vent and drain ports, as well as thermocouple connections to the transfer cask). Compared to multiple separate pieces of equipment for each function (e.g., hydrostatic testing, vacuum drying, etc.), this configuration minimizes occupancy time on top of the loaded system, reducing

personnel exposure. Elimination of multiple equipment manipulations also reduces the probability of human performance-induced events.

There are numerous factors not covered here, and certainly even the smallest benefit should not to be discounted when loading a large number of systems. No matter how small these safety benefits are, their compounded contribution during a large decommissioning pool defueling could amount to measurable improvements in safety.

Security

When nuclear fuel is removed from a reactor, and removed from the adjoining spent fuel pool, the security risk and security oversight footprint decreases significantly, permitting streamlined operations. After operating safety systems are terminated, and virtually all plant radioactive content is confined to an ISFSI facility with passive and robust casks, security staff monitor and oversee a smaller, convenient and well-protected area away from the RCA. Transitioning to dry storage is heralded as a key element of efficient decommissioning strategy. This conclusion is supported by some stakeholders near shutdown sites, pointing to the findings of the NAS report identified earlier, which indicated that dry cask storage is less vulnerable to attacks. In particular, the NAS report cited two key factors: (1) these are passive systems that rely on natural air circulation for cooling; and (2) it divides the inventory of that spent fuel among a large number of discrete, robust systems [5]. The NAS report further concludes that the robust construction of these casks prevents large-scale releases of radioactivity in all of the attack scenarios studied. This notion of inherent advantages over pool storage has created heightened interest with some stakeholders in getting fuel out of shutdown pools post 9/11 and post-Fukushima.

Economics

Besides the safety and security advantages of dry casks, there are strong economic drivers for removing fuel from pools to reduce the safety and security risk of the site. This can account for about $10 M−$30 M annual cost savings in reduced operations and security. At the Kewaunee shutdown site, the estimated savings with SAFSTOR approach after defueling the spent fuel pool is approximately $13 M per year when compared to operations with the fuel in the pool [6]. This is a compelling factor that incentivizes defueling of a shutdown site as soon as possible. However, in order to support this goal, a dry storage technology must be able to accommodate hotter and more challenging spent fuel contents and the system capacity must be optimized to minimize loading schedules and resource requirements. At Kewaunee, Dominion has selected NAC's MAGNASTOR to complete defueling operations within 3½ years of plant shutdown, which is at least two years earlier than originally anticipated. This is goal is achievable since the MAGNASTOR system has an optimized regional loading configuration to handle very hot fuel while putting colder fuel on the periphery and a high capacity (37 PWR assemblies) offering significant direct savings due to the reduced number of casks needed. Fewer cask loadings represent reduced impacts on plant operations, fewer oversight activities, fewer heavy load movements and related safety risks, lower worker radiation dose exposures, smaller storage facilities, fewer casks to maintain, smaller storage sites to monitor, and so on. Therefore, it is easy to conclude that a system that can store more spent fuel will allow a utility to defuel faster, and achieve higher dry storage economics, safety and operations by simply buying and loading fewer casks.

Stakeholder involvement

Stakeholders for decommissioning efforts range from utility investors to state and town commissions, state and local politicians to public interest groups, employees to members of the communities near the plants. The common theme among these stakeholders is they want to know that the plant will be decommissioned timely and safely, and they want transparent communication on decommissioning plans and progress. Since dry cask storage is an early issue to be addressed, it receives much attention from stakeholders. Most of the issues can be easily addressed by the inherent safety, security, and economic advantages of dry storage. However, NAC has found, especially in a post-Fukushima environment, that stakeholders expect that the decommissioning plan, and certainly the dry storage technology, adequately covers not only regulatory requirements, but conditions beyond the licensing basis of the casks, even if the probability of occurrences are negligible. Significant resources are invested by cask vendors to evaluate such events. In addition, some stakeholders have made demands that that are far less rigorous, such as consideration for aesthetics of the casks, local participation and engagement, environmental justice, etc.

Most of these stakeholder interests can be positively addressed by the dry storage system designers. In particular, features that drive safety, security and economic performances are routinely addressed as demonstrated in two recent NAC projects described next.

2.3. Recent decommissioning dry storage experience: Zion nuclear power station

The Zion nuclear power station (Fig. 1) is a two reactor PWR (1,040 MW(e)) plant located in the state of Illinois on Lake Michigan just north of the city of Chicago. The plant owner at the time, Commonwealth Edison (now Exelon), decided to permanently shut the units down in 1998, 16 years prior to its license expiration, primarily due to economic reasons. At the time, the plant required costly repairs that exceeded the value of the plant. When operations ceased, the plant owner realized that the plant decommissioning funds were not sufficient to immediately pursue decommissioning and dismantling activities and opted to place the facility in configuration for safe preservation. The initial plan was that decommissioning work would not begin until the mid-2020s by which time the decommissioning funds would have accumulated enough interest to cover the full cost of decommissioning.

In September 2010, EnergySolutions (ES) acquired the Zion plant from Exelon under a 'license stewardship' contract, whereby ES took ownership of the site during decommissioning. Under this agreement, ES became the plant licensee and had access to the plant decommissioning funds to perform decommissioning. At the time, it was estimated that full decommissioning would cost about $1 billion dollars over the next 10 years. The only remaining facilities after decommissioning would be the ISFSI with spent fuel, for which Exelon will retain the facility license and title of the fuel until the U.S. Department of Energy takes title to and possession of the fuel for subsequent interim storage or permanent disposal.

In order to meet this commitment for the Zion project, ES identified the transfer of spent fuel to dry storage as a critical activity in order to reduce the radiological and security risk during decommissioning. In 2010, ES selected the NAC MAGNASTOR system, with a capacity to store 37 spent PWR fuel assemblies. In September 2011, ES began the decommissioning process. Table 2 offers a profile of the Zion spent fuel transfer project.

TABLE 2. ZION DRY STORAGE PROJECT PROFILE

Plant owner:	ZionSolutions, a subsidiary of EnergySolutions		
Number of assemblies:	2226 total	Schedule	Date
Fuel contents characteristics:	• 36 High Burnup assemblies;	Contract	10/5/10
	• 57 Leakers or Damaged Fuel;	Fuel transfer started	12/20/14
	• 1481 Repaired (containing ITTRs, etc.).	Fuel transfer completed	1/11/15
Cask technology:	NAC MAGNASTOR 37 — Assembly PWR		
Project scope:	61 Transportable Storage Canisters (TSC) and Vertical Concrete Casks, 4 GTCC systems, two sets of transfer ancillaries, engineering and onsite support services.		

Zion key challenges

The Zion project has been the largest single loading campaign in U.S. with sixty-five systems loaded (61 loaded with fuel and 4 loaded with GTCC, Fig. 1). At the time the project started, the MAGNASTOR system was a relatively new design with the highest industry storage capacity licensed for any multipurpose canister based system. Although the MAGNASTOR technology offered significant technical advantages over other alternatives, there were various challenges that needed to be resolved to ensure the technology was ready for the Zion defueling campaign. Table 3 summarizes some of these challenges in their respective risks areas and strategy implemented to mitigate those risks.

NAC and ZionSolutions worked closely to tackle these challenges. A strong teamwork approach thrived throughout the project. ZionSolutions' loading experiences and lessons learned were incorporated into design improvements, which have benefited NAC's other customers.

FIG. 1. Zion nuclear power station (left) and ISFSI with 61 MAGNASTOR Systems and 4 GTCC casks (right).

TABLE 3. ZION DRY STORAGE PROJECT SUCCESS FACTORS

Risk area	Mitigating factors implemented
Risk of delivery − Fabrication of large number of systems (61 fuel canisters and 4 GTCC canisters).	NAC relied on diversified fabrication to minimize delivery risk: Three (3) fabricators were used (GEH and Petersen in the U.S., and Hitachi Zosen in Japan). Despite the increased oversight and associated costs, the use of multiple fabrication facilities was a prudent decision. The Zion loading campaign never stopped because of a delayed hardware delivery.
Schedule risk − Largest dry storage loading campaign to be implemented in very short duration (about 1 year).	To load systems in tandem utilizing two transfer casks and related transfer equipment. Procedures were optimized to ensure this efficient process could be supported with available resources. Based on this approach Zion was able to load about 61 systems in 55 weeks.
Licensing risks − Consideration for decommissioning spent fuel contents that includes damaged fuel and high burnup fuel and other special contents.	A large amount of Zion fuel (1478 assemblies) included tie-rod repairs, 3 assemblies contained guide tube anchor repairs. In addition, there were 36 high burnup assemblies, 57 damaged bundles and some "underburned" fuel. NAC worked with the Zion team to ensure the system licensing bases was amended to include all these contents.
Operation and implementation risks – Deployment of a relatively new system	During the Zion dry runs and initial loadings, NAC collaborated on improvements that could streamline the loading operations. There were some issues related to system tolerances and fabrication non-conformances that had to be addressed promptly. The TSC design and operational aspects were continually analyzed over the project, and the averaged loading time per canister continually improved.
Meeting site boundary dose rates.	NAC supported Zion project in order to optimize cask loading and placement so as to limit the dose at the site boundary, which is less than 550 feet south of the ISFSI. To address this requirement, the MAGNASTOR loading permits loading of lower source term fuel on the cask periphery and hotter fuel in interior cell locations. NAC supported Zion in developing a fuel-loading plan consistent with the ISFSI off-site dose calculation.

2.4. Ongoing decommissioning dry storage experience: Kewaunee power station

Dominion Energy's Kewaunee Power Station (Fig. 2) was a 1772 MW(th), PWR located in Kewaunee, WI. The plant operated from December of 1973 to 7 May 2013. Spent fuel is currently stored in both its spent fuel pool and an onsite ISFSI with 14 TN NUHOMS horizontal systems (Fig. 2). In 2013, the company selected NAC International to provide a turnkey solution based on the NAC MAGNASTOR system to offload the remaining spent fuel from the spent fuel pool into dry cask storage by December 2016, which is just about 3½ years from plant shutdown. Table 4 offers some highlights of the Kewaunee spent fuel transfer project.

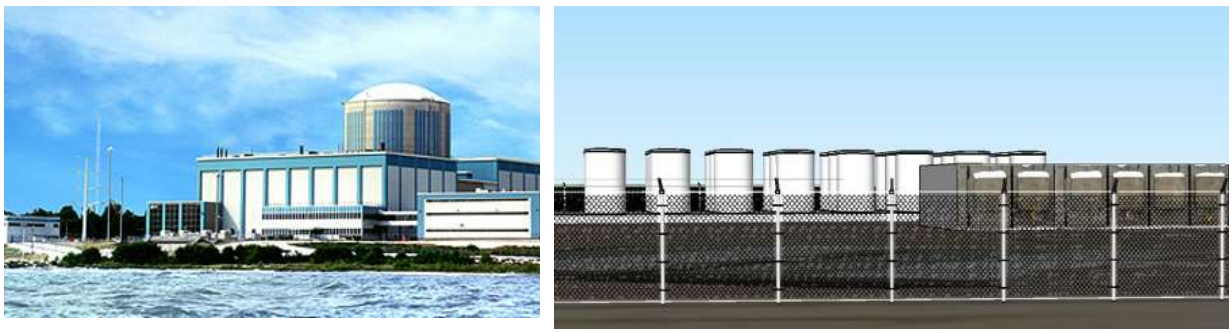

FIG. 2. Kewaunee nuclear station and rendering of future ISFSI pad.

TABLE 4. KEWAUNEE DRY STORAGE PROJECT PROFILE

Plant owner:	Dominion energy		
Number of assemblies:	887 total	Project schedule	Date
Fuel contents characteristics:	• Intact fuel assemblies;	Plant shutdown date	5/7/2013
	• Underburned assemblies;	Contract	12/11/13
	• High burnup fuel;	Fuel transfer begins	04/04/16
	• Damaged and repaired fuel;	Fuel transfer completed	11/15/16
	• Other non-fuel hardware contents.	(3.5 years after shutdown)	
Cask technology:	NAC MAGNASTOR 37-Assembly PWR		
Project scope:	24 Systems and lease of transfer ancillaries. Engineering, licensing, ISFSI pad design and construction, full scope pool to pad operations.		
Status:	MAGNASTOR CoC Amendment 5 published in the Federal Register. ISFSI expansion design and site A/E complete. Permits approved. System hardware and construction in progress. Site construction is underway.		

Kewaunee key challenges

At Kewaunee, there are various challenges that must be addressed to achieve operational readiness for the 2016 defueling campaign. The following table summarizes some of these challenges in their respective risks areas and the strategy being implemented to mitigate those risks.

TABLE 5. KEWAUNEE DRY STORAGE PROJECT SUCCESS FACTORS

Risk area	Mitigating factors implemented
Licensing risks − Consideration spent fuel contents that includes spent fuel with just 3 ½ year of cooling time, underburned spent fuel assemblies from the last cycle, damaged fuel, high burnup fuel and other special contents.	NAC developed a 4-zone regionalized cask loading pattern that will permit an optimized distribution of the Kewaunee contents among the 24 systems to be loaded. Using this approach, none of the systems will exceed 27 kW while utilizing the 37-assembly capacity, which is well below MAGNASTOR's current 35.5 kW rating. NRC approval of the license amendment to meet these content requirements is pending.
Schedule risk – Very aggressive implementation schedule, from contract award to pool offload in less than 3 years with limited dominion resources.	NAC scope for this project will include total pool to pad operations and ISFSI design and construction. NAC is pursuing an integrated project approach working closely with subcontractors such as Westinghouse (for loading the systems) and Sargent Lundy (design and construction of the ISFSI second pad). Lessons learned from previous MAGNASTOR deployment have been now incorporated in the Kewaunee procedures.
Operation and implementation risks – Addressing site specific issues such as Seismic requirements.	NAC designed a Secure Integrated Lift Yoke / Chain Hoist Assembly that will provide positive control of the loaded transfer configuration during canister transfer, by ensuring both the transfer cask and the loaded canister remain attached to the site's seismically qualified building crane during the entire transfer operation. This design yields significant ALARA and cask loading time improvements over the use of fixed seismic restraints.

The Kewaunee project is benefiting from the operational experience gained during the Zion dry storage project, which includes proven and optimized loading procedures, incorporation of design and operational enhancements to expedite canister processing while reducing dose rates, proven fabrication and construction specifications, proven training documentation, work control processes, and many other attributes of a mature dry cask technology. Combined with the Kewaunee specific activities outlined above, NAC and Dominion expects the project schedule to remain on track.

3. SUMMARY AND CONCLUSION

The decommissioning market in the U.S. is on the rise. In the last three years, five (5) nuclear commercial reactors have been shut down and a few others are at risk of closure. Shutdown site owners are required to assess the best decommissioning strategy considering driving factors such as safety, security, economics and stakeholder involvement. Predominantly, the preferred option aligned with these factors has been to defuel spent fuel pools at shutdown sites and defer decommissioning activities until a more optimum time. Modern dry storage technology, like NAC's MAGNASTOR system offers tremendous opportunity to accelerate a plant defueling schedule, saving the plant owners millions of dollars in reduced site security and operation costs. In this paper, NAC presented two key decommissioning projects, Zion and Kewaunee, each with unique spent fuel transfer requirements that were addressed with the MAGNASTOR cask technology. Significant operating experience was gathered during the Zion project and lessons learned are currently been applied to the Kewaunee project. The Kewaunee project is preparing to defuel the spent fuel pool by December 2016, two years before previously anticipated, thus saving the plant owner more than $20 million. The benefits of high capacity dry storage and accelerated defueling were also shown to demonstrate its relevance to and impact on a successful decommissioning strategy.

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SPENT FUEL STORAGE INTEGRATION IN THE UNITED STATES PLANNING FOR STORAGE AND TRANSPORTATION

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Abstract

Within the United States, the Nuclear Regulatory Commission (NRC) has the responsibility for licensing and inspection of spent fuel storage operations. This includes storage in the spent fuel pool, typically licensed under Title 10 Code of Federal Regulations (CFR) Part 50 regulations, and within spent fuel cask systems licensed by 10 CFR Part 72 regulations. The spent fuel cask systems are also referred to as dry cask storage systems. The spent fuel is adequately protected from criticality concerns and other events whether stored in the spent fuel pool or in a dry cask storage system. The licensee decides when to move the spent fuel to a dry cask storage system based on operational and economic needs. Over the past twenty-nine years, the United States dry cask storage systems have transitioned from single-purpose storage-only cask systems to dual-purpose cask systems, with the ability to be transported to a centralized storage facility or ultimately to the final disposal site. Occasionally, these dual-use cask systems experience defects during fabrication or deficiencies during spent fuel loading operations that may render the canister unable to meet storage requirements under 10 CFR Part 72 regulations and transportation requirements under 10 CFR Part 71 regulations. These defects and deficiencies may require exemptions by the NRC and tracking of the component by the licensee. This paper will briefly discuss: the transition within the United States from single-use cask systems to dual-use cask systems; several strategies considered for deciding when to move the spent fuel from wet storage into dry cask storage; processes to transition the canister loaded with spent fuel from a storage system to a transportable system; and deficiencies that can render a dual-use cask system into a single-use cask system.

1. INTRODUCTION

1.1. Background

Storage of spent nuclear fuel in dry canisters has been practiced in the United States since July 1986, when the NRC issued the first site-specific license to Surry Nuclear Power Station (Surry) in Virginia. The storage system used was a Castor V/21 nodular cast iron cask, which held 21 PWR fuel assemblies. The Castor V/21 canister like many of the early canisters licensed by 10 CFR Part 72 was licensed for storage-only. Each licensee understood that these storage only canisters would have to be offloaded and reloaded into another canister for transportation, resulting in additional radiological dose to the workers and expense to the licensee. The materials that were used for these initial storage-only canisters/casks were composed of cast iron, carbon steel, and stainless steel. The canisters were designed and constructed to use either a bolted lid or a welded lid design.

Initially, the reactor licensees planned to store the discharged spent fuel in the spent fuel pools for a nominal amount of time until the spent fuel could be shipped to a reprocessing facility or turned over to the U.S. Department of Energy (DOE). The U.S. Government declared a moratorium on spent fuel processing in 1977. Although the moratorium was later lifted, there has not been any commercial spent fuel reprocessing since. As the spent fuel pools continued to fill up, licensees turned to re-racking their pools to a higher density configuration to provide additional storage capacity. Ultimately, the licensees turned to storage of the spent fuel in dry cask canisters when additional re-racking of the spent fuel pool wasn't feasible. Surry, like most licensees, re-racked their spent fuel pool prior to implementing dry fuel storage.

Many licensees preferred to remain with the spent fuel in wet storage, located in the spent fuel pools, as long as possible to delay implementation and costs associated with dry fuel storage.

Following the Fukushima Dai-ichi event, there has been an increase in the number of concerns expressed that storage of the spent fuel in the spent fuel pool is unsafe.

> "*The NRC has reviewed the safety of the spent fuel in the pool, as compared to the spent fuel stored in dry storage, and has concluded that both spent fuel pools and dry casks provide adequate protection of public health, safety, and the environment and that the likelihood of an accident involving a radiological release from the spent fuel remains extremely small*" [1].

In 1990, the NRC revised 10 CFR Part 72 regulations to provide an alternate to the use of a site-specific license for the storage of the spent fuel. A provision was added to allow nuclear generating plants with a 10 CFR Part 50 license to use a general license for storage of spent fuel. A general license allows the 10 CFR Part 50 licensee to use a 10 CFR Part 72 cask system approved by the NRC as long as the conditions specified in the Certificate of Compliance (CoC) and technical specifications are met by the licensee.

1.2. Dual-purpose canisters

As the spent fuel cask industry matured, the spent fuel canister designs progressed to a dual-purpose canister that could be licensed for storage under 10 CFR Part 72 and for transportation under the provisions of 10 CFR Part 71. Most dual-purpose canisters are made from stainless steel and serve as the confinement vessel while the spent fuel assemblies are being stored in a concrete or metal cask under the requirements of 10 CFR Part 72. A dual-purpose canister may be loaded, stored, and transported to an interim or permanent storage location when loaded in an approved transportation cask. The material properties of these stainless steel canisters are more suitable for storage and transportation than many of the previous storage-only canisters. Many of these dual-purpose canisters were first licensed under the provisions of 10 CFR Part 72, with approval for transportation to be granted later by 10 CFR Part 71.

With the introduction of dual-purpose canisters, competition drove the dry fuel vendors to design canister types that would hold more spent fuel assemblies with higher fuel burnup values and the capability to store damaged fuel assemblies. With the cost to load a spent fuel canister remaining approximately constant, many utilities compare the cost of spent fuel storage based on the average storage cost per fuel assembly. Thus, canisters that hold more fuel assemblies are seen as more cost effective. Initial dual-purpose canisters were typically licensed to hold 24 Pressurized Water Reactor (PWR) assemblies and 64 Boiling Water Reactor (BWR) assemblies. Current spent fuel storage canister capacities that are being licensed for storage and transportation include 37 PWR fuel assemblies and 89 BWR fuel assemblies.

There is no benefit to loading a single-purpose canister that is licensed for storage-only compared to the benefits of a using a dual-purpose canister. The existing single-purpose canisters may continue to be relicensed, allowing for extended storage at the respective Independent Spent Fuel Storage Installation (ISFSI) sites. However, current regulations require that the spent fuel assemblies from the single-purpose canisters be unloaded and reloaded into transportable canisters before shipment to an interim storage or permanent disposal site. Currently, no new single-purpose canisters are being manufactured or loaded in the U.S. As discussed later, even canisters that are licensed for dual-purpose use can be fabricated or loaded incorrectly in such a way that the ability to store and transport the spent fuel is at risk.

1.3. Multi-purpose canisters

The Nuclear Waste Policy Act, enacted by Congress in 1982, required that utilities establish contracts with DOE to take the spent fuel at the end of the fuel cycle. In December 1987, Congress amended the Nuclear Waste Policy Act to designate Yucca Mountain, Nevada as the only site to be characterized as a permanent repository for all of the nation's nuclear waste. DOE, the ultimate owner of the spent fuel on the back end of the U. S. spent fuel cycle, was developing the storage requirements for the final disposal facility to be located at Yucca Mountain. The Safety Analysis Report (SAR), based on the submittal by the DOE, assumed that the majority of the commercial spent fuel assemblies would be shipped to Yucca Mountain in Transportation, Aging and Disposal (TAD) canisters. For those utility sites where the spent fuel could not be repackaged into TAD canisters on-site, the multi-purpose canisters would be transported to a repository, where they could be stored or repackaged into a TAD canister for ultimate disposal [2]. The TAD canisters would be made from a corrosion-resistant Alloy-22 outer shell and a stainless-steel (Type 316NG) inner shell to provide structural support. The Alloy-22 consists mostly of nickel, chromium (up to 22.5 percent), and molybdenum (up to 14.5 percent). Type 316NG stainless shell consists mostly of iron, chromium (up to 18 percent), nickel (up to 14 percent), and molybdenum (up to 3 percent). To minimize the heat load generated from the waste packages associated with long-term disposal and the effects upon the rock between the tunnels in Yucca Mountain, the number of spent fuel assemblies that each canister could contain was planned to be a much smaller number than licensees currently load into dual-purpose canisters. The contents for the TAD canisters were specified to be 21 PWR spent fuel assemblies or 44 BWR spent fuel assemblies. The TAD canisters were to be designed to meet the requirements of 10 CFR Part 72 for storage, 10 CFR Part 71 for transportation, and 10 CFR Part 63 for repository disposal [3]. The NRC never licensed these multi-purpose canisters although the canisters were reviewed as part of the Yucca Mountain application. There are no plans to develop these canisters further by the DOE at this time.

2. DISCUSSION

2.1. Utility loading

Many utility customers delayed implementation of dry fuel storage to avoid associated expenses as long as the spent fuel pools contained sufficient capacity. Common methods for delaying the construction of an ISFSI were to re-rack the spent fuel pool to high-density storage racks. Newer nuclear power plants and those with larger spent fuel pools were able to delay implementation of dry fuel storage for a considerable amount of time. Eventually, the pools had been re-racked to maximum capacity and the utility must transition into construction and operation of an ISFSI to continue to operate the reactor. Many utilities delayed implementation until very near the time that the spent fuel pools would lose the capacity to accept the fuel assemblies discharged from the reactor at the end of the operating cycle, also known as full core discharge capacity. Delays in the utility preparing for initial implementation of the ISFSI could, and in some cases did, lead to loss of full core discharge capacity for a period of time.

In preparing to load the spent fuel canisters, several methods have been successfully employed by the reactor licensees. Many times the initial spent fuel loading plan at a reactor site used fuel assemblies that had lower decay heat loads and had been discharged from the reactor for a longer period of time. Selecting this fuel provided additional time for the licensee to complete canister loading operations before needing to cool or circulate the water inside the canister to avoid the water approaching the boiling temperature. An additional benefit of this

choice was to lower canister dose rates during the initial loading process while the loading crews were gaining experience. As the licensee's cask loading experience increased the reactor site would typically use spent fuel with increased decay heat loads. Many reactor sites that wanted to keep radiological dose rates down during the loading operation would load the fuel assemblies with lower decay heat around the outside edge of the canister with the higher decay heat spent fuel assemblies located in the center of the canister. The fuel assemblies with lower decay heat helped to provide radiological shielding for the assemblies located in the center containing higher dose rates, thus reducing overall worker exposure.

Fully decommissioned reactor sites do not have a spent fuel pool to repackage the canisters licensed for storage-only. Currently, all the decommissioned reactor sites in the U.S. have utilized dual-purpose canister designs that are capable of being licensed for transportation. Additionally, these decommissioned reactor sites no longer have fuel buildings with overhead cranes to transfer the canisters from the storage casks into transportation casks. Some utilities have planned accordingly for this potential problem. One utility fabricated a transfer station that was essentially a large tripod where a transfer cask could rest over the storage cask and the canister could be lifted up into the transfer cask. The storage cask was then removed from beneath the tripod and replaced with the transportation cask. Finally, the dual-purpose canister would be lowered into the transportation cask and preparations for shipment could occur. Another utility placed all the dual-purpose canisters into casks licensed for storage and transportation so they would be ready to ship when the decision was made to move the casks to an interim or final disposal site. There are several plants that have constructed cask transfer facilities outside of their 10 CFR Part 50 facility that are specifically designed to lower the canister into the storage cask. These cask transfer facilities could also be used for future loading of the canister into a transportation cask.

2.2 Canister loading issues

The loading requirements for dual-purpose canisters licensed for use by general licensees are included in the CoC and technical specifications. These loading requirements include such items as fuel types, fuel loading patterns, cooling time, decay heat load, initial enrichment, burn-up, and fuel assembly degradation. In the U.S., one of the key attributes for selection of the fuel assemblies is whether the fuel is considered damaged or undamaged. The current version of NRC Interim Staff Guidance (ISG) 1 provides a performance-based definition of damaged, undamaged, and intact fuel based on whether the canister is being used for storage (Part 72) and/or for transportation (Part 71). Damaged fuel is defined as any fuel rod or fuel assembly that cannot fulfill its fuel-specific or system-related functions. This would include fuel assemblies with defects larger than pinhole leaks or hairline cracks that could not be handled by normal means. Additionally, the absence of fuel pins not replaced by dummy rods and missing structural components, such as grid spacers, could also qualify the fuel assembly as damaged. Intact fuel contains no structural damage and no cladding defects. Undamaged fuel does not contain structural damage but may contain cladding defects that are no larger than pinhole leaks or hairline cracks. Current regulations require that the damaged fuel must be loaded in damaged fuel cans inside of the dual-purpose canister to be transportable. The licensees use a variety of methods to assist in determining if the discharged fuel is undamaged or damaged. These methods consist of reviewing core operating records that indicate whether failed fuel pins were present during the operating cycle, in-mast sipping of the fuel assemblies, vacuum sipping, and use of ultrasonic testing to examine the spent fuel pins for evidence of water. While the use of core operating records and sipping techniques are very reliable, the use of ultrasonic testing has a greater degree of uncertainty.

If a fuel assembly is found to contain defects larger than pinhole leaks or hairline cracks resulting in a grossly breached spent fuel rod with or without structural damage, the spent fuel assembly must be placed in a damaged fuel can inside of a dual-purpose canister licensed for the damaged fuel. If a fuel assembly is considered as intact or undamaged the fuel assembly may be loaded into canister without the use of a damaged fuel can. If the spent fuel assembly loaded into a canister is found to not meet the definition of undamaged fuel it must be removed from the canister. If this is not possible because the issue was discovered after the canister had been welded shut, the damaged fuel assembly must either be removed by unloading the canister or may remain loaded if an exemption from the applicable 10 CFR Part 72 storage requirements can be justified by the licensee and approved by the NRC. Typically this type of exemption would void the dual-purpose benefits afforded by the canister, rendering the canister approved for storage only. There have been several examples where fuel assemblies have been loaded into a canister that was not licensed for damaged fuel and was later found to contain an assembly that was classified as damaged.

In two of the examples described below, the fuel was examined by the use of ultrasonic testing methods to determine if the fuel was damaged by evidence of water located in the fuel pin(s). The fuel was from operating cycles where there was evidence of damaged fuel based on the water chemistry. The use of in-mast sipping techniques hadn't been employed at the time when these spent fuel assemblies were discharged. The licensee had employed ultrasonic testing techniques along with a visual examination to determine if these older fuel assemblies met the definition of undamaged fuel for inclusion in the loading of the spent fuel canister.

In the first instance, the licensee employed a specialty contractor to perform the ultrasonic testing. As part of the process, the contractor provided the licensee results based on the initial ultrasonic readings in a preliminary report before leaving the site. The contractor indicated that all the fuel rods containing water had been properly identified in the preliminary report and assured the licensee that the results of the final report would mirror the preliminary report. The licensee had been provided similar reports in the past from this contractor, which was always mirrored in the final reports. Based on this preliminary report, the licensee selected the fuel assemblies for loading into the dual-purpose canisters holding 32 PWR fuel assemblies. Approximately one month later, the specialty contractor notified the licensee that there were 13 additional fuel assemblies that were classified as damaged that had not been identified as damaged in the preliminary report. By the time the report had been received, five of these 13 fuel assemblies had already been loaded into four separate canisters that were already in storage on the ISFSI pad. It was too late to perform additional inspections on these five assemblies to determine whether the defect size on the fuel pins was larger than a pinhole leak or hairline crack. The licensee subsequently classified the five assemblies as damaged and notified the U.S. NRC of the event [4]. The licensee requested and obtained an exemption from the NRC to the requirements of 10 CFR 72.212 and 72.214 allowing the continued storage of the affected canisters containing the damaged fuel assemblies under the provisions of 10 CFR Part 72 [5].

On another occasion, the licensee had selected fuel to be loaded into a canister licensed for storage of undamaged fuel, which included older fuel that had also been tested using the ultrasonic testing method. The licensee used a forced helium drying method to remove the moisture from the canister prior to backfilling with helium. During the forced helium drying process, the pressure is reduced exhausting helium to an appropriate discharge location. After the discharge occurred, the control room received an alarm alerting the staff to an abnormal condition. Upon closure examination, the discharge from the forced helium drying process was determined to have emitted Krypton-85 gas. The only source of this noble gas was one of

the fuel pins located in the canister that was currently being dried. The lid-to-shell weld had already been completed on this canister by the time that the damaged fuel assembly was identified. The licensee continued with the loading process by drying the canister, backfilling the canister with helium, and welding the vent / drain ports shut. The NRC was notified of the event after the canister had successfully been backfilled with helium and welded shut. The presence of the noble gas indicated that at least one of the fuel pins had a leak. Unfortunately, there was no method available to quantify the size of the leak using gas samples after the canister helium backfilling and welding operations had been completed [6]. The licensee requested and obtained an exemption from the NRC to the requirements of 10 CFR 72.212 and 72.214 allowing the continued storage of the affected canisters containing damaged fuel under the provisions of 10 CFR Part 72 [7].

In both of the examples discussed above, the NRC exemptions reminded the licensee that the requirements of 10 CFR 71.87(a) specify that, prior to transporting the canister, the licensee must verify the contents are appropriate for the transport packaging. Currently, the inclusion of a fuel assembly classified as damaged in a canister not licensed to contain a damaged fuel assembly would not be authorized for transport under 10 CFR Part 71.

The final example of loading damaged fuel assemblies into a canister not licensed for their storage involved a utility that had performed the classification of their spent fuel assemblies prior to guidance being issued by the NRC. The DOE definition of damaged fuel at the time of fuel classification defined failed fuel as evidence of structural deformity, damage to the cladding or spacers, which may require special handling. Additionally, the licensee defined damaged fuel as an assembly with a pin that had a gap greater that a fuel pellet length and greater than one-half a fuel pellet width as determined by a visual examination. Prior to the initial fuel loading, the NRC issued the site-specific license to the utility that incorporated the NRC definition of damaged fuel included in ISG-1, revision 0, which included defects in the cladding larger than pinhole leaks or hairline cracks. However, the utility did not recognize that the definition of damaged fuel that had been included in the license was different than the DOE criteria used during the fuel classification performed four years previous. Six spent fuel assemblies that would have been classified as damaged where loaded into five canisters before the change to the damaged fuel definition had been recognized. The NRC reviewed the circumstance and revised the site-specific license to allow the five loaded canisters to remain in storage, but did not approve the misloaded canisters for transport [8].

There have been several examples where the utility loaded spent fuel into a canister that didn't meet the CoC requirements specified in the general license for decay heat and burn-up limits. One such example involved a licensee that incorrectly loaded five fuel assemblies that exceeded the maximum decay heat loads specified in the CoC at the time of loading into four different canisters. The affected fuel assemblies had been in storage for 2.5 years at the time the error was discovered and had subsequently decayed to a value below the specified CoC limit. A onetime exemption to the requirements of 10 CFR 72.212(b)(3) and 72.212(b)(11) was granted by the NRC to the licensee for storage on the ISFSI pad [9].

More recently, one utility has been attempting to address a welding issue that has occurred on six loaded dry fuel canisters. Five of the canisters are located at the ISFSI and the sixth canister is still located on the refueling floor. The NRC resident inspector observed the Non Destructive Exam (NDE) technician perform a dye penetrant exam and questioned whether the cleaning and dwell times were sufficient to detect potential flaws. The licensee determined that the dwell times for both the penetrant and developer for the lid-to-shell weld and the closure plate weld were not in compliance with procedure requirements. This led to a review

of not only the canister that was located on the refuel floor, but also included the additional five canisters located on the ISFSI pad that had been examined by the same two NDE technicians. Based on reviews and additional exams all six canisters have been declared inoperable [10]. The utility has examined several options to return the canisters to an operable condition including unloading the canisters, removing portions of the canister welds and re-welding the canister, and asking for an exemption to the requirements of 10 CFR Part 72 to store the canisters on the ISFSI pad [11]. Before the NRC can approve an exemption, additional supporting data is needed to ensure that the canister will be able to maintain confinement capability for the life of the canister. The utility has been working to develop volumetric examinations using a phased-array ultrasonic testing technique, to determine if there are flaws in the cover weld. Results of such an examination might be used with an appropriate engineering examination to demonstrate whether the canisters could be stored in their current condition using an exemption request to meet the requirements of 10 CFR Part 72.

The issues described above were attributable to errors made by the licensee or their contractors. There have also been examples where the cask vendors have made errors that could affect the use of the canisters for storage. In the example discussed below the cask vendor incorrectly implemented the requirements of 10 CFR 72.48 for performing an evaluation to determine if the proposed change required approval by the NRC prior to implementation.

The cask vendor performed a 10 CFR 72.48 safety evaluation for the removal of a requirement to perform a helium leak rate test of the shop fabricated canister shell to test the welded confinement boundaries. The cask vendor's 10 CFR 72.48 evaluation determined that prior NRC approval was not necessary. However, the NRC determined that this was a violation of 10 CFR 72.48(c)(2)(viii) which required an amendment before altering a design basis limit for fission product barrier or result in a departure from a method of evaluation as described in the Final Safety Analysis Report. The NRC had accepted the helium leak rate test as one of the tests performed by the cask vendor to demonstrate integrity of the canister confinement boundary. This level of assurance by the NRC was considered necessary for the maximum heat load and source term limits permitted to be loaded in the canister. There were 107 canisters that had been fabricated and loaded with spent fuel without being helium leak tested prior to discovery of the 10 CFR 72.48 violation. These canisters were evaluated by the cask vendor and found to maintain sufficient helium to ensure fuel cladding temperatures were within acceptable limits and effluent doses would be bounded by the dose rates for normal, off-normal, and accident conditions of storage [12]. The NRC accepted this evaluation and the dual-purpose function of the canisters were not affected. The canisters that had not been loaded at the time of discovery were helium leak rate tested by the cask vendor before being loaded with spent fuel by the utilities.

3. CONCLUSION

The dry cask industry continues to mature as evidenced by the advancement of canisters that can hold more spent fuel assemblies loaded with higher heat loads and with high burn-up fuel. However, with all these technological advancements, the processes are not so advanced that errors cannot occur. The industry must remain vigilant in the fabrication and loading processes associated with dry fuel storage to avoid invalidating the dual-purpose advantages of the current cask designs and avoid human error traps.

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DEVELOPMENT OF AN EXECUTION STRATEGY ANALYSIS (ESA) CAPABILITY AND TOOL FOR STORAGE OF USED NUCLEAR FUEL (UNF)

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Abstract

An Execution Strategy Analysis (ESA) capability and tool is being developed to evaluate alternative execution strategies for future deployment of a consolidated Interim Storage Facility (ISF) using a consent-based siting process per the Administration's Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste (Strategy). Application of an ESA approach not only leverages on but also goes beyond traditional project analysis tools. The ESA tool allows for on-going performance assessment of the evolving project execution plan that takes into account significant assumptions, risks, and uncertainties throughout the project lifecycle. The ESA process and tool are used to support the development of plans, budgets, and alternative execution/ implementation strategies for meeting the goals in the Strategy. The paper describes the process by which the ESA capability and tool are continuing to be developed and addresses the value of applying such a process in implementing a long-term strategy for managing used nuclear fuel (UNF) and High Level Radioactive Waste (HLW) [1].

Key Words: dynamic simulation modeling; project performance assessment; risk-informed decision support; alternate implementation strategies.

This paper has been removed from the proceedings as it has been reproduced elsewhere. Interested readings should refer to Ref. [1] and may view the entire paper at the following link: http://www.predicusconsulting.com/wp-content/uploads/2015/06/IAEA-ESA-Paper-150605.pdf

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DESIGN CONCEPTS AND OPTIONS FOR CONSOLIDATED INTERIM STORAGE OF SPENT NUCLEAR FUEL IN THE UNITED STATES

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Abstract

The United States Department of Energy (DOE) is beginning to lay the groundwork for implementing interim storage of spent nuclear fuel (SNF) as recommended by the Blue Ribbon Commission on America's Nuclear Future (BRC). These plans include activities to: 1) establish one or more Interim Storage Facilities (ISF) using consent-based siting; and 2) prepare for large-scale transport of SNF. The BRC's report to the Secretary of Energy was published on January 26, 2012[1]. The Administration's Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste was released on 11 January 2013. The Strategy includes a phased, adaptive, and consent-based approach to siting and implementing a comprehensive management and disposal system. As part of a series of ongoing foundational activities the DOE has completed initial studies to more fully develop and document a suite of generic design alternatives for an away from reactor interim storage facility focused on the receipt and storage of SNF in welded canisters. Concepts developed include: 1) The currently deployed and U.S. licensed above grade vertical and horizontal storage systems associated with each canister design; 2) A standardized storage over pack; 3) Below grade cylindrical vertical storage cavity and closure lid that provides radiation shielding and structural protection during storage; and 4) Vault System above and below grade. This paper will review some of the design concepts developed. The paper draws upon material produced by Chicago Bride and Iron for the United States Department of Energy.

1. COMMERCIAL SPENT NUCLEAR FUEL INVENTORY IN THE UNITED STATES

Commercial Nuclear Power Reactors (NPRs) have operated in the U.S. since about 1960. A total of 130 commercial NPRs have been built for civilian nuclear power generation. Nine of these were early prototype or demonstration reactors which have since been or are in a state of being decommissioned and for which Spent Nuclear Fuel (SNF) no longer remains on site. The high temperature gas cooled Fort St. Vrain demonstration reactor which was also decommissioned and the SNF is stored in a DOE Independent Spent Fuel Storage Installation (ISFSI) near the reactor site. Of the remaining 120 NPRs, one was never operated at power to make electricity and was decommissioned. A second was disabled and the fuel debris is owned by the DOE and stored at the Idaho National Laboratory (INL).

Therefore, 118 commercial power reactors have SNF remaining on site in pools or dry storage casks. Of these, 19 reactors have shut down, currently leaving 99 NPRs licensed to operate. There are 4 new reactors under construction and two reactors being completed where the construction had been previously suspended. The General Electric facility at Morris, Illinois is currently the only Nuclear Regulatory Commission (NRC) licensed storage facility in operation that is not co-located at a reactor site. Fig. 1 provides the approximate location of the SNF and identifies those sites that have implemented dry storage.

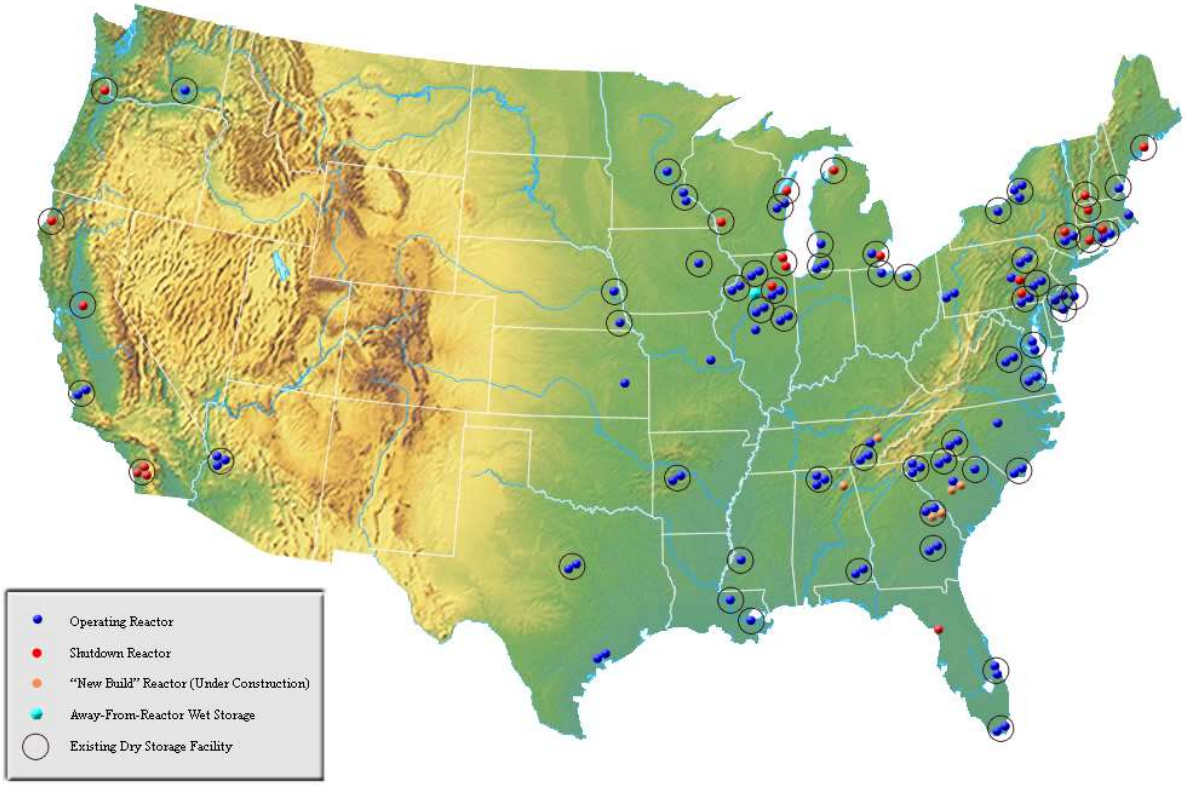

FIG. 1. Utility owned commercial nuclear power spent nuclear fuel in the United States.

The 19 shutdown reactors are comprised of 4 groups. Ten reactors (on 9 sites) ceased operations prior to 2000. Reactor decommissioning is complete or nearing completion at these sites. The Blue Ribbon Commission (BRC) identified these sites as "Stranded", since after reactor decommissioning, the only nuclear operations remaining will be monitoring the dry fuel and GTCC-LLW in storage [1]. The SNF at these sites is in dry storage. Six reactors (on 4 sites) have either ceased power generation or ceased efforts to restart since 2013. These are often referred to as "Early Shutdown" reactors since they did not operate for the NRC allowed 60 years[1]. The current total number of shutdown reactors at sites with no continuing nuclear operations is 16 reactors on 13 sites. There are 3 permanently shut down reactors on 3 sites co-located with operating reactors. Additionally Oyster Creek does not plan to operate for 60 years and has a utility-announced early shutdown date of 2019.

This shutdown reactor fuel represents a planning basis for the first consolidated Interim Storage Facility (ISF). Fig. 2 provides additional details on the number of fuel casks, GTCC LLW casks, assembly count, and metric tons of heavy metal for these permanently shutdown reactors. This fuel is expected to be moved to dry storage to support site decommissioning. Fig. 2 indicates the number of fuel casks loaded and those forecast. The first ISF focused on shutdown fuel could reasonably range from 250 to 660 fuel casks containing approximately 3000 to nearly 10 000 metric tonnes (MT) of SNF. An additional 15 to 34 casks of GTCC-LLW generated by the decommissioning of the commercial power reactors will also be

[1]The Nuclear Regulatory Commission licenses power reactors for an initial 40 year period and currently allows a single 20 year renewal period for a total of 60 years of operation.

transferred to the ISF in order to completely de-inventory the shutdown reactors sites of nuclear materials.

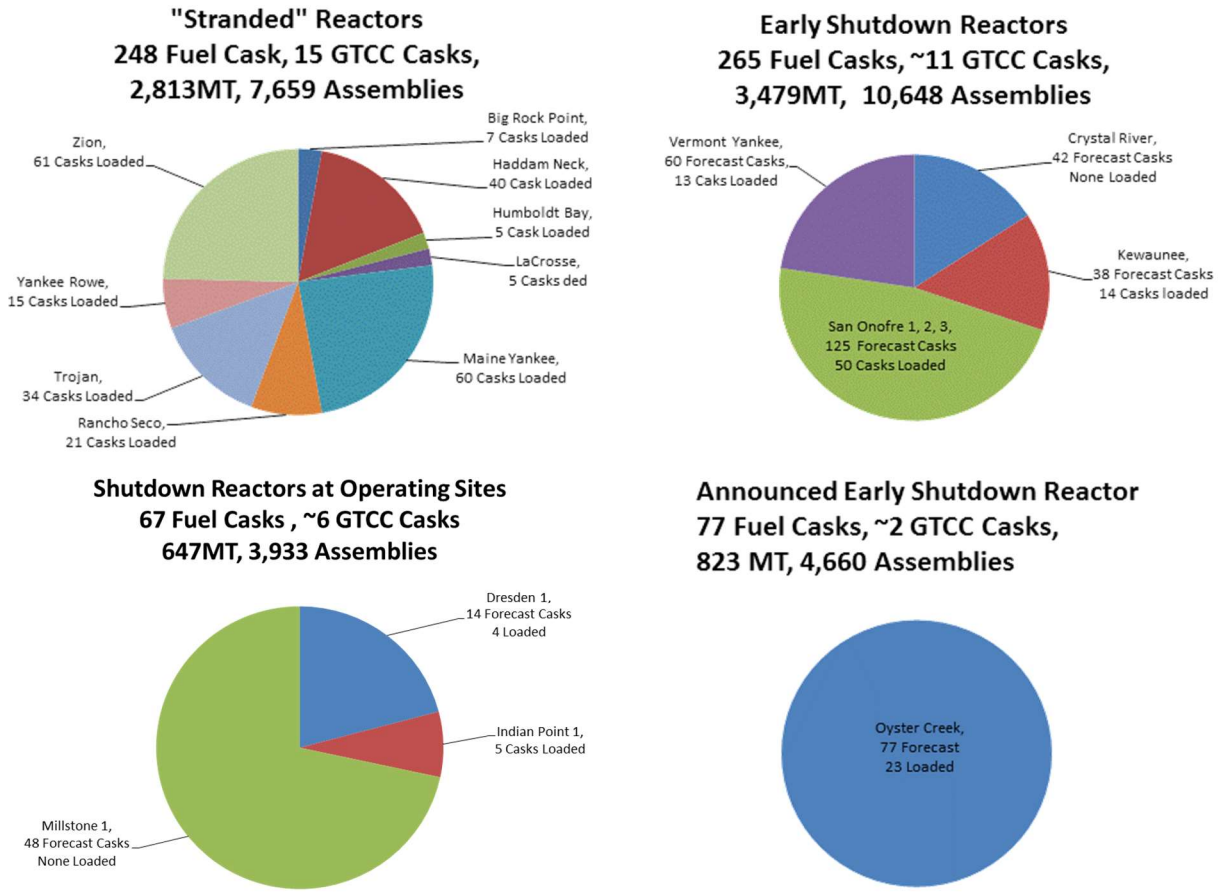

FIG. 2. Shutdown reactor fuel.

2. CURRENTLY DEPLOYED DRY STORAGE SYSTEMS

During SNF loading and dual purpose canisters (DPC) transfer between the fuel pool and storage cask, a metal transfer cask provides physical protection and radiation shielding. During storage, a concrete overpack provides those functions. The storage overpack uses a bolted lid. The concrete storage overpacks provide passive heat transfer by natural convection from the DPC through air vents.

Two design variations of the canister-based system concept are vertical storage inside an overpack and horizontal storage of the DPC inside a concrete horizontal storage module. There are differences between the two systems in the concrete storage overpack design, canister orientation, and transfer process. Vertical storage overpacks are placed in arrays on the concrete storage pad. Horizontal storage modules, which are a pre-cast reinforced concrete structure, are placed side-by-side in a linear configuration on the concrete storage pad. The vertical DPC is typically transferred into a storage overpack in the nuclear plant and then transported to the concrete pad. The horizontal DPC is transported to the pad in the transfer cask and then transferred into the horizontal storage module.

Four companies provide DPC systems. AREVA Transnuclear provides horizontal storage configurations, while EnergySolutions, Holtec International, and NAC International provide

vertical storage configurations. The shutdown reactor ISFSIs have deployed 10 storage systems offered by these vendors in about a 60% / 40% vertical / horizontal ratio.

The shutdown reactor canister-based systems are dual purpose; licensed under Title 10 of the Code of Federal Regulations Part 72 (10 CFR 72) for storage and 10 CFR 71 for transportation. They use a DPC where SNF assemblies are placed into a welded sealed metal container which provides the primary confinement boundary for the SNF. Table 1 summarizes the deployed shutdown reactor ISFSI storage cask systems, comprised of 21 canister families, using potentially 33 DPC variants and requiring the use of 9 transportation overpacks to move the canisters from the reactor ISFSI to the ISF.

TABLE 1. COMMERCIAL OVERPACKS AND ASSOCIATED CANISTERS

Cask system	Canister family	Canister	Transport cask	Total canisters[a]
Fuel solutions	W74	W74T	TS125	7 loaded
HI-STORM 100	MPC-32 (HI-STORM)	MPC-32 (HI-STORM)	HI-STAR 100	5 loaded
	MPC-68 (HI-STORM)	MPC-68 (HI-STORM)	HI-STAR 100	70 forecast, 13 loaded, remainder canister uncertain
		MPC-68F (HI-STORM)	HI-STAR 100	
		MPC-68FF	Not available	
		MPC-68M	Not available	
HI-STORM TranStor	MPC-24 (TranStor)	MPC-24E (TranStor)	HI-STAR 100	34 loaded, canister uncertain
		MPC-24EF (TranStor)	HI-STAR 100	
HI-STAR 100	MPC-68 (HI-STAR)	MPC-68 (HI-STAR)	HI-STAR 100	4 loaded, canister uncertain
		MPC-68F (HI-STAR)	HI-STAR 100	
HI-STAR 100HB	MPC-HB	MPC-HB	HI-STAR 100HB	5 loaded

[a] loaded canisters as of 9/30/2014, forecast quantities based on continued use of the same system.

TABLE 1. COMMERCIAL OVERPACKS AND ASSOCIATED CANISTERS (CONT.)

Cask system	Canister family	Canister	Transport cask	Total canisters[a]
Standardized NUHOMS	NUHOMS 32PT	32PT-L100	MP197 or MP197HB	14 loaded, canister uncertain
		32PT-L125	MP197 or MP197HB	
		32PT-S100	MP197 or MP197HB	
		32PT-S125	MP197 or MP197HB	
	NUHOMS 32PTH1	32PTH1-L	MP197 or MP197HB	42 forecast, none loaded, canister uncertain
		32PTH1-M	MP197 or MP197HB	
		32PTH1-S	MP197 or MP197HB	
	NUHOMS 61BT	NUHOMS 61BT	MP197 or MP197HB	125 forecast, 23 loaded, remainder uncertain,
	NUHOMS 61BTH	61BTH Type 1	MP197HB	
		61BTH Type 2	MP197HB	
	NUHOMS FC-DSC	FC-DSC	MP187	18 loaded
	NUHOMS FF-DSC	FF-DSC	MP187	1 loaded
	NUHOMS FO-DSC	FO-DSC	MP187	2 loaded
Advanced NUHOMS	NUHOMS 24PT1	NUHOMS 24PT1	MP187	17 loaded
	NUHOMS 24PT4	NUHOMS 24PT4	MP197HB	149 forecast, 33 loaded, remainder uncertain
NAC-MPC	CY-MPC, 26 Assy	CY-MPC, 26 Assy	NAC-STC	40 loaded
	LACBWR	MPC-LACBWR	NAC-STC	5 loaded
	Yankee-MPC	Yankee-MPC	NAC-STC	15 loaded
NAC-UMS	UMS-PWR	TSC-Class 1	Universal Transport Cask (UTC)	60 loaded, canister uncertain
		TSC-Class 2	UTC	
		TSC-Class 3	UTC	
NAC-MAGNASTOR	TSC PWR	TSC PWR	MAGNATRAN	85 forecast, 61 loaded

[a] loaded canisters as of 9/30/2014, forecast quantities based on continued use of the same system.

Figs 3–6 provide photographs and illustrations of the length and diameters of selected vendor systems currently deployed.

FIG. 3. AREVA Transnuclear NUHOMS storage system-Licensed canister diameters and lengths illustrated.

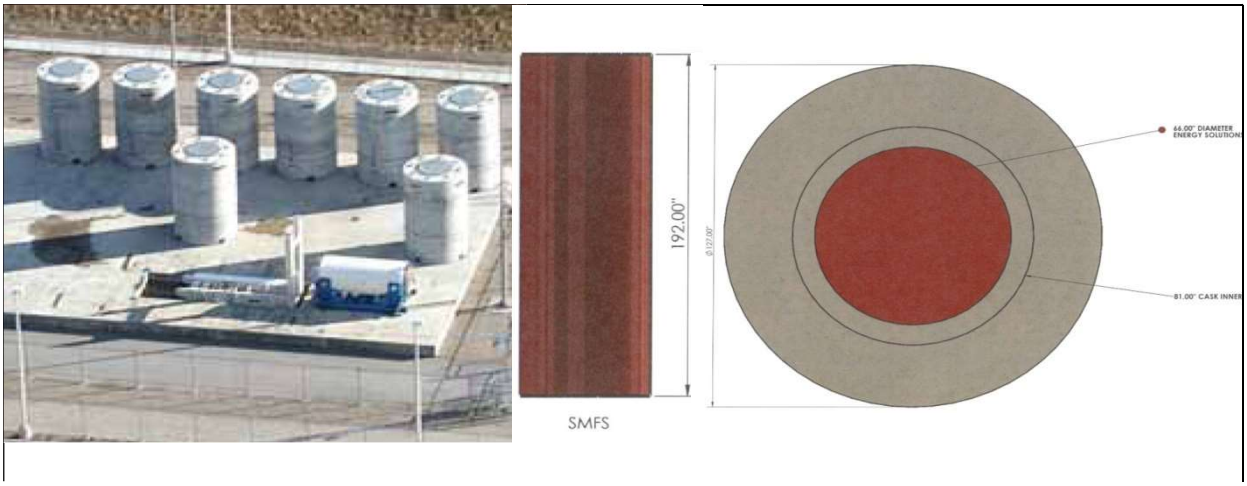

FIG. 4. Fuel Solutions storage system at the Big Rock Point ISFSI-Licensed canister diameters and lengths illustrated.

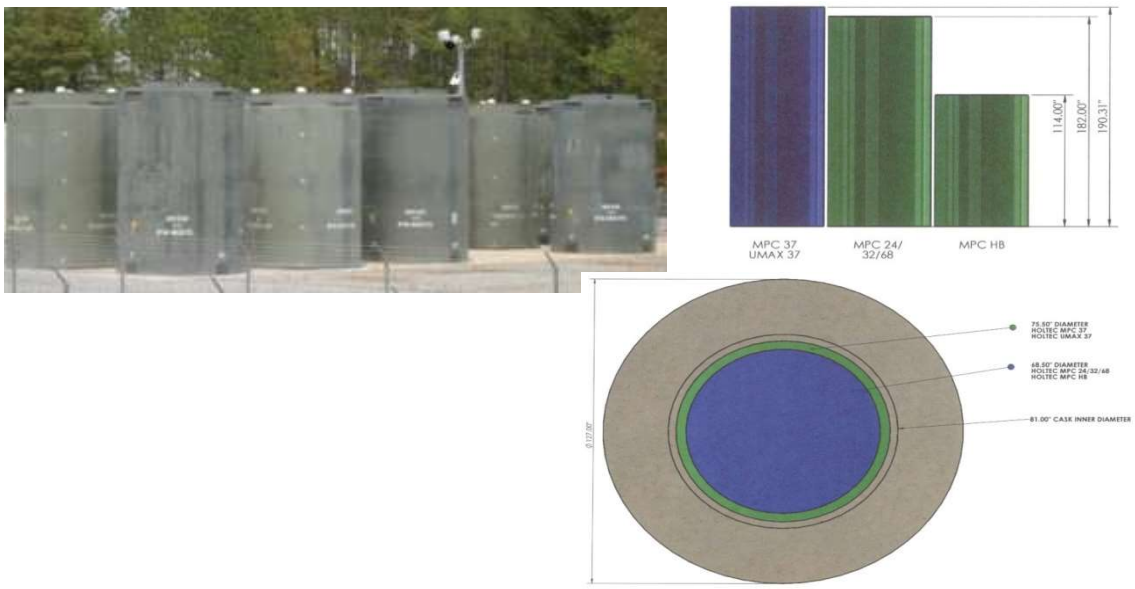

FIG. 5. Holtec HI-STORM storage system at the Plant Hatch ISFSI-Licensed canister diameters and lengths illustrated.

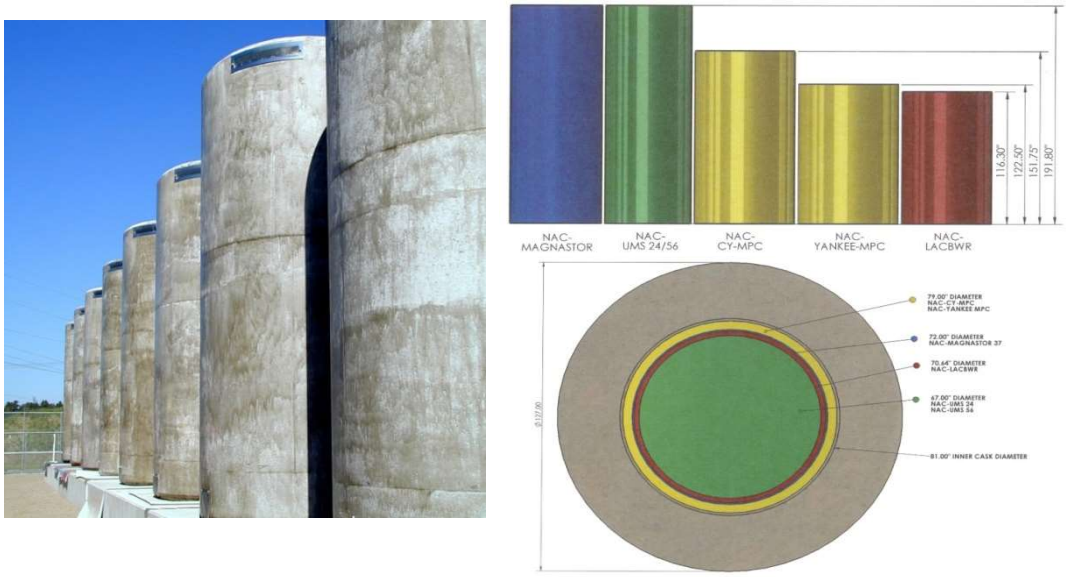

FIG. 6. NAC UMS storage system at the Maine Yankee ISFSI-Licensed canister diameters and lengths illustrated.

While the ISF will be a 'new' facility, DOE recognizes that the welded canisters being placed into storage are existing components and some will have been in reactor ISFSI storage for more than their initial term authorized by the license under which they were stored at the site ISFSIs. Therefore the ISF will need to address aging management issues normally associated with an ISFSI license renewal.

3. TASK ORDER 16

DOE contracted with Chicago Bridge and Iron to develop and document a suite of generic design alternatives for an ISF for the receipt and storage of the SNF and greater than Class C (GTCC) waste. The contract statement of work (SOW) required the contractor to develop concepts for "facilities and infrastructure needed to transfer large dry storage canisters from transportation casks into dry storage. This concept includes building and operating a Canister Handling Building, Canister Transfer Facility, a Storage Cask Fabrication Facility, an Administration Building, Visitors Center, and expandable storage capacity. This storage capacity is estimated to hold 5000 MTHM received at an average rate of 1000 MTHM/year with a maximum rate of 1500 MTHM/year". Concepts were developed to provide DOE with maximum flexibility for additional capacity with planned execution such that expansion could be achieved in an orderly manner.

4. ISF CONCEPT FOR CURRENTLY LICENSED STORAGE SYSTEMS

Use of the currently licensed storage systems is a straightforward application of multiple UNF storage technologies brought together at a common site. For vertical systems, the DPC transfer from the transport cask to the Storage Overpack will require a canister transfer facility or Cask Handling Building (CHB) to simplify dry storage canister transfer operations. For horizontal systems, the transport cask will be unloaded from the railcar and the DPC transfer from the transport cask to the horizontal storage module will be performed at the pad. Fig. 7 provides the conceptual layout for this alternative.

The cask handling operations are impacted by several labour intensive steps that slow the overall throughput process and add radiation doses to workers. These impacts affect horizontal DPC transfer operation to some degree and vertical DPC transfer operations to a larger degree. The contactor developed cask handling building (CHB) concepts to: 1) receive UNF shipments (railcar and transport cask) in an environmentally controlled and radiologically shielded area; 2) provide the facilities to offload transport casks from railcars and place them on the horizontal cask transporter for horizontal systems; or 3) offload transport casks to an radiological shielded area and transfer the DPCs from the transport casks to storage over packs for vertical systems.

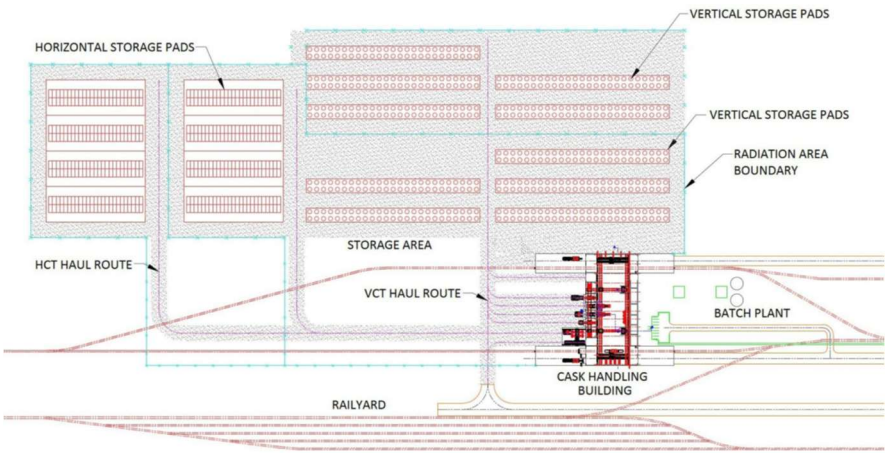

FIG. 7. ISF concept for currently licensed dry storage cask systems.

The CHB concept incorporates automation features to improve throughput and operation as well as reduce radiation dose to workers. The building (Fig. 8) consists of two sets of a rail bay and truck bay servicing 2 vertical type canister transfer cells.

The CHB has two dedicated 200 ton single-failure-proof overhead bridge cranes that can be used across the entire building but can travel independently from rail/truck bay to the transfer cells so that two cask handling operations can be performed at one time. When performing the operations evaluation, it was determined that a single rail/truck bay and 2 transfer cells would accommodate the required throughput of 1500 MTU. The second half of the building is added for redundancy so that any one equipment failure will not jeopardize the required throughput.

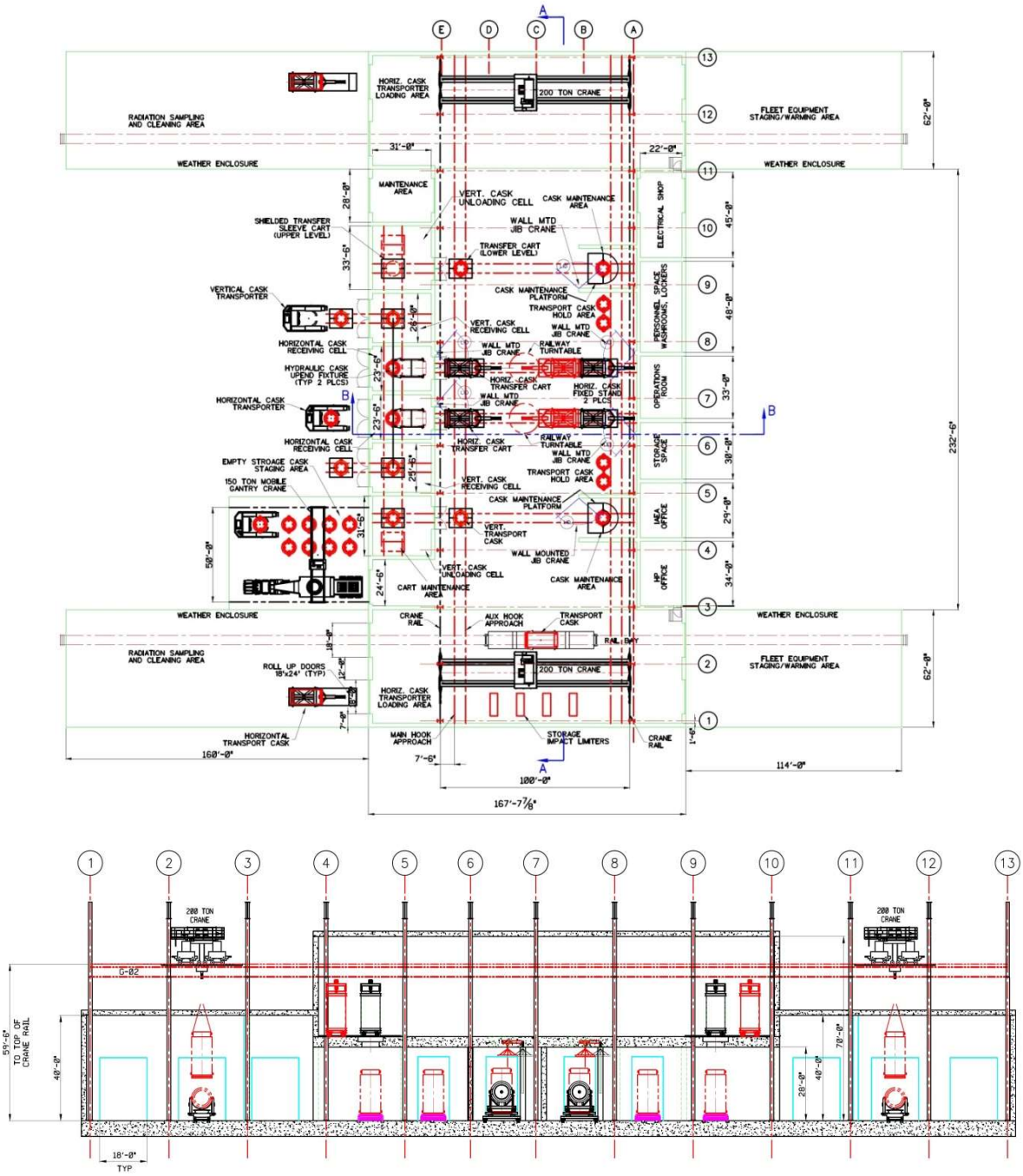

FIG. 8. Cask handling building for improved throughput and reduced dose-plan and elevation views.

5. UNDERGROUND STORAGE CONCEPTS

Another concept evaluates storage of commercial DPCs in an underground silo. Currently only Holtec International provides an underground storage system designated as the UMAX system.

> "*The Callaway nuclear plant is constructing the first UMAX system. The UMAX storage system is an underground UNF storage concept that consists of a thick monolithic block of concrete (low strength concrete sandwiched between a top and bottom reinforced concrete pad) with embedded metallic cavity enclosure containers (CECs). Each DPC is stored individually in one of these underground CECs. Each storage location is individually cooled via passive cooling channels in the UMAX vertical ventilation module (VVM). The VVM consists of the Closure Lid, the Divider Shell and the CEC. The DPC is placed inside the Divider Shell that is concentric to the CEC with air inlets at the bottom. The air inside the Divider Shell is warmed by the decay heat from the fuel inside the DPC and rises and is released from the stack built into the Closure Lid. Cool air is drawn into the VVM via cool air inlets at the periphery of the Closure Lid and introduced into the inside of the Divider Shell via the penetrations at the bottom of the Divider Shell. The inlets and exhaust stacks on the closure lid have been designed to be able to function regardless of wind blowing across the storage site*". [2]

Fig. 9 illustrates the underground storage concepts.

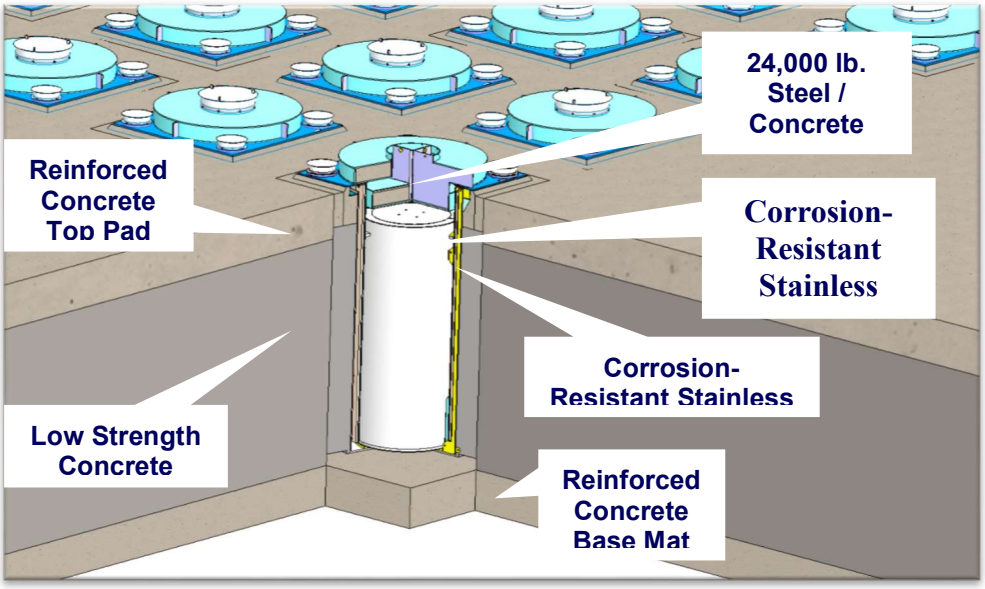

FIG. 9. Underground storage concept.

> "*This approach is more resistant to seismic events than other approaches and is less exposed to environmental and other hazards than above ground approaches. It provides greater shielding for the radiation given off by the UNF in storage since it is underground. The Closure Lid is very substantial and is designed to withstand a wide range of postulated events. This concept also presents a more acceptable appearance to the public than conventional*

above ground SNF storage systems. More importantly, the low profile of the finished storage location makes it easier for the site security team to visually ascertain that no intruders are advancing on the site using the storage over-packs as cover for their movement". [2]

However there are licensing challenges to adapt the wide variety of existing canisters to this concept, including the need to rotate the current horizontal configurations to a vertical orientation. Due to the existing lifting mechanisms the most direct approach to rotation involves a lifting cage and results in the assembly being turned upside down relative to its in reactor orientation. The various canister diameters and lengths also require the use of spacers, to ensure canister tip-over and movement during seismic events are acceptable.

6. VAULT STORAGE CONCEPTS

Another alternative evaluates the use of below grade, air-cooled vaults to store DPCs. Internationally, air-cooled vaults have been used to store SNF and high level wastes from reprocessing plants. An air-cooled vault design was chosen to house the SNF from the decommissioned Fort St. Vrain (FSV) High Temperature Gas Cooled Reactor.

"*A large shielded structure is constructed that houses an array of storage locations into which DPCs from legacy sites can be placed. It has a large service hall covered by an overhead traveling bridge crane. The floor of this hall is the shield structure covering the air-cooled vault. A shield plug is fitted into the floor over each storage location. Below this shield plug is a seismic restraint system that secures the DPC in a way that prevents movement in the event of a seismic event. The vault area beneath this shield floor is designed to encourage passive air flow around the DPCs. Exhaust stacks on one side of the vault allow the air warmed by the DPCs to escape while air inlets on the other side of the vault draw cool outside air into the building. This natural draft system provides bulk cooling to remove the decay heat from the UNF*". [2]

Fig. 10 illustrates the below grade vault system.

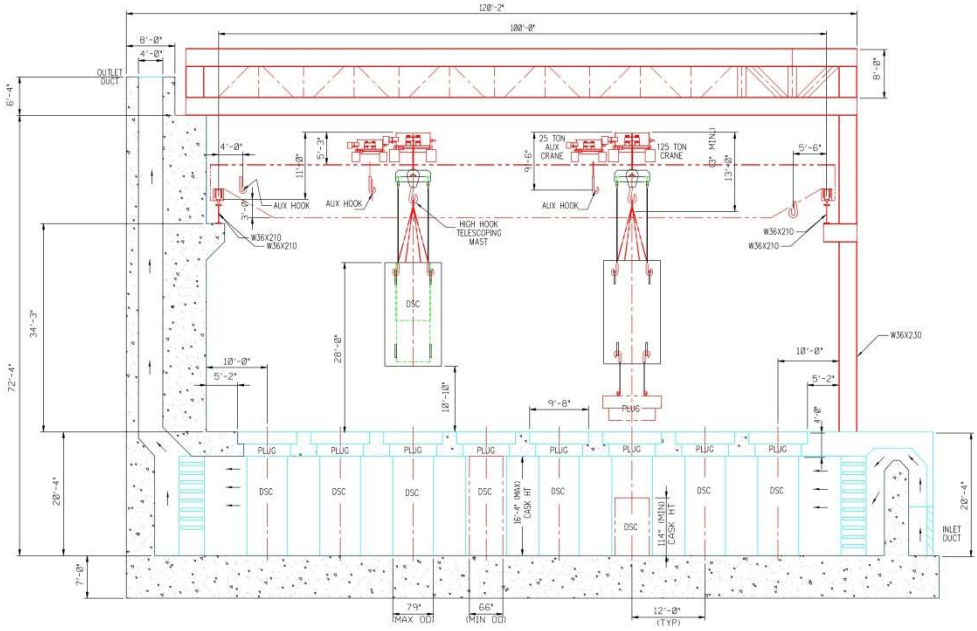

FIG. 10. Below grade vault cross section.

The concept is challenged by licensing issues similar to those described for the underground system.

ACKNOWLEDGEMENTS

The authors acknowledge the Department of Energy, Office of Fuel Cycle Technology support.

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IMPACT OF THE FRONT END OF THE NUCLEAR FUEL CYCLE ON THE BACK END

(Session 6)

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HOW INNOVATIVE APPROACHES AND TECHNOLOGIES THROUGHOUT THE FUEL CYCLE ARE SUPPORTING NPP OPERATIONS WHILE ANTICIPATING FUTURE BACK-END CHALLENGES

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Abstract

Used fuel management is more and more one of the major topics that nuclear countries and utilities have to face when managing existing nuclear power plant as well as new build. Stakeholders concerns are growing in parallel of the steady increase of used fuel inventories due to fading away of the back-end endpoint in most nuclear countries. Utilities are additionally being challenged further with new operation conditions to accommodate Fukushima-driven safety requirements and to adapt to new grid demand due to growing share of renewables, intermittent energy by nature, while anticipating the ever-growing competition in electricity liberalized market. To respond to this challenging environment and combination of constraints, AREVA has never stopped over the past few years developing and implementing new technologies at every step of the fuel cycle. Starting from the Front-End, through enrichment process and innovation in fuel design and fabrication as well as Core design and management optimization with drastic reduction of used fuel discharged per electricity production and even direct re-use of earlier discharged used fuel with both corollary uranium savings and decrease of used fuel inventory and environmental impact. If the innovation and new technologies in the very reactor operations have helped also responding to the above mentioned combination of constraints and requirements, innovations in the Back-End (BE) of the fuel cycle have also helped to respond to the challenging environment whereas through advanced innovative solutions in a specific area, where used fuel inventory as well as plutonium balance of a utility are both brought to a minimum down to zero. This paper/presentation will briefly describe challenges and issues at stake and present and illustrate through industrial scenarios how some AREVA's innovative approaches and technologies throughout the fuel cycle are safely and economically supporting NPP operations while anticipating future back-end challenges.

1. INTRODUCTION AND ISSUES AT STAKE

The present decade has already brought, and will bring, continued challenges to all nuclear electric utilities. New regulations and strategies targeting power plant various parameters squeeze existing facilities whilst economics continue to favour inexpensive natural gas and, at some extent, renewables thanks to proactive national policy, over nuclear for new power generation.

In a depth analysis, the 'Centre of Climate Change and Energy Solutions' [1] warned that policies that promote the development of "low production cost renewables" and "low natural gas prices" are putting pressure further on the ability of existing nuclear reactors to compete in the market place.

Moreover, utilities are being challenged further with new operation conditions to adapt to new grid demand due to growing share of renewables, intermittent energy by nature, while anticipating the ever-growing competition in electricity liberalized market.

This is shown in the first nuclear power country in the world, the US, where this country has experienced a rapid increase in natural gas production from shale and being nowadays the largest producer of natural gas in the world while owners of NPP have voiced growing concerns about the profitability of their units, sparking discussion of possible additional nuclear retirements than those, the five reactors, representing 4.2 GW of capacity retired by 2015 [2].

In this fast-changing world, used fuel management, remains one of the major topics that nuclear countries and utilities have to face when managing existing nuclear power plant as

well as new build. Stakeholders concerns are growing in parallel of the steady increase of used fuel inventories due to the fading of the endpoint of back-end (BE), namely Reprocessing-Recycling and Deep Geological Repository within most nuclear countries. Additionally, operators and Waste Management Agencies are challenged further with new operation conditions to accommodate Fukushima-driven safety requirements.

Preserving the existing nuclear fleet as long as possible along with nurturing the development of new build are both pivotal elements in the transition to a low-carbon future and the Nuclear Industry has to respond to this challenging environment and combination of constraints. Solutions with innovative technologies, products and services, throughout the fuel cycle at each of it phases and steps within each phase are key to take up present and future challenges

2. TAKING-UP PRESENT AND FUTURE CHALLENGES

The different phases and their associated steps throughout the nuclear fuel cycle are closely interrelated. Decisions, considerations and available options and solutions at each phase not only may be constrained by those taken or chosen at previous phases but also may affect those of subsequent phases.

Therefore a key challenge in achieving effective support to current NPP Operations while anticipating future back-end challenges with respect to safe, effective, and sustainable used fuel is that overall responsibility for final disposition of SF must be clear. It enables decisions that consider the entire life cycle and discourages local optimization achieved at the expense of overall system performance.

Recognizing this fact, the Council of the European Union adopted in 2011 “a first European directive establishing a community framework for the responsible and safe Management of spent fuel and radioactive waste” [3]. It directs EU Member States to establish and maintain a national legislative, regulatory and organizational framework for used fuel and radioactive waste management that allocates responsibility and provides coordination between relevant competent bodies. It further directs that national policies take into account the interdependencies between all steps from the design and fabrication of the fuel, through its irradiation during electricity production up to generation and management of used fuel and radioactive waste.

The next section briefly describes challenges and solutions of a number of actual industrial cases where upfront decisions and innovations brought by AREVA related to one or more phases are designed to affect positively those within other phases of the fuel cycle and definitely the BE end-point.

Integration across design, operational, regulatory, and policy aspects of fuel cycle management have enabled AREVA to come up with decisions that ensure compatibility of materials, optimization of operations, innovation in equipment, optimization of packaging and waste form with existing and upcoming used fuel in each of their management steps, thus reducing associated risks and life-cycle costs.

3. ACTUAL INDUSTRIAL CASES

3.1. Fuel design and fabrication case

The challenge: As nuclear power industry continues to improve performance, upgrades may reach their maximum achievable impact within the performance and limitations of some current licensed fuel cladding and assembly.

As an example, PWR operators' requirements are increasingly demanding and include, but not limited to, power uprates of up to 20% or beyond in some cases, maximum batch average burn-up increased from 40 to ~60 MWd/kgU, lower neutron leakage with cycle lengths of up to 24 months and challenging water chemistry conditions, extended storage period.

All these factors affect the available margins of the fuel assembly which must remain robust against grid-to-rod fretting (GTRF) — the most frequent cause of fuel failure in PWRs worldwide — while enhancing thermal hydraulic performance and also after discharge and storage period and following transportation to the BE interim step or end-point, namely disposal or reprocessing and recycling.

Moreover, the need to provide safety margins regarding new regulatory rules worldwide for RIA and LOCA class IV accidents might become more and more challenging for some type of fuel.

The Solution: Improvements in nuclear fuel are key drivers of nuclear power plant operational performance, safety, and economics as well as all steps of used fuel management from interim storage, transport and reprocessing or disposal requirements.

Developed to meet the increasingly demanding requirements of utilities, GAIA, the new AREVA Generation Fuel for 17 × 17 PWR fuel assembly ensures the safety and robustness of fuel operations, while offering optimal performance and a high burn-up capacity [4].

GAIA enhanced reliability and robustness through higher thermal performance and fretting resistance, high debris filter efficiency, high grid stability, high fuel assembly dimensional stability.

GAIA fuel assembly boasts M5® advanced cladding which offers:

- Superior margins and meets anticipated changes in the U.S. regulatory requirements;
- Ready for used fuel reprocessing and meets all requirements for recyclability of recoverable materials into a new fresh fuel and minimization of remaining high level waste, mainly fission products and cladding materials, through qualified robust and safe conditioning (refer to section 3.4).

Furthermore, M5® higher resistance to corrosion and extremely low hydrogen uptake (Figs 1 and 2) are both key to flexibility in used fuel management which:

- Allows safer retrieval from extended interim storage and following mandatory transport prior to the following phase, namely disposal or reprocessing;
- Enhance used fuel integrity which and minimizes risk during storage or risks handling operations whilst avoiding questioning predisposition up to directly dispose of used fuel.

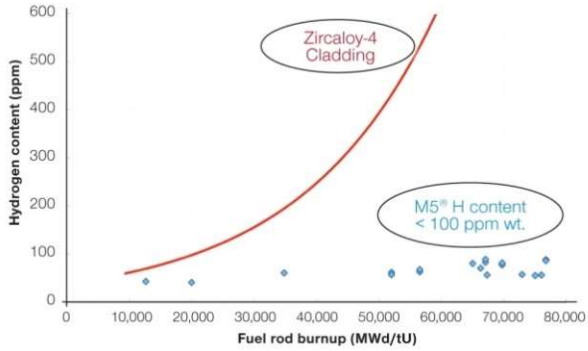

FIG. 1. A factor of 4 less thickness at high BU and no increase in data scatter with increasing irradiation demonstrates the stable microstructure and low sensitivity of M5® to PWR duty power and interim storage.

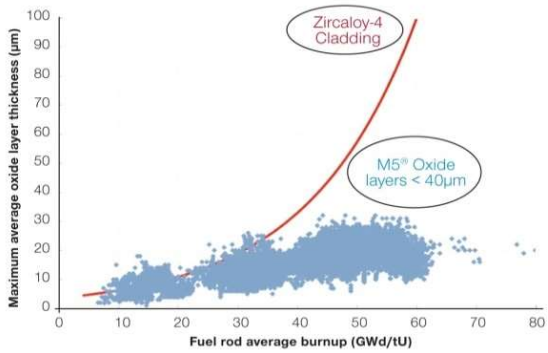

FIG. 2. The hydrogen pick-up of M5® is significantly lower than Zircaloy-4 and other Zirconium alloys, providing for greater ductility needed during power operation and used fuel management.

Conclusion: The best way to manage spent fuel or waste is to reduce them at low as possible. This is why nuclear fuel robustness and reliability avoids waste generation throughout nuclear operation at NPP and up until back-end end-point by avoiding discharging fuel prior to reach the expected burn-up and thus avoiding fabricating a new fuel to replace the former.

New material for cladding supports higher performance and flexibility during operation needed to respond to increasingly demanding electricity market and regulations. Furthermore it increases safety margins and reduces risks going along used fuel Management during all the steps following the discharge from the core into the spent fuel pool (SFP) [5].

3.2. Passive fuel decay heat removal technology solutions

The Challenge: Storage of used nuclear fuel in water pool has been a standard practice since nuclear reactors first began operation around 70 years ago. Today, SFP at reactor is still the starting point for used fuel management strategy including all interim storage technologies.

With NPP life time extension spreading and delay in implementation of back-end strategy, most NPPs are challenged by limited storage space at SFP.

In addition, Fukushima lessons learned have shown that SFP, the starting point of used fuel management, were an aggravating factor in the very difficult emergency situation faced by the nuclear operator and rescuers which in response led safety regulators across the world during the following years to more and more stringent safety requirements at existing SFP at reactor in operation or permanently shut down.

Overtime, SFP operating and maintenance costs at stake with the end-of-life operation approaching and after shutdown of the reactor.

The solution: In order to accommodate used fuel inventories while responding to stringent safety requirement and demonstration, AREVA never stopped to offer solutions to ship off used fuel from SFP, including to reprocessing and recycling plants. As a matter of fact, by removing discharged used fuel in reactor pool at the earliest, the reprocessing and recycling option, the proven used fuel management option with more than three decades of excellent safety and operational track record, helps as of today the nuclear industry in implementing measures to reduce radionuclide inventories in SFPs and ultimately to enhance racking configuration safety and consequently grace period in case of loss of electricity.

However, aware of the issues and timeline needed prior to implement a new option as regard to used fuel management, AREVA has also developed qualified and successfully implemented and passive systems, technologies and solutions [6, 7] to remove decay heat from used fuel stored in dry or wet conditions even in the case of loss of electrical power. Key arguments for passive systems are simplicity, inherent safety and maintenance and surveillance costs reduction.

Spent fuel storage and more particularly decay heat removal in a wet condition is a prerequisite to all spent fuel management option or strategy as the reactor pool is the starting point and step to all existing spent fuel management.

The Fig. 3 gives a typical example of proven technologies implemented by AREVA since 2008 [4] that help implement passive removal systems solution at the very early time within the spent fuel management, namely at nuclear power plant or at the following step in a centralized storage pool, where the risk of system failure is reduced by eliminating all components in the cooling chain requiring external power sources or forces (active components). Moreover, cooling efficiency increases automatically with rising temperatures.

Additionally, passive systems allow considerable savings during operations, as no active components requiring power are needed. Since no mechanical or electrical parts have to be maintained or tested thus very low maintenance and almost no other direct operating costs are expected as well as allowing dose reduction.

The relative savings increases with end-of-life operations approaching, post-operation and ultimately dismantling phases.

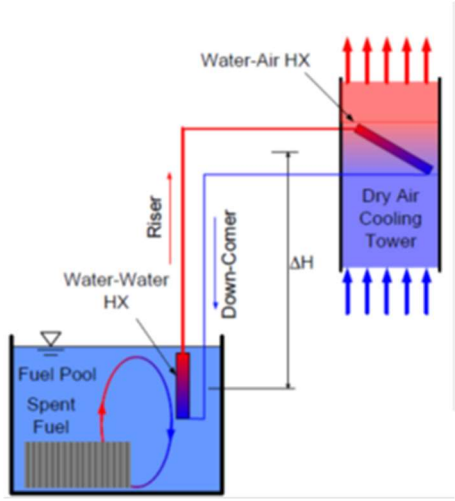

FIG. 3. Operation principle of passive cooling systems.

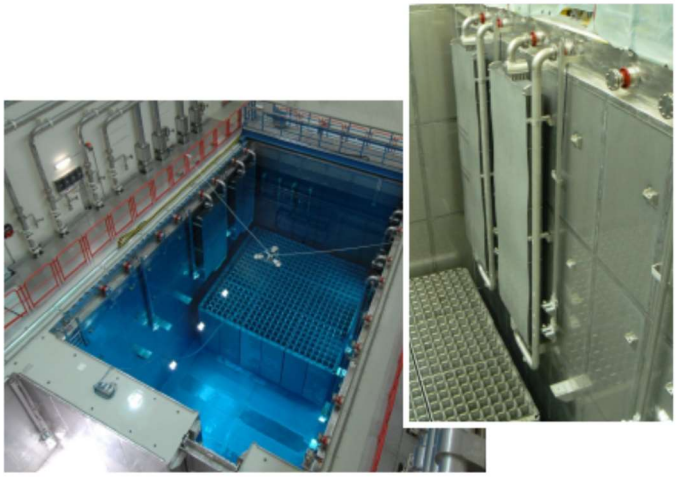

FIG. 4. AREVA water-water heat exchange.

Conclusion: If reprocessing option definitely contributes to the reduction of decay heat as well of radionuclides inventory at the nuclear plant through the early removal of the spent fuel discharged in pools , it is pivotal however that the nuclear industry keep enhancing all the parameters that contribute to enhance the safety and reduce the risks through technical innovations and systems that contributes to enlarge grace period during abnormal and accidental conditions in interim storage systems as well by better managing the decay heat removal through solutions with system and mix of technologies.

However, to respond to various requirements and with industrial cases implemented since 1990 [6, 8] (CASCAD Passive heat removal system in Vault type at Cadarache, France) and Wet passive heat removal system since 2008 [7] (at Goësgen, Switzerland), AREVA never stopped enhancing further centralized and high capacity storage solutions with passive cooling. This is still currently illustrated through various technology and system innovations that contribute to the Management of Decay Heat removal at wet storage through advancing and testing new technologies and systems (such the passive multi-phase [9]) to respond to more demanding operational and safety requirements. And these demanding requirements starting as early as since SFP at reactor.

3.3. Recycling used nuclear fuel

The challenge: the broader Nuclear Industry issue concerning nuclear used fuel management options and strategies, namely that used fuel was, and is still perceived as one of the crucial unresolved issues when referring to nuclear energy.

Additionally, utilities with no perspective of new builds and with shutdown foreseen in a few decades are left with used fuel inventory while the perspective of a deep geological repository implementation is fading away in the country.

By removing discharged used fuel and reducing their inventory, and consequently radionuclide inventory at the NPP at the earliest, recycling definitely helps countries and

utilities. However limited fleet within a country coupled with reactor lifespan raise the difficulty for complete materials recycling.

The solution: In cooperation with the plant operator, AREVA has developed and implemented an advanced recycling scheme, named 'Precycling', which enables the recycling of used fuel from various reactor fleet owners, including utility with a single reactor or small reactor fleet with limited lifespan.

Precycling consists in an anticipated use of MOX fuel (Mixed Oxide Fuel) prior the implementation of reprocessing (Fig. 5) of used fuel and recovering of plutonium needed for MOX fuel fabrication (Fig. 6).

FIG. 5. The precycling scheme — Step1/2 Pu loan for MOX fuel fabrication and reactor operation.

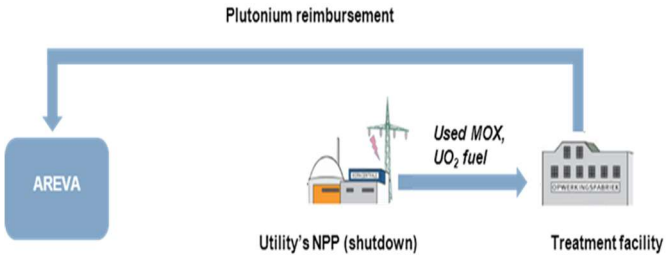

FIG. 6. The recycling scheme — Step 2/2 used fuel, including irradiated MOX Fuel, is reprocessed and plutonium recovered and returned to AREVA.

Precycling consists in an anticipated use of MOX fuel prior the implementation of reprocessing (Fig. 5) of used fuel and recovering of plutonium needed for MOX fuel fabrication (Fig. 6).

Using the customer utility's current installed fleet as a plutonium burner of its own plutonium by loading enough MOX fuel, in the very same reactor that have produced the plutonium during irradiation of the uranium fuel, utility's plutonium balance as well as used fuel plant's inventory can be brought to zero (Fig. 7) in the remaining NPP operating lifetime [10].

FIG. 7. Intrinsic Pu consumption during MOX Irradiation. Figures given are Core's Pu balance sheet at the end of every 3 cycles, based on equivalent 900 MW(e) power reactor.

Conclusion: Precycling, is an innovative approach to the reprocessing and recycling option for closing the fuel cycle for countries or utilities with any type of fleet size, including a single or small reactor fleet, until the end of life its (or their) plant(s) with limited lifespan. The innovation here is based on the early and sufficient plutonium consumption (through using above 30% in MOX rate in the Core) to offset the plutonium production during fuel irradiation and electricity generation in NPP (Fig. 7).

This is definitely safer and optimized used fuel management that reduces reactor front-end requirements, drastically reduces SFP inventory, brings to zero the balance of used fuel and fissile materials at end-of-life of NPPs. This management leves only compacted and vitrified waste to be disposed of at reactor end-of-life and reducing consequently and drastically daily requirements (IAEA safeguards, limited environmental footprint, easy handling, transportation and storage…) as well as future liabilities with regard to deep geological repository and all pre-disposal steps in between.

3.4. Waste

The challenge: The global challenge in the field of waste management could be entitled 'ambitioning a zero radioactive waste future for nuclear energy'. The proposed strategies have however to be stuck in reality taking into account the technical and economic conditions of the moment but also the availability of existing and proven technologies. To serve this ambition, they need to be enriched not only through relevant and timely innovation and R&D programs but also through practices and culture changes in the way of operating plants.

Ensuring that every waste generated has or will have a final disposal solution is part of the challenge. The principle of disposal consists in isolating the waste from people and the environment for very long periods of time. Most of Nuclear countries have chosen to set up a long-term solution based on disposal, and more specifically based on a deep geological repository when it comes to HLW or ILW-LL. However, the complexity in the implementation of repository projects especially has made them a scarce resource worldwide. The complexity lies in the interconnection of technological, environmental, economical and societal challenges.

Based on these considerations, the permanent challenge concerning the long term management of radioactive substances, one of the most important of which being used fuel, lies in defining the best compromise between reduction at source, potential treatment (including characterization, sorting, processing), recycling, storage and disposal.

Along with safety which remains the top priority, parameters including the scarcity of repositories thus their costs, the availability of efficient processes and technologies, the ratio 'cost of treatment (in its most general sense) versus cost of repositories' are part of the global equation enabling to select the routes connecting the NPPs, the fuel cycle facilities and the repositories. The challenge then consists in striking the right balance between these parameters, and thus specifying what could be the most appropriate status for all managed substances.

The solutions: AREVA advocates for 'zero waste' strategies within its own fuel cycle facilities and sites but also for serving other players in the nuclear industry. This ambition, more than a target, lies on the design and implementation of solutions aiming at maximizing recycling, while reducing waste to the minimum, even eliminating them.

Maximizing recycling to reduce waste from NPP operations through the 'Universal Canister Strategy':

AREVA has implemented recycling capabilities associated with an optimized ultimate waste management with the so-called 'Universal Canister' (UC) strategy. This strategy based on recycling and waste conditioning standardization by using a single type of package for long lived waste, leads to important savings in terms of volume (up to 5 compared to direct disposal) and long term radiotoxicity (up to 10 compared to direct disposal). This ultimate waste consists mainly in fission products and minor actinides traces, as well as structural assembly parts.

The fission products are incorporated into a stable borosilicate glass matrix and conditioned in the UC-V (Universal canister for vitrified waste, ensuring stability and containment during a very long period (over than 100 000 years).

The structural assembly parts, hulls and end pieces, are compacted and conditioned in UC-C (Universal canister for compacted waste — Fig. 8). Typically, the recycling of one ton of used fuel leads to approximately one UC-V and one UC-C. One of the challenges is to further reduce the ratio UC/tHM taken into account the used fuel evolution including the increase in Burn-up.

The UC strategy provides the ability to rationalize the global waste management from reception and storage of waste to final disposal. On-site handling, transport operations, and geological disposal selection, design and construction are indeed facilitated thanks to the packaging uniqueness with smaller volumes and reduced thermal constraints. The resulting costs are therefore lowered.

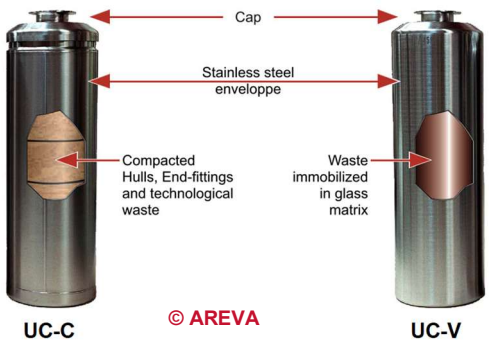

FIG. 8. Universal canisters for compacted hulls and ends (UC-C) and vitrified waste (UC-V).

The UCs does not contain IAEA safeguarded materials avoiding thus safeguards to be implemented for storage and repository. This feature consolidates Safety regarding notably criticality and contributes to further enhance proliferation resistance.

In addition, this strategy is suitable for a progressive implementation of final disposal through a safe and simplified interim storage. Indeed, the related interim storage facilities are currently licensed for about 100 years. This duration is fully consistent with the operation of the deep geological repository which should last, in the French case, over 100 years. The storage duration is also an important parameter in the minimization of the repository footprint, taking into account the decrease in thermal power of vitrified residues.

The UC strategy complements the recycling strategy in an economically sound way [11]. Solving the waste issue has become the prime contributor to nuclear program sustainability: recycling with practical and already visible steps in the minimization of the nuclear legacy to the future generations through the UC strategy contributes to making the final waste issue solvable.

In the French case, the geological repository should have a footprint of around 15 km² [12] taking into account that the whole used fuel inventory would be recycled. The shift in the policy from recycling to direct disposal would lead to a significant increase in the volume of HLW to be stored (90 000 m^3 instead of 10 000 m^3) and thus to a significant increase in the repository footprint (25 km^2).

Minimizing waste from fuel cycle facilities:

Along with used fuel recycling dealt with previously, AREVA's range of nuclear activities covers the entire fuel cycle from mining, extraction, conversion, enrichment, manufacture and recycling, to logistics and storage services. The development of all these activities has been anchored in a policy for the systematic reduction of volumes and harmfulness of the waste based upon 5 golden rules, associated to a set of solutions.

3.4.1. Avoid waste at source from the design to the operation of equipment and facilities

The best waste being the waste that hasn't been produced, the efficiency in the reduction or even the elimination of waste at source has been significantly improved starting from the design stage.

For example, the latest vitrification equipment implemented at La Hague, based on the cold crucible technology, has been designed to significantly reduce the technological waste. Indeed, the lifetime of the cold crucible is 10 times longer than that of the melting pots because its cold structure is not sensitive to corrosion, protected by the layer of solidified glass formed between the structure and the glass bath.

During operation, the implementation of simple rules has led to significant improvements. Only bring equipment that is strictly necessary for the operations into the installation, avoid packaging, restrict consumables are examples of these rules.

Strengthen the zoning of facilities with a systematic assessment of the waste zoning relevance; as shown in Fig. 9. The outcome has been a significant reduction of the volume of radioactive wastes which were 'declassified' into conventional waste.

FIG. 9. Waste zoning.

3.4.2. Maximize sorting and decontamination

The disposal cost being incremental with the depth of repositories, the optimization of the distribution of waste between management routes (Fig. 10) is also part of the challenge, allowing a waste to be accepted in a less costly management route that it was initially intended for.

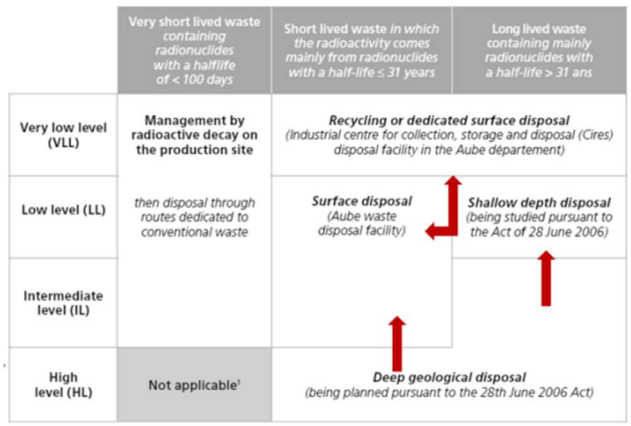

FIG. 10. Distribution of waste between management routes in France.

The ultrasonic treatment of plutonium oxide empty cans at the AREVA MOX Fuel Fabrication Plant is an illustration of the implemented 'decategorization strategy'. Indeed the treatment of the cans leads to a significant decontamination. This will make the ultimate waste resulting from the conditioning of these cans suitable for surface repository instead of deep geological disposal.

3.4.3. Reduce volumes and concentrate the activity

A set of solutions developed and implemented within AREVA facilities for 40 years have proved their efficiency in terms of volume reduction and activity concentration, notably based on:

- Cutting;
- Compaction;
- Incineration;
- Fusion;
- Cementation / Vitrification.

As illustrated, their implementation at La Hague Recycling plant has led to six fold reduction of the HLW and LL-ILW volume in 20 years (Fig. 11).

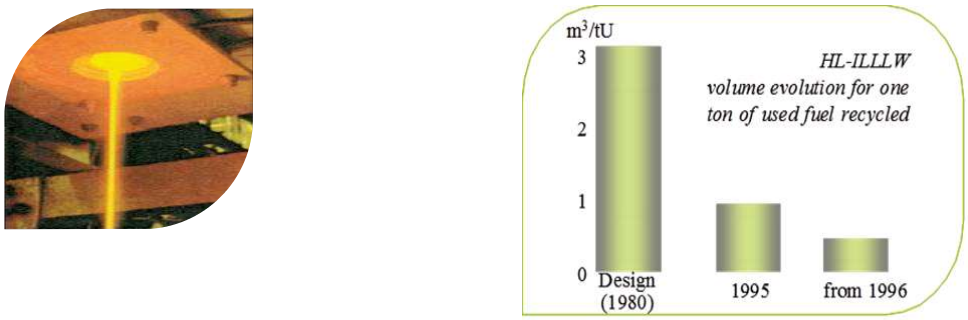

FIG. 11. Illustration showing the reduction in the volume of HL and LL-ILW in reprocessing one ton of used fuel at the La Hague recycling plant.

3.4.4. Condition at the earliest

Safe and proven technologies solutions resulting from decades of R&D and industrial operation do exist. Safety and the overall efficiency in the management route can be further enhanced by resorting to the existing solution in order to condition routine waste or legacy waste at the earliest.

3.4.5. Initiate the relevant R&D at the right time

R&D and innovation have always been and shall remain a cornerstone in the management of waste with two main orientations:

- Innovate for legacy waste and for the already planned inventory by adapting existing routes with the aim at increasing performance, or, by creating new routes only when necessary ;
- Pursue R&D efforts for long term nuclear evolution ambitioning a 'zero radioactive waste' future.

As an example, the effluents stemming from La Hague plants operation have been for longtime bitumized. AREVA has optimized their management by increasing their recycling within the process and sending materials with residual activity to vitrification plants in order to be incorporated into standard 'UC-B' containers of vitrified residues belonging to the Universal Canister family. The cold crucible melter commissioned in 2010 is at the heart of the optimization. Resulting from 25 years of R&D in close cooperation with the French research organization CEA, the cold crucible melter allows waste to be vitrified at very high temperatures over 1200°C, compared to a maximum of 1100°C with the hot crucible melter accommodating the vitrification facilities. The cold technology increases the lifespan of the crucible, allows for doubling the output of the La Hague plant, and can vitrify both a wider range of waste materials and more corrosive waste. It allows for example the conditioning of the UMo fission products from the processing of the first fuel from the graphite-moderated gas-cooled reactors at the La Hague site, with highly corrosive features due to the significant level of molybdenum. Some wastes do not currently enter in any of the existing or projected final management routes because of special physical or chemical characteristics. Designing and implementing a dedicated processing/conditioning is part of the challenge. They currently represent less than 0.3% of the total French waste inventory (all French operators). This is for example the case of spent oils and solvents for which AREVA is currently assessing solutions.

Specific efforts are also made by AREVA in association with the CEA in developing solution for conditioning technological waste with an alpha-prevailing spectrum of metallic and organic materials; these waste result notably from the operation of the AREVA La Hague and Melox plants. Similar waste is still being produced. Fusion —Incineration — Vitrification are the processes considered in the related R&D program.

4. CONCLUSION

This paper has presented briefly through a few industrial cases how innovation in various steps of the fuel cycle are responding to existing and evolving requirements, whether innovations in the approaches and/or in technologies the paper has also showned how AREVA, through bringing solutions throughout the Fuel Cycle, is safely and economically supporting NPP operations while anticipating future back-end challenges to utilities and countries for a better management of the back-end of the fuel cycle.

Maximizing safety, minimizing the overall environmental impact, in particular the geological repositories footprint, increasing operational efficiency, thus reducing the overall cost, and enhancing public acceptance have been and remain the main drivers for taking-up the challenges in the field of waste and materials management.

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SPECIFIC ASPECTS OF HIGH BURNUP OR MIXED OXIDE FUEL RODS DURING DRY STORAGE

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Abstract

Besides the long term behavior of the fuel itself the cladding behavior is an important issue for interim and long term storage of spent fuel in casks. In dry storage the fuel rod cladding is important for the retention of fission products. It is still the first barrier even if the storage casks are designed to contain the fission products during the designated storage period. In Germany today the period for dry storage is limited to 40 years starting with the loading of the cask. To ensure the cladding integrity the cladding hoop stress and the cladding hoop strain as well as the maximum temperature during drying and storage are limited to certain values. High burnup fuels as well as mixed oxide fuels have some properties that have to be considered in dry storage with respect to cladding integrity especially when the storage time has to be extended. An extension of the dry interim storage time beyond 40 years might be necessary due to the lack of a final repository. Both high burnup fuel and mixed oxide fuel have a higher fission gas release and a slower decrease of the post irradiation decay heat. Therefore, the number of high burnup or mixed oxide fuel assemblies is limited in most cask loading patterns. Further aspects of these kinds of fuel which should be addressed with respect to dry storage are the influence of the high burnup structure and the cavity pressure of the pores of the fuel. Whether these effects have an influence on fuel degradation and on the cladding integrity during the dry storage is not examined up to now. The influence of high burnup or mixed oxide fuel on other cladding degradation mechanisms like the inner and outer oxide layer, hydrogen pick up, irradiation damages in the cladding material crystal structure are well understood during fuel operation and pool storage, but the long term consequences still aren't verified due to the lack of experience and long term experiments. One approach to consider these effects in the future is to extend existing fuel performance codes with appropriate models and the consideration of all stages of the fuel life from the more short term operational behavior to the long term storage time. Within such assessments conservative as well as best estimate approaches should be considered. For long term predictions deterministic and probabilistic methods should be used in parallel.

1. INTRODUCTION

The amendment to the German Atomic Energy Act [1] of 2002 excluded the reprocessing of spent fuel from German power reactors. With this act the fuel cycle was changed from a closed cycle to an open cycle at the back end. The only legal treatment of spent nuclear fuel is the final disposal. Since a final repository was (and is) not available, spent fuel has to be stored in casks if the wet pool storage capacity at the power plants is exhausted or the power plant is to be decommissioned. The utilities were legally bound to build interim storage facilities at the sites of their nuclear power plants. Together with the three central interim storage facilities 16 interim storage facilities are operating today in Germany.

According to the legal phase out the last German nuclear power plant will shut down for decommissioning in 2022. After that a number of round about 2000 casks with spent fuel have to be stored in the storage facilities. Today dry storage of spent nuclear fuel in Germany is limited to 40 years from the loading of the fuel into the casks. The debate on a final repository started again from scratch with the new act for the search and selection of a site for a final repository for heat generating radioactive waste from 2013 [2]. According to this law the nuclear waste repository location selection process should be finished in 2031. The commissioning of the repository is optimistically foreseen for 2050. This process will inevitably result in an extension of the interim storage of spent fuel in casks from the current 40 years to maybe 60 years or more. There is no experience on fuel behaviour for such a long time.

An extension of the storage period was already considered in the recommendation on dry interim storage of spent fuel and heat generating radioactive waste in casks [3] of the German Nuclear Waste Management Commission (ESK) in 2012. The ESK considers in their guidelines a storage period of 40 years but states also in case this period is not sufficient then additional and appropriate verifications on long term behaviour of the casks and the fuel have to be provided. Within these verifications the available experience can be considered. In this paper we want to discuss some aspects that should be addressed for an extension of the dry storage period in casks beyond 40 years with respect to high burnup and mixed oxide fuel from the point of view of TÜV NORD, who is acting as a technical support organization on behalf of the regulator. With the attribute "high burnup" fuel with a burnup higher than 55 MWd/kgHM is addressed.

2. REQUIREMENTS ON FUEL AND CLADDING FOR DRY STORAGE

The IAEA Safety Standard on Storage of Spent Nuclear Fuel [4] gives generic requirements on dry interim storage. In Germany the requirements from this standard are implemented and formulated in the ESK guidelines [3]. From these generic requirements technical and material specific criteria have been derived. These specific criteria are normally base on experiments and include safety margins. Beside the cask itself the fuel matrix and the fuel rod cladding are stipulated as barriers for fission product retention. For the safe enclosure of fission products a systematic fuel rod cladding failure must be excluded during the storage time by limitation of the cladding corrosion, cladding hoop stress and cladding hoop strain to material specific values. Cladding corrosion during the storage is caused by the remaining water after the internal drying of the cask. The cladding hoop stress is limited to 120 MPa and the cladding hoop strain to 1%. A more detailed description on these criteria is given in [5].

For the fuel, here e.g. the uranium oxide pellets, mixed oxide pellets or the coated particles, no specific requirements are formulated. The fuel matrix is mentioned as a barrier for the retention of fission products. Following the common understanding a part of the fission products are released to the free volume of the fuel rod during the irradiation at higher temperatures. After the in pile life of the fuel no significant release from the fuel is assumed. The amount of gas in the free volumes of the fuel rod is the sum of filling gas from manufacturing and the released fission gas during the irradiation time. Together with the geometry of the free volumes of the fuel rods (e.g. plenum and gap volumes) and the temperatures of the free volumes the amount of gas gives the inner fuel rod pressure. The inner fuel rod pressure is the driver for the hoop stress in the fuel rod cladding.

High burnup fuel has a greater inventory of gaseous fission products and a higher fission gas release from the fuel. This leads to higher inner fuel rod pressures at room temperature which is the reference to calculate the inner fuel rod pressure under storage conditions using the temperature profile from the positions of the fuel in the casks. Additionally, high burnup fuel as well as mixed oxide fuel has a higher post irradiation decay heat and a slower decrease of the decay heat due to the higher content of actinides. Mixed oxide fuel does not differ much in the gaseous fission product inventory from pure uranium oxide fuel with the same burnup at overall high burnup levels. Therefore, the number of high burnup or mixed oxide fuel assemblies is limited with respect to their decay heat power and radiation in most cask loading patterns.

Cladding integrity is besides the retention of fission products important for several safety aspects. Most calculations carried out for the dry storage like calculations on criticality and heat transfer assume an intact geometry of the fuel. The failure of fuel can be divided into two scenarios.

In the first scenario the cladding of the fuel is getting untight and the gaseous inventory, which is not trapped within the fuel matrix, will be released into the casks free volume atmosphere. This will impair the thermal conductivity of the casks inner atmosphere which will lead to higher temperatures of the inventory. This scenario can be neglected if only a few fuel rods are failing. A larger portion of failed rods, caused by a systematic failure mechanism, is not considered in the assessment of the thermal behaviour of the casks and their inventory. This would also increase the inner gas pressure, which is under normal condition lower than the atmospheric pressure outside the cask, and could lead to additional loads on the sealing of the lids. But regardless of the number of failed rods the cask is designed to restrain the fission products from the environment.

The second scenario results in a heavier degradation of the fuel with broken fuel rods and fuel dispersal. In this scenario the original geometry of the fuel is no longer assured. All assessment results for criticality and thermal properties, which are based on the original geometry, are violated. The second scenario must be taken into account under accidental conditions and during transportation of the casks or reloading the fuel, if the cladding of the fuel rods is brittle after a long storage period. The second scenario envelopes the first scenario with only leaking fuel.

The explanation given above gives an overview of the importance of fuel cladding integrity during the interim dry storage in casks. In the German licensing and surveillance processes a systematic cladding failure during the storage should be excluded. Today we focus on a time scale of 40 years. The experience available on dry storage was analyzed and is the basis of the assessment of the fuel behaviour for this storage time span. The analyses of a CASTOR V 21 cask and the examination of the fuel after a storage time of 15 years confirm this concept [6], [7].

Within the knowledge of the material properties of irradiated fuel and especially of the irradiated cladding a forecast of the fuel rod performance under the boundary conditions of storage in casks was done. Thereby the following effects where identified to be important for the fuel performance during the dry storage up to 40 years:

(1) The cladding hoop stress should not exceed 120 MPa. Within this criterion stress corrosion cracking and hydride reorientation as failure mechanisms will be excluded;
(2) The cladding hoop strain should not exceed 1%. Within this criterion an increasing of the hoop stress at higher strain levels and the failure caused by strain should be excluded;
(3) The cladding temperature should not exceed 370°C. Within this criterion the annealing of irradiation hardening and the dissolution of the hydrogen precipitates in the cladding will be excluded;
(4) Cladding corrosion during the storage time should be limited.

A more detailed description of these effects was presented during the OECD International Workshop on Safety of Long Term Interim Storage [5]. Within this workshop some aspects of the extension of dry storage in casks and the ESCP gap list [8] were discussed.

For a longer time period than 40 years one can easily calculate the cladding properties using the same models and assumptions as made for the time span of 40 years and will find that all requirements are met for a time span up to 100 years and even longer. It has to be evaluated, whether the requirements for 40 years are still valid and applicable for a longer time span. For the longer time span the effects not driven by the temperature as hoop stress due to the inner fuel rod pressure and hoop strain due to creep of the cladding may become more important.

Other effects of degradation may take over the leading part in the assessment of cladding integrity. Since the high burnup fuel and mixed oxide fuel in the IAEA Standard on Storage of Spent Nuclear Fuel [4] are identified as the fuel with the highest loads and with the highest potential for systematic failure we limit this paper only to this kind of fuel. Other fuel with moderate burnup will be covered if it can be shown that high burnup fuel and mixed oxide fuel will withstand the extended storage periods.

In the following we want to address some effects on fuel and cladding degradation mechanisms, which should from our point of view, be part of the future assessment of fuel performance for an extended storage time period. In this regard, it is important to consider high burnup fuel and mixed oxide fuel because the experiments [6, 7], carried out on spent fuel are only valid for a burnup up to 45 MWd/kgHM [7].

2.1. Reorientation of hydrogen precipitates

Within the cladding hydrogen is picked up from the waterside corrosion during the in pile life of the fuel. After a certain concentration is reached the hydrogen precipitates following the thermal gradient in the cladding. These precipitates are orientated circumferential and have only minor effects on the load of the hoop stress. During the storage these hydrogen precipitates can re-orientate in the radial direction by dissolving and precipitating again after a heat up and cool down phase. This should be avoided by limiting the maximum cladding temperature to 370°C. But the reorientation of the hydrates is not fixed to a certain temperature. Under the conditions within the cladding e.g. temperature and hoop stress the reorientation can be going on slowly and will lead to delayed hydrogen cracking after a long period of time. This can be classified as a systematic failure mechanism for cladding with a certain amount of hydrogen and should be assessed for an extended storage time. It is known that the cladding of high burnup fuel will have the highest hydrogen content.

2.2. Fission gas release from high burnup structure

Another mechanism that should be addressed is the long term fission gas release from the fuel pellets. During the operation of the fuel the ratio of released fission gas to the produced fission gas is up to a range of 20% depending on the fuel type and the power history. More than 80% of the produced fission gas is trapped in pores, grain boundaries and in intergranular bubbles. A part of this gas can be released during the storage time. Especially the rim zone of high burnup fuel and the zones of the plutonium grains in mixed oxide fuel, which shows a high closed porosity with high cavity pressure, can be addressed as a source for a fission gas release during the storage. This additional gas release will increase the inner fuel rod pressure with its consequences to the hoop stress, hoop strain and hydrogen precipitates reorientation. Another effect that is a candidate for increasing the inner fuel rod pressure is the alpha decay that leads to a higher helium release from the fuel. These effects should be addressed for longer storage periods.

2.3. Cladding degradation by irradiation damages and fuel side chemical interactions

As a third mechanism we want to address special degradation mechanisms of the cladding material like corrosion and embrittlement of the cladding by oxygen uptake, stress corrosion cracking and failures caused by changes in the crystal microstructures by mobile dislocations and line defects. Cladding corrosion at the outer side of the cladding is limited by the low oxygen and water content in the casks atmosphere and is already considered in the assessments made today. Corrosion on the inner side of the cladding is not in focus so far, since it is assumed that no oxygen is available and the ceramic fuel is chemically stable. But it

is not known today whether this is also valid for the high burnup structure (HBS) at the outer rim of the pellets of high burnup fuel. Since the material micro structure has changed in the HBS after a certain burnup and this region of the pellet has a lot of pores with fission products there may be chemical interaction with the oxide fuel that may lead to an additional cladding corrosion and to an embrittlement by oxygen uptake as well as a cladding wall thinning. Stress corrosion cracking initiated by iodine and other halogens is considered by the limitation of the hoop stress, but long term consequences which can lead to cladding failures even at lower stress states are not considered so far. Changes in the cladding microstructures by annealing irradiation defects and mobile dislocations and line defects are mechanism that will take effect after very long periods of time. This should be analyzed with respect to the foreseen storage time.

The assessments of these effects will cause some experimental examinations of spent fuel after a longer storage time. These examinations should be used for the development of models to be used for the forecast of the fuel performance during longer storage times. To avoid a pile up of conservative assumptions in this kind of performance calculations, best estimate models should be established. This allows the evaluators to use the models for several aspects. Errors of the experiments and other uncertainties can be considered with probabilistic methods. These methods can also be used to identify the main driving mechanism. In these calculations the history of the fuel should be considered. Therefore we suggest using the available fuel performance codes which have in most cases the possibility to perform probabilistic calculations. These codes have to be extended with new models addressing the effects mentioned within this paper.

4. SUMMARY

Starting from the legal background in Germany for interim storage of spent fuel with respect to cladding integrity the aspects and the technical background for a dry storage time period of 40 years is presented. Some, from the authors' point of view, important technical aspects to be addressed for the extension of interim storage have been discussed with respect to high burnup fuel and mixed oxide fuel.

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MULTIPLE RECYCLE OF REMIX FUEL BASED ON REPROCESSED URANIUM AND PLUTONIUM MIXTURE IN THERMAL REACTORS

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Abstract

The report covers REMIX fuel consumption in VVER-1000. REMIX fuel is fabricated from unseparated mixture of uranium and plutonium obtained during NPP spent fuel reprocessing with further makeup by enriched natural uranium. It makes possible to recycle several times the total amount of uranium and plutonium obtained from spent fuel with 100% loading of the VVER-1000 core. The stored SNF could be also involved in REMIX fuel cycle by enrichment of regenerated uranium.

1. INTRODUCTION

The problem of spent fuel accumulation could be solved by utilization of regenerated uranium and plutonium in thermal reactors. The most popular and thoroughly elaborated are the variants of plutonium mixing with depleted uranium as MOX fuel and fabrication of fuel from regenerated uranium after its re-enrichment. These methods have several drawbacks which restricted their wide spreading. The alternative method has been developed based on using regenerated uranium and plutonium, mixed in several methods for fuel fabrication for the same thermal reactors which generate the SNF.

2. USAGE OF REMIX FUEL

2.1. Recycling of unseparated uranium and plutonium

The concept of REMIX (REgenerated MIXture of U, Pu oxides) fuel consumption in VVER-1000 (Russian version of PWR) has been developed in Russia. REMIX fuel is fabricated from unseparated mixture of uranium and plutonium obtained by SNF reprocessing with further makeup by enriched natural uranium for providing the required content of the fissile nuclides in the recycled fuel [1]. The calculations have shown that the composition of REMIX fuel (1,0- 1.5 % Pu + 2.5% ^{235}U) makes possible multiple recycling of the total amount of uranium and plutonium recovered from spent nuclear fuel at 100% loading of VVER-1000 core [2]. The sketch of regenerated uranium and plutonium multiple recycling is demonstrated in Fig. 1, radioactive waste being removed from spent fuel to disposal and the mixture of regenerated uranium and plutonium is the significant component of the refabricated fuel.

The change in nuclide composition of REMIX fuel for VVER-1000 fabricated from the regenerated uranium and plutonium after makeup by enriched natural uranium as a function of recycle factor is given in Table 1.

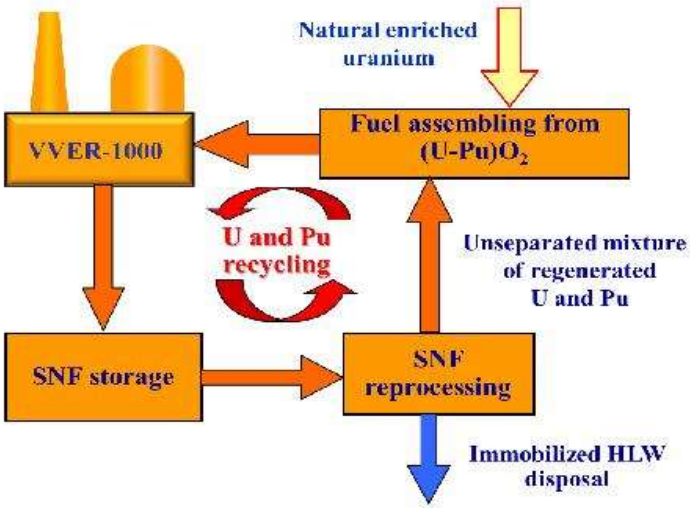

FIG. 1. The sketch of regenerated uranium and plutonium multiple recycling in VVER-100 as a REMIX fuel.

TABLE 1: U AND PU CONTENT IN REGENERATED REMIX FUEL (KG/TU) OF VVER-1000 (U AND PU FROM SNF – 80 %, ENRICHED URANIUM – 20 %, BURN-UP 50 GWD/T)

Nuclides	REMIX fuel composition before recycling		
	1st	3rd	5th
U-232	2.4 × 10-6	5.65 × 10-6	7.3 × 10-6
U-235	41.71	43.12	44.65
U-236	4.71	9.74	12.19
Pu-238	0.24	0.76	0.98
Pu-239	5.38	7.25	7.23
Pu-240	2.54	3.65	3.71
Pu-241	1.13	1.78	1.86
Pu-242	0.71	1.90	2.67
ΣPu	10.0	15.34	16.45
Pu-239 + Pu-241 + U-235	48.2	52.15	53.74

It shows that recycling results in increase of U and Pu even isotope content, which are a kind of neutron poisons, however, this increase is relatively small and could be easily compensated by additional makeup of enriched natural uranium.

The testing of REMIX fuel usage could be performed at the experimental-demonstration centre (EDC) being under construction now at MCC, Zheleznogorsk [3]. The SNF reprocessing flowsheet designed for EDC could be used for these tests with minimum flowsheet changes presented in Fig. 2.

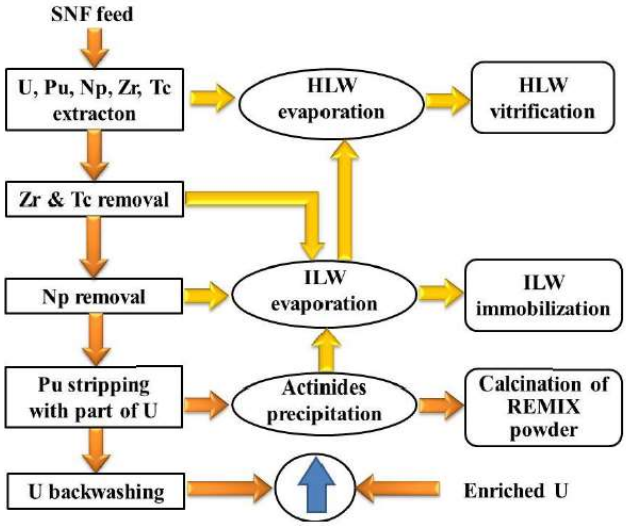

FIG. 2. The advanced flowsheet for fabrication of REMIX fuel for VVER at EDC (MCC).

The unseparated mixture of uranium and plutonium can be obtained as a regular solid solution (Fig. 3) of PuO_2 in UO_2 by precipitation or by direct thermal or thermochemical denitration of the evaporated joint uranium and plutonium backwash.

Pellets fabrication and assembling of fuel elements could be disposed in 9 research shielded cells specified in the EDC project.

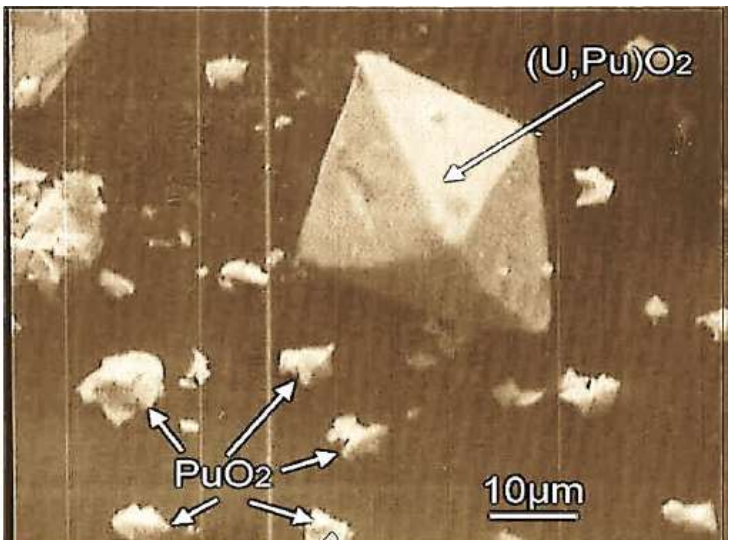

FIG. 3. A picture of (U, Pu)O_2 species, obtained by direct denitration in aqueous solution scanned by back reflected electrons.

2.2. The use of the stored Pu and NPP SNF

Regenerated mixture of uranium and plutonium produced by SNF reprocessing could be made up by the stored plutonium, reducing the make up by enriched natural uranium. Consumption

of the natural uranium could be reduced from 45% for unseparated mixture produced by SNF reprocessing, which contained 1% of plutonium fissile isotopes, to 20% when plutonium content in REMIX fuel is increased up to 5%. It is still possible to provide 100% loading of the core, because the neutron physics characteristics of the stationary loading of the core change insignificantly with increase of plutonium concentration in REMIX fuel within this range [2]. The REMIX concept could be enlarged to involve uranium and plutonium from the stored SNF, reducing their stock. The supposed flowsheet for involving the stored SNF is presented in Fig. 4.

It is based on enrichment of required part of regenerated uranium produced by reprocessing of the stored SNF and its mixing with another part of uranium and plutonium which is separated during the reprocessing [4]. Plutonium content in this fuel would be greater than 2% (up to 5%). It has been calculated that in the described above flowsheet of REMIX fuel usage for 1 t of thermal reactor fuel fissile nuclides of uranium and plutonium from 2-3 t of the stored SNF are used. As a result, one half of the NPP capacity could be loaded with the REMIX B fuel at the steady state of nuclear fuel cycle.

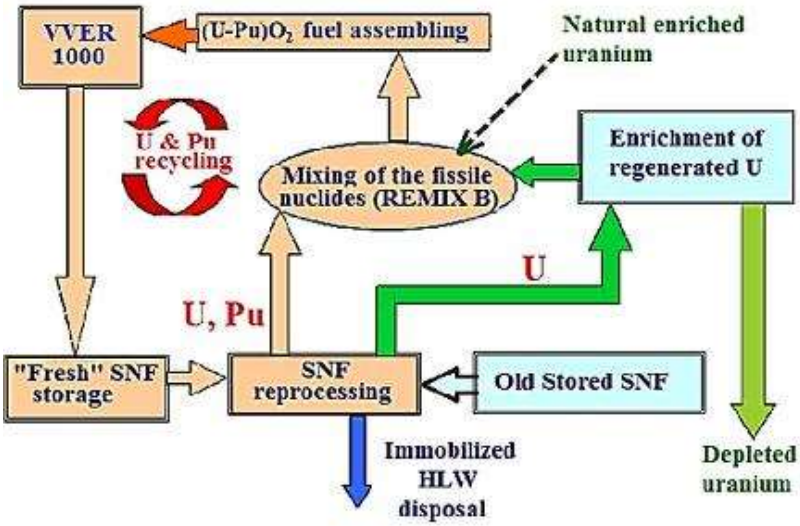

FIG. 4. The flowsheet for the stored SNF involving into the closed fuel cycle of thermal reactors with the use of REMIX fuel.

3. ECONOMIC OUTLOOKS

Estimation of technical and economic assessment on the basis of MIT method has demonstrated that the fuel cycle with the use of REMIX fuel could be less expensive than the open fuel cycle with direct SNF disposal if the cost REMIX fuel fabrication would be no more than five-fold expensive as compared to typical natural uranium fuel.

The proposed above approach to the closed fuel cycle for thermal reactors has potential to improve economic indexes by reducing the number of process steps during SNF reprocessing and fuel refabricating, namely the joint backwashing of plutonium and uranium mixture which leads to reducing the several stages of fuel powder and pellets fabrication as compared to MOX fuel fabrication.

Production of 1 t of MOX fuel containing 7-10% of total plutonium mixed with depleted uranium requires reprocessing of 5-7 t of SNF containing 1.5% of plutonium. If the total cost

of fuel cycle is included in the cost of MOX fuel considering increase in 5-7 times of the cost of reprocessing, the final cost of MOX fuel would be rather high — around $6000/kgU.

Production of 1 t of REMIX fuel requires reprocessing of 0.8 t of SNF, so the cost of REMIX fuel would include the cost of reprocessing with the factor 0.8 instead of 5-7 for MOX fuel. The estimation of REMIX fuel fabrication cost demonstrated that it would be less than the cost of MOX fuel fabrication because of several times higher amount of REMIX fuel production [5] and depending on the cost of REMIX fuel fabrication and the cost of natural enriched uranium the total cost of REMIX fuel would be 2-4 times less than that of MOX fuel.

Thus, REMIX fuel involving in the nuclear fuel cycle of thermal reactors makes possible to reduce the amount of SNF because of high reuse of regenerated fissile materials at 100% reactor core loading with the single type of regenerated fuel and decrease the amount of consumed natural uranium.

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ON THE IMPACT OF THE FUEL ASSEMBLY DESIGN EVOLUTION IN THE SPENT FUEL MANAGEMENT

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Abstract

For decades, fuel assembly design evolution has been partly focused on performance and reliability improvement to increase the end of life burnup and extend fuel assemblies operation life. However, more attention was needed on the impact of the designs and their consequences on the subsequent spent nuclear fuel (SNF) management. SNF management requires handling, drying, transportation, storage, etc; at normal, off-normal and accident conditions. This work collects three main fuel assembly design fields: fuel rod design, fuel cladding materials and fuel assembly structural materials; and describes an example for each field of their impact on spent fuel management. First, as fuel rod design related example, high as-built rod internal pressure and short plenum lengths in the early fuel rod designs generate hoop stresses close to 90 MPa, the stress limit imposed by NRC ISG-11 Rev. 3, even after 30 years of cooling. Second, as structural materials related example, stress corrosion cracking of top nozzle sleeves made of sensitized 304 stainless steel in specific PWR designs. Finally, as cladding materials related example, local embrittlement of Zircaloy-4 cladding due to zirconium oxide spalling of fuel rods irradiated to high burnups, which affects to cladding integrity and needs analysis for SNF transportation and storage. Giving the impact of these particular issues in the SNF management, deep knowledge of the fuel assembly designs and the potential effects during spent fuel management are fundamental in order to accomplish an optimum dry storage and transportation.

1. INTRODUCTION

ENUSA, the Spanish nuclear fuel manufacturer since the 1980s, is the main specialist in Spain of LWR nuclear fuel design and performance. During this time, ENUSA has acquired deep knowledge and experience on the fuel assemblies irradiated by its customers, and currently stored in the spent fuel pools (SFP). Eventually, ENUSA has also specialized in SNF management, offering integrated engineering and on-site services to the Spanish plants. In fact, ENUSA currently has the lead in classification of SNF for dry storage and transportation projects developed in Spain.

Spanish SNF dry storage and transport is regulated by the national safety council (CSN), and Spanish regulations, however for fuel classification the US NRC legislation is followed. Particularly, damaged SNF definition is contained in the Interim Staff Guidance, ISG-1 Rev 2 [1]1]: any fuel rod or fuel assembly that cannot fulfill its fuel-specific or system-related functions.

Once the fuel assemblies reach their end of life, they are handled to the spent fuel pools for wet storage and cooling. The assemblies remain in the SFPs during several years until they may be introduced into dry storage containers after enough cooling. Therefore, once operation ends, SNF will spend several decades in other environments different to the primary coolant and at different temperatures. Eventually, the fuel assembly will be handled for different reasons (inspections, dry storage container loading, etc) and will be transported from the independent spent fuel storage installations (ISFSI) at the power plants towards a centralized temporary storage facility, where they will be dry stored during another 100 years. Therefore, SNF management requires decades of wet storage, handling, drying, transportation, dry storage, etc; and at normal, off-normal and accident conditions.

Traditionally, fuel assembly design evolution has been partly motivated by performance and reliability improvement to increase the end of life burnups and, therefore, extension of the fuel assembly operation life. Almost 100% of the fuel assemblies have operated properly under the design limits without leakers or degradation problems. However, the second part of

the cycle was not always considered in the past, and some specific design features should be analyzed when SNF is going to be dry stored and transported.

This work collects three main PWR fuel assembly design fields: fuel rod design, fuel assembly structural materials and fuel cladding materials; and describes an example for each field of their impact on SNF management. In the following sections, it is given a brief description of each example, its impact on dry storage management and actions taken by ENUSA or the industry in order to success in correct and efficient dry storage management.

2. IMPACT OF FUEL ROD DESIGN

After fuel assembly operation, fuel rods internal pressure is due to the presence of helium and fission gasses released during operation in the reactor. Fuel rod internal pressure at the time of drying and storage also increases due to temperature raising and determines cladding hoop stresses at the time of dry storage, which are limited by ISG-11 Rev. 3 to a maximum value of 90 MPa [2][1]. Hoop stresses above 90 MPa could promote radial hydrides precipitation, affecting to the cladding mechanical response during dry storage and transportation.

The first PWR fuel rod designs were designed to avoid cladding collapse during operation into the reactor. For that reason, fuel rods were pressurized to very high internal pressure during fuel rod manufacturing. Additionally, fuel rod designs included low plenum length to compensate the effects of irradiation growth; non-controlled porosity during pellet manufacturing producing non-optimized pellet swelling; and constant pitch plenum spring design which occupied more volume than current design (variable pitch designs). Therefore, the overall effect in some old designs is that cladding hoop stresses due to internal pressure could be high enough to difficult an easy justification of the maximum hoop stress value of 90MPa according to ISG-11 Rev. 3.

To meet ISG-11 Rev. 3 quick conservative hoop stress calculation has shown effective for optimized later fuel rod designs. However, special calculations could be necessary for old fuel rod designs which exceed 90 MPa applying conservative hypotheses. Therefore, in order to justify the fulfillment of the ISG-11 Rev. 3 requirement (σ_Θ <90 MPa), ENUSA has developed a methodology for cladding hoop stresses calculation, using a fuel rod thermomechanical code and considering the axial temperature profile of the fuel assemblies inside the container [3].

Optimized later evolution of the fuel rod designs included lower internal pressures, higher plenum lengths, controlled pellet porosity and plenum spring optimized design. These changes increased the volume to accommodate the internal gases, however, end of life burnup and fission gas release also increased, contributing to raising of the cladding hoop stresses. Therefore, if needed, the same methodology could be applicable also for high burnup fuel assemblies.

3. IMPACT OF FUEL ASSEMBLY STRUCTURAL MATERIALS

In 2001, in North Anna NPP, while one assembly was being returned to its spent fuel rack after an inspection previous to dry storage, the top nozzle separated from the assembly and the bundle dropped about 3.6 m into its storage cell. A historical review revealed that similar events had occurred in the past, for example in Prairie Island in 1981, and several events of top nozzle detachment without assembly dropping in other plants [4].

In all cases, the fuel bundle separated from the top nozzle due to cracking of the sleeves that connected the top nozzle to the guide tubes. Fig. 1 shows a schematics of the top nozzle — skeleton joint. The sleeves are joined to the skeleton through bulges and welded to the top nozzle. Root cause analysis performed in the past revealed that the primary cause of the top nozzle detachments was intergranular stress corrosion cracking (IGSCC) of the sleeves in the bulged regions [4].

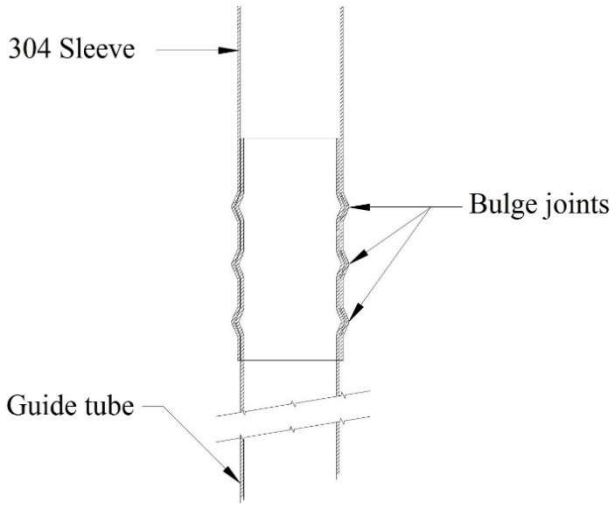

FIG. 1. Schematics showing the bulge joints between the guide tube and the top nozzle sleeve.

Stress corrosion cracking (SCC) phenomena develop when three factors happen at the same time: susceptible material, aggressive environment and stress. The sleeves were made of stainless steel type 304. During fuel assembling process, before attachment to the skeleton and top nozzle, the sleeves were joined to the top inconel grid by brazing. After brazing, top grid were heat treated to age harden the inconel straps at temperatures and times which produced sensitization of 304 sleeves to intergranular corrosion. The aggressive environment were the spent fuel pools with presence of sulfur acid species. The stresses were residual stresses in the bulged areas generated during the bulging (cold deformation) process [4].

Once detected, the issue was solved by a series of corrective actions oriented to prevent the SCC process: replacing 304 by 304L stainless steel as sleeves material, redesign of the top nozzle joint and additional control of the spent fuel pool chemistry including sulfate species. The implementation of these actions avoided SCC in assemblies later stored in the spent fuel pools. However, about 18 000 fuel assemblies were already affected worldwide [5].

SCC degradation on sleeves impacts to handling of the assemblies (in the SFPs and after drying) and, therefore, to fuel retrievability safety function. An assembly candidate to dry storage must meet ISG-2 Rev. 1 requirements regarding fuel retrievability [6]. Currently, fuel assemblies affected by SCC must be visually inspected to detect SCC indications, subjected to handling limitations, handled using special tools or repaired in order to be handled by normal means [5].

4. IMPACT OF FUEL CLADDING MATERIALS

Increasing of the fuel burnup implies higher corrosion on cladding material. Therefore, a lot of investigations have been performed to improve the corrosion performance of the clad material and to decrease the oxide growth rate during irradiation. Clad material for PWR fuel

assemblies have evolved from Zircaloy-4 to Improved Zircaloy-4 (low tin) and ZIRLO®. However, for transition cycles among one cladding material to another, fuel assemblies using old clad material sometimes have been irradiated at demanding conditions, more appropriated for the new materials. Due to this situation, there is a population of Zircaloy-4 cladding fuel assemblies with fuel burnup about 45-50 MWd/kgU and significant oxide thickness.

During operation in the reactor, Zircaloy-4 cladding is oxidized forming a waterside oxide (ZrO_2) layer. If oxide thickness overcome high enough values, detachment of the oxide layer may occur, contacting the base metal with the primary coolant. Spalling of ZrO_2 oxide has been observed in Zircaloy-4 cladding. The appearance of a high burnup (about 64 GWd/tU) Zircaloy-4 cladding is shown in Fig. 1[7].

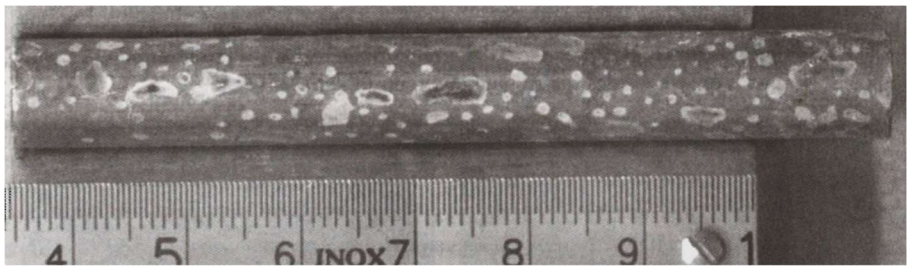

FIG. 2. External appearance of a Zircaloy-4 spalled cladding [7].

Additionally, during cladding oxidation, the reaction between water and zirconium to form ZrO_2 produces hydrogen that is partially absorbed by the metallic clad. The initial charging of hydrogen is in solid solution until the solubility limit is reached. Hydrogen solubility limit depends on the alloy composition and temperature. When hydrides precipitate, the hydride distribution across the clad thickness depends on temperature distribution which depends on the heat flux.

A local hydride blister can be formed when the oxide is spalled off due to a stepped temperature gradient in the clad thickness. Fig. 3 shows a hydride blister precipitated below a spalled area in a Zircaloy-4 cladding. Since oxide layer is a thermal insulation barrier, a cold spot is formed under the spalled oxide, where the hydrogen migrates and precipitates massively as zirconium hydride. At high levels of hydrogen, the hydride precipitates can form a hydride lens or blister. Since hydrides are relatively brittle compared to the clad metal, this metallurgical process of formation of massive hydride blisters compromise the structural integrity of the Zircaloy-4 cladding.

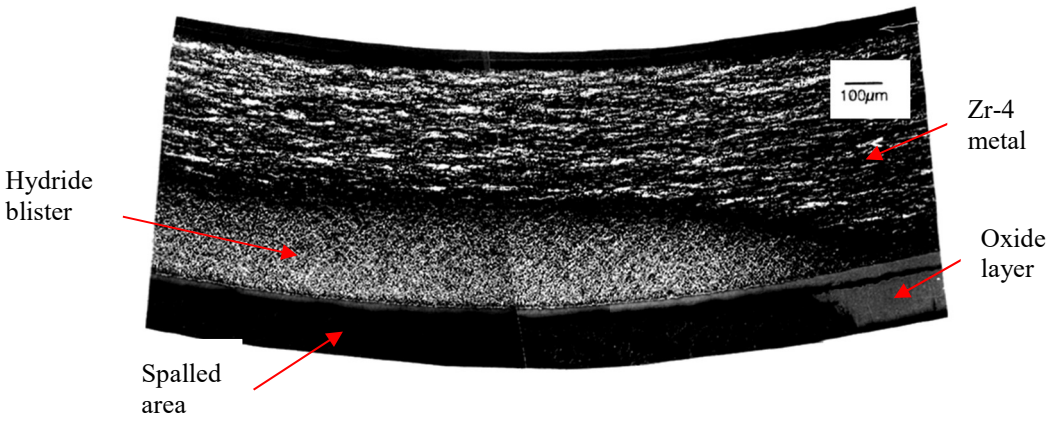

FIG. 310. Zirconium hydride massive precipitation below a spalled area in an irradiated Zircaloy-4 cladding [8].

Enough cladding integrity is necessary for dry storage operations. Hydride blister precipitation locally reduces the resistant cladding section concentrating the stresses in the affected areas during drying, transportation and storage at high temperatures in normal and accident conditions. Compared to non spalled area, local stresses concentrations in spalled areas due to hydride blister precipitation could degrade the mechanical response in case of dynamic loads during transportation in normal and accident conditions or increase creep rates during dry storage.

ENUSA, in collaboration with other partners inside the Spanish nuclear industry, has analyzed by calculation the spalled and hydride blister precipitation affected Zircaloy-4 cladding integrity during dry storage and transportation assuming conservative hypotheses. Besides, ENUSA is involved in R&D programs focused in obtaining fundamental knowledge and data about the fracture modes of fuel cladding affected by hydride blister precipitation during dry storage and transportation [9].

No spalling has been observed in ZIRLO® cladding material, therefore, due to the evolution of the cladding material, oxide spalling is not present in the current fuel.

5. SUMMARY

This paper analyze the design evolution on three main PWR fuel assembly design fields: fuel rod design, fuel assembly structural materials and fuel cladding materials; and describes an example for each field of their impact on spent fuel management.

- Fuel rod design: early cladding designs high hoop stresses, close to 90 MPa limit;
- Fuel assembly structural materials: stress corrosion cracking of the top nozzle sleeves affecting fuel assembly retrievability;
- Fuel cladding materials: local embrittlement due to massive hydride precipitation in spalled areas; affecting mechanical Zircaloy-4 cladding integrity during drying, transport and storage.

Each paragraph includes a brief description of each example, their impact on dry storage management and actions taken by ENUSA and/or its partners in order to success in efficient dry storage management.

The best lesson learned from this work is that giving the impact of these particular issues in the SNF management, deep knowledge of the fuel assembly designs and the potential effects of design changes are fundamental in order to accomplish an optimum classification for dry storage and transportation. Moreover, since fuel assemblies design keeps continuous evolution to improve their performance and extend their operation life, the design process should include in the design process an analysis of the impact of the design changes in the future back end.

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COMPARATIVE ANALYSIS OF ISOTOPE COMPOSITION OF VVER 1000 WESTINGHOUSE AND TVEL SPENT FUEL

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Abstract

This work has analyzed the impact on VVER-1000 spent fuel isotope composition caused by the different operational conditions, such as the presence or absence of absorber-rods, oscillating the concentration of boric acid, dissolved in the moderator (water) during the campaign, fuel and/or moderator temperature, as well as changes in water amount at the periphery of an assembly due to its location in the central or periphery part of the core and/or due to changes in inter-assembly gaps. Also, impact caused by technological allowances applied while manufacturing fuel assembly was analyzed by weight of fuel and by its enrichment. Calculations were made for reactor cells of fuel assemblies for VVER-1000. They were composed of the typical fuel assemblies of Russian TVEL suppliers and the new fuel assemblies of Westinghouse company.

This has been removed from the proceedings as it has since been reproduced elsewhere. A copy of the paper can be obtained from Ref. [1].

REFERENCE

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REPROCESSING OF SPENT MIXED NITRIDE U-PU FUEL FROM BREST-OD-300 REACTOR

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Abstract

BREST-OD-300 is pilot demonstration lead-cooled fast reactor fuelled with mixed nitride uranium-plutonium fuel (MNIT). Nuclear fuel cycle with BREST reactors concept implies that the fuel has an burn-up about 10 % h.a. 'Equilibrium' isotopic composition of fissile materials (FM) fuel is used in the nuclear fuel cycle and the duration of the external fuel cycle does not exceed one year. Combined (pyro+hydro) technology (PH-process) is proposed for reprocessing of MNIT BREST SNF. PH-process consists of a combination of pyrochemical, including pyroelectrochemical, head operations and hydrometallurgical operations for refining of end product (U-Pu-Np-Am) and waste treatment. Development of PH-process has been run since 2011 by RIAR, Bochvar Institute, and Khlopin Radium Institute. In 2014 manufacturing of experimental setups to test engineering solutions for equipment for pyroelectrochemical operations was started and installation for study of innovative systems for hydrometallurgical operations was completed. In 2015 engineering of reprocessing MNIT SNF and waste treatment unit was started. Reprocessing unit will be located at the same site with the pilot demonstration power complex with BREST-OD-300 reactor on the territory of SCC (Seversk).

1. INTRODUCTION

BREST-OD-300 is pilot demonstration lead-cooled fast reactor (FR) fuelled with mixed nitride uranium-plutonium fuel (MNIT). Nuclear fuel cycle with BREST reactors concept implies that the fuel has an burn-up about 10% h.a.. "Equilibrium" isotopic composition of fissile materials (FM) fuel is used in the nuclear fuel cycle and the duration of the external fuel cycle does not exceed one year. It means that within one year spent nuclear fuel (SNF) should be reprocessed and fresh MNIT-fuel should be fabricated from the products of the reprocessing. At the present time a carbothermic synthesis technology is chosen as a basic process for the fabrication of mixed uranium-plutonium powders and pellet technology – for fuel fabrication. Thus manufacture on reprocessing of MNIT SNF from BREST reactor should provide reprocessing of fuel with cooling time not exceeding one year, FM content of 10-15%, burnup about 10% h.a. and obtain a mixture of actinides with coefficients of purification about 1 000 000. In our opinion PH-process (combined (pyro+hydro) technology) of reprocessing SNF can meet all these requirements. PH-process consists of a combination of pyrochemical, including pyroelectrochemical, head operations and hydrometallurgical operations for refining of end product (U-Pu-Np-Am) and waste treatment. Development of PH-process has been run since 2011 by RIAR, Bochvar Institute, and Khlopin Radium Institute.

2. COMBINED (PYRO + HYDRO) REPROCESSING OF MNIT SNF (PH-PROCESS)

The proposed combined pyro + hydro flowsheet of SNF FR reprocessing, which had got the name PH-process [1, 2] is based on the use of non-aqueous head end operations, assigned achieve the purification of fissile materials from fission products with a purification factor 100, followed by a row of aqueous processes to get the starting material for new fuel fabrication (Fig. 1).

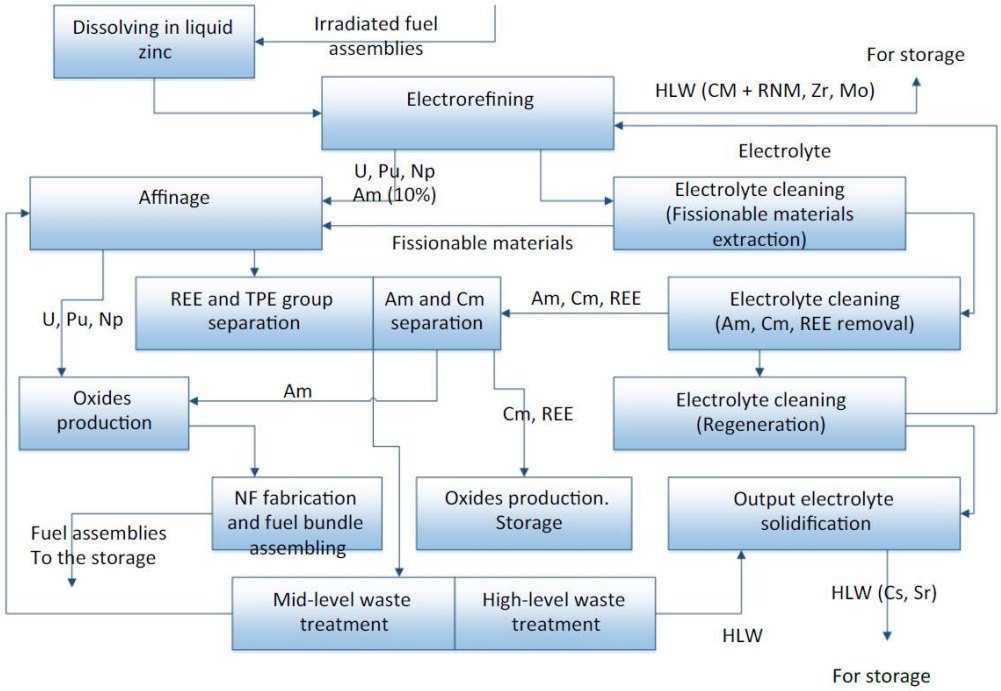

FIG. 1. PH-process principal flowsheet.

The pyrochemical head end of PH-process comprises the following operations: fuel decladding by liquid zinc; preliminary MOX reduction to metal state with lithium vapors in Li_2O – LiCl melt; electrorefining of any dense (including MNIT, metal or metallized MOX) fuel with liquid cadmium electrodes; local gas purification Cs, Sb, systTem, Se, for pyrochemical head end for trapping radioactive noble gases and ^3H); vacuum distillation of cadmium metal and salt electrolyte; oxide precipitation of U, Pu, Am, Cm and rare earth element (REE) oxides from salt electrolyte; alkaline and alkaline earth selective crystallization form electrolyte followed by the vitrification of the product and return of the purified salt to the process.

The aqueous part of PH-reprocessing of SNF FR includes the following operations: dissolution of all products containing uranium, plutonium, neptunium, americium and curium in nitric acid; solutions clarification; separation of fission products from U-Pu-Np using extraction–crystallization scheme; extraction recovery of minor actinides; Am-Cm separation; evaporation of high and medium level liquid wastes; preparation of U, Pu, Np and Am oxides; preparation of curium oxides for long-term storage; medium level waste management; high level waste solidification.

3. DEVELOPMENT OF SOME OPERATION OF PH-PROCESS

First stage of hydrometallurgy part of PH-process is dissolution. A metal ingot consists of U – 84.5%; Pu – 15%; Np - 0.5% was prepared for dissolution test. This ingot with mass 66.3 g was separated on three parts (Fig. 2). Dissolution was tested under following conditions:

- 8 M HNO_3, 4 hours and concentration of actinides in final solution – 450 g/l, Tmax = 132°C;
- 10 M HNO_3, 6 hours and concentration of actinides in final solution – 400 g/l; Tmax = 105°C.

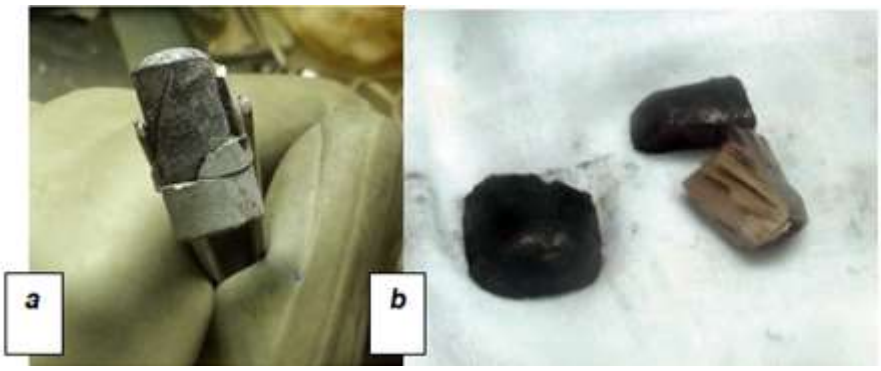

FIG. 2 Initial (a) and seperated (b) U-Pu-Np ingot.

Weights of insoluble residue after dissolution under first and second condition set were 16.5 and 1.5% from initial metal weight, respectively. The residues contain mixed oxides (PuO:89, UO:11)O_2, (PuO:5,UO:5)O_2 and metal.

The extraction and crystallization affinage is a main operation of aqueous part of pH-process. This technology is developed jointly by Bochvar Institute and Siberian Chemical combine. The crystallization experiments were performed the solutions obtained by purification of U and Pu using tributylphosphate (TBP) solutions extraction (first serial of tests) or by mixture of $UO_2(NO_3)_2$ with Pu purified by sorption with anion exchange resin (RH-1) (second serial of tests). Before the starts of crystallization study Pu in all initial solutions was oxidized to Pu (VI). Composition of initial solutions for the first series of tests is given in Table 1.

TABLE 1. COMPOSITION OF INITIAL SOLUTIONS

Test	[Pu], g/l	Contain of Pu(VI), %	[U], g/l	[HNO3], M	T, °C
1	143	69.8	363.4	1.7	0
2	130	44.0	330.3	1.55	- 17
3	95.3	63.1	242.3	6.5	- 17
4	130	69.8	330.4	1.64	- 17

All tests were performed using 5 ml glass tubes. The closed tubes were kept under needed temperature with periodical shaking for 1 hour. The mother solutions were sampled from any tubes after crystallization. The co-crystallization coefficients were calculated using Eq. (1). The results are given in Table 2. It should be noted that increase of Pu co-crystallization coefficient was observed for high content (44%) of Pu (IV). The explanation of this fact needs additional study.

$$K_{Me} = \frac{G_{MeKr}/G_{MeSum}}{G_{UKr}/G_{USum}} \quad (1)$$

Where

G_{MeKr} is amount of component in crystal;

G_{MeSum} is the total amount.

TABLE 2. PU CO-CRYSTALLIZATION COEFFICIENTS (FIRST SERIES OF TESTS)

Test #	Solution composition,		T, (°C)	Pu co-crystallization coefficient	Initial U/Pu ratio
	HNO_3 (M)	Pu+U			
1	1.7	2.13	0	0.65	2.87
2	1.55	1.93	- 17	1.03	2.74
3	6.5	1.42	- 17	0.68	2.67
4	1.63	1.93	- 17	0.53	2.81

Second series of tests mixture of Pu after sorption purification and U after extraction purification were evaporated. Evaporated residue was kept under boiling temperature for 2 hours. 77.4% of Pu were oxidized to Pu(VI). Compositions of initial solutions for second series of tests are given in Table 3. Experimental data are given in Table 4.

TABLE 3. COMPOSITION OF INITIAL SOLUTIONS (FIRST SERIES OF TESTS)

Test #	[U], g/l	[Pu],g/l	[HNO_3], M
5	356.8	202.6	5.9
6	356.8	202.6	5.9
7	317.1	180.1	7.7
8	277.5	157.6	9.5
9	396.4	225.2	7.5
10	365.9	207.8	7.5
11	339.8	193.0	7.5
12	339.8	193.0	7.5

The obtained data shows that there is an enrichment of crystals with uranium. Pu co–crystallization coefficients are < 1 for all experiments. It should be noted that increasing of [HNO_3] from 5.9 M to 9.5 M leads to decreasing of Pu co–crystallization coefficients from 0.96 to 0.71 at –20°C, and decreasing of temperature from −5 to −20°C leads to increasing of Pu co–crystallization coefficient.

TABLE 4. PU CO-CRYSTALLIZATION COEFFICIENTS (SECOND SERIAL OF TESTS)

Test #	Solution composition (M)		T, (oC)	Pu co-crystallization coefficient	Initial U/Pu ratio
5	5.9	2.35	- 5	0.86	1.62
6	5.9	2.35	- 20	0.96	1.10
7	7.7	2.09	- 20	0.74	1.47
8	9.5	1.83	- 20	0.71	1.71
9	7.5	2.61	- 5	0.76	1.57
10	7.5	2.41	- 5	0.67	1.82
11	7.5	2.24	- 5	0.73	1.58
12	7.5	2.24	- 20	0.80	1.76

The next important operation of PH-process is recovery and separation of Am and Cm. This operation is need for transmutation of MA. The extraction method of transplutonium (TPE) and REE intragroup separation based on well-known CMPO (carbamoylmethilenephosphine oxide) extractant and novel diluent formale-H_2 (octafluoropentanole, $[H(CF_2)4CH_2OCH_2OCH_2(CF_2)4H]$) was tested for PH-process [3]. The test flowsheet is given on Fig. 3. Dynamic tests of this flowsheet using simulated solution (Table 5) and real rafinate of PUREX process of VVER-1000 SNF with burnup 60 GWd/t(HM) show the high efficiency of this method for TPE and REE intragroup separation. Thus novel extraction mixture (0.2 M CMPO in mixture 30% vol. TBP – 70% vol. formale-H_2) effectively recovers Am and other actinides.

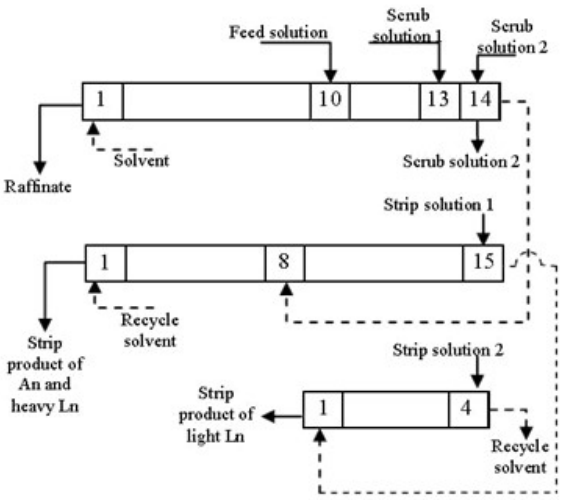

FIG. 3. Principal flowsheet of intragroup REE and TPE separation by extraction mixture − 0,2 M CMPO in mixture 30% vol. TBP −70% vol. formale-H_2 (from [3]).

TABLE 5. DISTRIBUTION OF COMPONENTS DURING INTRAGROUP REE AND TPE SEPARATION BY EXTRACTION MIXTURE – 0.2M CMPO IN MIXTURE 30% VOL. TBP – 70% VOL. FORMALE-H_2 FROM SIMULATED SOLUTION (FROM [3])

Element	Raffinate	Distribution of elements, % from initial	
		Am and heavy REE strip solution	Light REE strip solution
Ce	0.04	0.6	100
Eu	0.04	95	11
Fe	0.01	1	3
La	0.005	0.01	100
Nd	0.3	1	100
Pm	0.3	1	100
Sm	0.2	90	10
U	0.01	0.8	100
Am	0.01	98	4

Apart from the technology of actinides mixed oxides preparation was studied. Traditional technology actinides mixed oxides preparation is precipitation from ammonia solution and calcination of precipitate at ~800°C in Ar-H_2 atmosphere. It is a multistage technology with rather big volumes of mother solution. The actinides mixed oxides preparation from nitric acid solution in the presence of hydrazine hydrate described in Refs. [4, 5, 6] were proposed for PH-process. It was shown that thermal treatment at ~90°C solutions of U, Np and PuC) (in the presence of hydrazine hydrate leads to form of hydrated oxides of actinides, which performs to crystalline dioxides at 300°C. The thermalize of hydrated oxides of actinides U, PuNp under same conditions allows to obtain firstly (in solution) hydrated U, Np, Pu)$O_2 \cdot nH_2O$, which performs under calcination (air, 800°C) in crystal. This method allows to obtained the solid solutions of actinides with uncertain composition. It was shown that prepared by this technique U-Np-Pu oxide is X-ray amorphous compound [4, 5]. X-ray analysis identified or (U,Pu,Np)O_2 phase or homogenous mixture of three dioxides in calcinated under 800°C oxides mixture (Fig. 4).

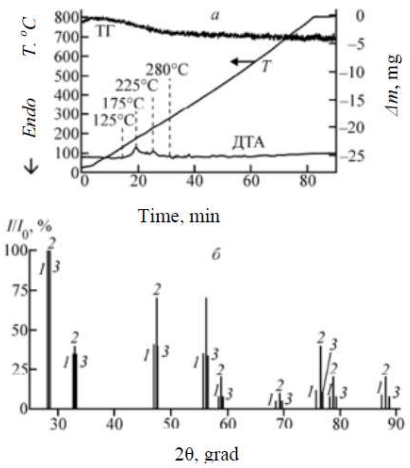

FIG. 4. Thermal analysus of hydrated oxides of U, Pu and Np (a) and X-ray analysis of this mixture after calcination (air, 800°C) (from [5]).

4. ELABORATION OF PILOT SCALE FACILITIES FOR THE PH REPROCESSING TEST

At the present time two Russian enterprises including for RIAR and “SOSNY” are engaged to the development of the inert hot cell with simultaneous mounting of the principal apparatus of pyro-branch for the pilot scale test of PH reprocessing (Fig. 5).

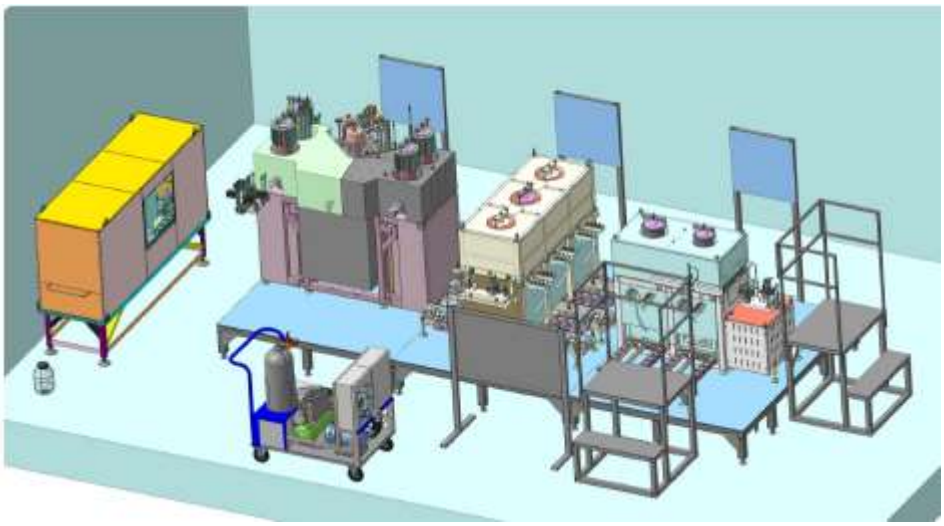

FIG 5. Placement of the PH reprocessing pyrochemical facility in the inert hot cell.

Bochvar Institute finished the mounting of a facility for pilot scale gas purification system, steps of dissolution, partitioning and actinide oxide preparation (Fig. 6).

FIG. 6. Dissolution, extraction, gas purification and actinide oxides preparation facilities.

At the same time, Bochvar Institute in co-operation with Siberian Chemical Combine is mounting scale-up facility of extraction-crystallizing to evaluate the technology of extraction and extraction-crystallizing refining (Fig. 7a). Joint R&D work of Bochvar Institute, Khlopin Radium Institute and SverdNIIChimMash resulted in the finishing of the equipment scale-up and in the start of the experiments testing the processes of nitric acid solutions evaporation and ammonium nitrate (and/or) organics thermal decomposition (Fig. 7b).

FIG.7. Extraction — crystallization (a) and evaporation — thermal decomposition facilities (b) (Siberian Chemical Combine).

The R&D with the mentioned facilities directed to check a number of operations involved to PH process, including gas purification system [7, 8, 9], dissolution, partitioning [3,10] and actinide oxides preparation [4, 5, 6] is planned to be carried out in 2015–2017.

5. MODELLING OF SNF REPROCESSING

A pilot software “Balance model” has been created earlier [11]. It allows to execute express updating of calculation model and to balance settlement all over the PH-process flowsheet of SNF reprocessing, fuel re-fabrication and waste management. A block library, containing the modules for the material balance calculation for separate processes, involved to the reprocessing, written in C#, was created as a part of developed software. Work on integration ‘Balance models’ in a software system on the basis of platform ATEK that perform a number of different tasks on modeling of chemical and physical processes and operations was started.

The work on integration of ‘Balance models’ into ‘WizArt’ software system simultaneously with carrying out of balance models improvement was completed. The inclusion of “Balance model” in the software system structure allowed implementing the visualization of calculation model plotting in design mode, to simplify the task of input of compositions of incoming products, of input of operation parameters, of output of calculation data using the means of user interface. In addition, the studies directed to the implementation of functional on dynamic balance modeling of the flowsheet taking into account productivity of equipment, of transport lines and storage containers capacity are carried out. This development will allow providing the calculations of the equipment functioning cyclograms, using the blocks of integrated balance model and calculation scheme variant.

6. CONCLUSION

Combine (pyro + hydro) technology for MNIT BREST SNF reprocessing (PH-process) is under development. PH-process would allow: reprocessing of FR SNF with high burn up and low cooling time; decrease the volume and time of SNF storage; to reprocess any type of SNF; to prepare the final uranium – neptunium – plutonium product running the requirements for fabrication of the final pelleted mixed MNIT fuel.

By now following basic technological operations have been tested in laboratory conditions on real products: steel cladding removal by dissolution in molten zinc; pyroelectrochemical recovery of uranium, plutonium, and neptunium; extraction refining of uranium, plutonium, and neptunium; separation of rare earth elements and transplutonium elements.

In 2014, manufacturing of experimental setups to test engineering solutions for equipment for pyroelectrochemical operations was started and installation for inspection innovative systems for hydrometallurgical operations was completed. In 2015, engineering of reprocessing MNIT SNF and waste treatment unit was started. Reprocessing unit will be located at the same site with the pilot demonstration power complex with BREST-OD-300 reactor on the territory of SCC (Seversk).

ACKNOWLEDGEMENTS

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RESEARCH AND DEVELOPMENT REQUIRED TO DELIVER AN INTERGRATED APPROACH

(SESSION 7)

Chairpersons

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THE IAEA CRP BEFAST AND SPAR – THIRTY-FOUR YEARS OF EXPERIENCE WITH THE STORAGE OF SPENT FUEL AND FACILITY COMPONENTS

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Abstract

The IAEA Coordinated Research Programmes (CRPs) on the behaviour of spent fuel and storage facility components during long term storage were initiated in 1981 under various titles. The first programme was designated BEFAST-I (BEhaviour of spent Fuel Assemblies in STorage), followed by BEFAST-II in 1986 and BEFAST-III in 1991. Follow-on CRPs were called SPAR-I to -III (Spent fuel Performance Assessment and Research), respectively. Overall, 20 countries and one international organization participated in one or more of the CRPs. During the CRPs, the participating countries contributed their R&D results on fundamental questions regarding spent fuel storage. The overall objectives of the CRPs were to develop a technical knowledge base on long term storage of commercial power reactor spent fuel through evaluation of operating experience and research by participating Member States, and to extrapolate predictions of spent fuel behaviour over long periods of time. A final report for each stage of the CRPs was prepared and published as an IAEA Technical Document (TECDOC). Towards the end of the BEFAST-III project, it became apparent that the R&D component of the project was decreasing steadily; more emphasis was being placed on the operation and implementation of storage technology. However, the storage technology (particularly dry storage) was undergoing a rapid evolution: new fuel and material design changes were coming on stream and target discharge burnup were steadily increasing; in addition, storage durations were predicted to be much longer than anticipated. Because of all these changes, a subsequent project was proposed to address the impact of these new parameters on long-term storage and to determine their consequences on subsequent operations, especially handling and transportation. The paper describes the research objectives and a selection of the findings related to spent fuel and storage system performance of the BEFAST and SPAR activities from 1981 until today.

1. INTRODUCTION

The peaceful use of nuclear energy was supported from the beginning by international cooperation. The front and back ends of the nuclear fuel cycle benefitted from bilateral and multilateral exchanges of experience. Even after the commercialization of nuclear energy, the transfer of knowledge on the peaceful use of nuclear energy has continued at many different levels. Information exchange through the International Atomic Energy Agency (IAEA) is one of the most efficient mechanisms, among others, which exists to the mutual interest and benefit of both developed and developing countries.

Spent fuel storage is an important step in the back end of the fuel cycle. From very early in the development of power reactors, fuel assemblies for light water reactors (LWR) were manufactured typically of ceramic UO_2 in the form of pellets enclosed in zirconium-based alloy tubing. Consequently, the generic questions related to the storage of spent fuel are

common to all Member States operating light- and also heavy-water reactors. Member States who operate different reactor systems were also invited to contribute to the projects, and in return, were able to derive benefits from their participation.

International cooperation in this field has always benefitted the contributing parties. International cooperation in this area was started by the OECD/NEA in the seventies; later that responsibility was transferred to the IAEA. Since 1981, the IAEA has been organizing Coordinated Research Projects (CRPs) on the behaviour of power reactor spent fuel during long-term (or extended) storage. The programmes were initially called BEFAST (BEhaviour of spent Fuel Assemblies in STorage), and later SPAR (Spent fuel Performance Assessment and Research), respectively.

The overall objectives of the BEFAST and SPAR CRPs were to develop a technical knowledge base on long term storage of commercial power reactor spent fuel through evaluation of operating experience and research by participating Member States, and to extrapolate predictions of spent fuel behaviour over long periods of time. In addition, other spent-fuel-storage-related topics, for example, the encapsulation of spent fuel for disposal, were also investigated.

This paper presents a selection of the findings related to spent fuel and storage system performance of the BEFAST and SPAR activities from 1981 until today.

2. HISTORY OF COOPERATION

Upon completion of each BEFAST or SPAR CRP, a summary of the results was published in an IAEA technical document (TECDOC) [Refs 1−6].

The first phase of the BEFAST project (1981-1986) involved 12 organizations from 11 countries: Austria, Canada, Czechoslovakia (CSSR), Finland, Federal Republic of Germany (FRG), German Democratic Republic (GDR), Hungary, Japan, Sweden, USA, and USSR.

A subsequent programme “BEFAST-II, Behaviour of Spent Fuel and Storage Facility Components during Long-term Storage” implemented during the years 1986-1991, involved organizations from 12 countries: Argentina, Canada, FRG, Finland, GDR, Hungary, Italy, Republic of Korea, Japan, UK, USA, and USSR.

BEFAST-III, implemented during the years 1991-1996 involved 15 organizations from 12 countries: Canada, Finland, France, Germany, Hungary, Japan, Republic of Korea, Russian Federation, Slovakia, Spain, UK and USA. There was also an observer from Sweden.

During the three BEFAST Co-ordinated Research Projects (CRPs), the participating countries contributed their R&D results on fundamental questions of spent fuel storage.

Towards the end of the BEFAST-III project, it became apparent that the R&D component of the project was decreasing steadily; more emphasis was being placed on the operation and implementation of storage technology. The storage technology (particularly dry storage) was undergoing a rapid evolution, new fuel and material design changes were coming on stream and target discharge burnup was steadily increasing; in addition, predicted storage times were expanding steadily. The increased burnup resulted in a higher heat load, higher amounts of fission gas and fission products, increased sheath (cladding) strains and an increased cladding oxidation and hydriding. Because of all the changes that occurred during the course of BEFAST-III, a subsequent project was proposed to address the impact of these new

parameters on long-term storage and to determine their consequences on subsequent operations, especially handling and transportation.

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

The second phase of the IAEA Spent Fuel Performance Assessment and Research (SPAR-II) CRP (2004-2008) involved the European Commission as well as 13 other organizations from 12 countries: Argentina (joined later), Canada, France, Germany, Hungary, Japan (2 participants), Republic of Korea (for a limited period), Slovakia, Spain, Sweden, UK, and USA. Sweden and the UK participated in the project as active observers.

The third phase of the IAEA Spent Fuel Performance Assessment and Research (SPAR-III) CRP (2009-2014) involved the European Commission as well as 13 other organizations from 11 countries: Argentina, France, Germany, Hungary, Japan (2 participants), Republic of Korea, Slovakia, Spain, Switzerland, United Kingdom (UK), and United States of America (USA, 2). Switzerland and the UK participated in the programme as active observers.

Overall, 20 countries and 1 international organization participated in one or more of the CRPs.

With the completion of the third phase of SPAR, it was considered timely to gather all the findings from the BEFAST and SPAR CRPs into one place. This activity is currently in progress and we look forward to issuing a consolidated technical document in 2015.

3. KEY OBJECTIVES

As already stated, the overall objectives of the CRPs were to develop a technical knowledge base on long term storage of power reactor spent fuel through evaluation of operating experience and research by participating Member States, and to extrapolate predictions of spent fuel behaviour over long periods of time. Specific research objectives included:

- Performing timely survey of the accumulated experience with spent fuel storage in participating Member States;
- Investigating potential fuel and storage components degradation mechanisms under wet and dry storage conditions;
- Assessing the potential impact of interim storage on subsequent spent fuel management activities (such as handling and transport);
- Developing a capability to assess the impact of potential deterioration mechanisms on spent fuel and storage components over extended periods of time;
- Exchanging results from surveillance and monitoring programmes of spent fuel storage facilities;
- Creating a synergy among research projects of the participating Member States;
- Facilitating technology transfer by documenting technical bases for spent fuel storage.

4. TECHNICAL RESULTS FROM BEFAST AND SPAR CRPS

Maintaining structural assembly and fuel cladding integrity is an important consideration. ‘Unfailed’ cladding acts as the primary containment barrier for both fuel pellets and fission products over the storage period and during subsequent fuel handling and transport

operations. This is valid for all the types of spent power reactor fuels considered: LWR, RBMK and WWER, AGR, Magnox and CANDU. For CANDU and some BWR fuel designs, cladding integrity may also be important for fuel bundle (assembly) integrity, because some of the fuel rods (pins) also act as structural elements.

4.1. Degradation mechanisms in Zirconium-based alloy claddings

Apart from a limited number of early fuel designs, all current LWR fuel rod claddings are made of zirconium-based alloys. In BWRs, Zircaloy-2 is used, whereas in PWRs and WWERs, zirconium-based alloys containing ~1% or less niobium (Nb) are the preferred materials. Zircaloy-4, which used to be the dominant choice for PWRs, is being progressively displaced by the newer Zr-Nb alloys. CANDU fuel has much lower burnups than LWR fuels; this results in smaller amounts of fission products and a lower decay heat, which leads to low storage temperatures. The effect of the lower temperatures is beneficial for most cladding degradation processes, except for some phenomena associated with hydrogen effects.

4.1.1. Wet storage

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

4.1.1.1. Uniform Corrosion

Uniform corrosion rates in pool water at 30°C to 45°C are a factor of ~1010 lower[1] than those under operational reactor conditions. Therefore, zirconium-based alloys are virtually immune to uniform corrosion during wet storage, at least under controlled chemistry water conditions.

4.1.1.2. Pitting, galvanic, and microbiologically-influenced corrosion (MIC)

Galvanic corrosion has not been observed in nuclear fuels when direct contact between Zircaloy and stainless steel occurs, either in-reactor or in wet storage. Zirconium-based alloys are not susceptible to pitting corrosion under water chemistry conditions existing in spent fuel pools. MIC of zirconium-based alloys has not been observed.

4.1.1.3. Hydriding

When the fuel is at storage pool temperatures, hydrogen in solid solution in the alloy is in the form of zirconium hydride, except possibly for the hydrogen that is associated with (or trapped in) irradiation defects such as vacancy dislocation loops. Hydrogen redistribution by thermal diffusion and formation of so-called pits by an Oswald ripening is not observed and can be ruled out during long-term wet storage at the pool temperatures in practice.

4.1.2. Dry storage

Potential degradation mechanisms that may affect cladding integrity of LWR fuels during dry storage and subsequent handling and transportation operations are out-lined below.

The first four mechanisms have the potential to result in through-wall cladding defects, while the last two are unlikely to result in failures during storage; rather, they have the potential to

[1] Putting this in context, the corrosion resulting from one full power day is equivalent to the corrosion that would occur over a period of several millions years in wet storage.

impair the ability of the cladding to effectively withstand potentially adverse mechanical challenges resulting from handling or transportation accidents.

4.1.2.2. Air oxidation

Inert atmosphere conditions during dry storage –preceded by appropriate drying and inerting operations– rule out the presence of significant amounts of air in contact with stored spent fuel. Therefore, oxidation is not an active degradation mechanism under normal inert storage conditions. Air oxidation is possible for off-normal or accident conditions resulting in the failure of the storage system seals. Two scenarios can be considered: (1) fuel rods with no through-wall defects; and (2) fuel rods with through-wall defects. Because oxidation rates of zirconium-based cladding by air are relatively low at temperatures typical of dry storage conditions[2], losses in cladding wall thickness are limited to a few percents, even for prolonged air exposure. Therefore, cladding remains an effective containment barrier protecting the fuel pellets from exposure to air for fuel rods that do not contain through-wall defects.

In case of cladding with through-wall defects, air will come in contact with the fuel inside the rod. Oxidation of spent fuel in air is a two-step process of the form:

$$UO_2 \rightarrow U_4O_9 \rightarrow U_3O_8 \qquad (1)$$

UO_2 oxidation rates remain very low below ~250°C. If the temperature exceeds 250°C, UO_2 conversion to U_3O_8 becomes possible. The formation of U_3O_8 results in a volume expansion of greater than 36%, which causes high stress levels on cladding, and gross cladding rupture, or "unzipping," may result.

4.1.2.2. Thermal creep

Spent fuel rods in dry storage undergo creep deformations at a rate determined by the internal fuel rod pressure, cladding temperature, and composition and metallurgical conditions of the rod cladding. When assessing cladding integrity during dry storage, thermal creep is considered as a potentially limiting degradation mechanism. It should be noted, however, that creep deformations tend to be self-limiting, because the creep-induced void volume expansion in a fuel pin leads to a decreasing cladding stress. As a result, the creep strain is unlikely to exceed the critical strain domain leading to plastic instability and pin rupture.

4.1.2.3. Stress Corrosion Cracking (SCC)

Because the combination of SCC agent and stress conditions required for crack propagation are normally absent, it can, therefore, be concluded that all fission-product-driven defect mechanisms such as stress corrosion cracking, uniform fuel rod internal fission product corrosion of the cladding, and localised fuel rod internal fission product corrosion of the cladding are not active.

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[2] Using the equation developed in The Corrosion of Zircaloy-Clad Fuel Assemblies in a Geologic Repository Environment [Hillner E., D.G. Franklin, and J.D. Smee, WAPD-T3173. 1998], it is predicted that only 20 µm of additional cladding would be lost for fuel held at 400°C for 100 days. The Hillner et al.'s equation was developed for steam corrosion, but the corrosion rates in steam and air are similar.

4.1.2.4. Delayed Hydrogen Cracking (DHC)

Although DHC has been observed in Zircaloy specimens that are sufficiently thick, DHC is not expected to be an active degradation mechanism in cladding tubes, given that the latter do not appear to have enough wall thickness to generate the conditions under which DHC might be expected to occur expected.

4.1.2.5. Hydride re-orientation

The degree to which spent fuel cladding, after a number of years in dry storage, can retain sufficient strength and ductility to resist catastrophic failures under post-storage handling and transport conditions may depend, among several factors, on the spatial distribution, morphology and orientation of the hydrides in the cladding. The cladding tube manufacturing process generates a texture in the cladding that causes the hydride-platelets to precipitate in a predominantly tangential orientation. Dry storage and/or transport operations may cause significant increases in cladding temperature. The temperature increase from pool conditions to dry storage / transport conditions results in hydride dissolution and in a corresponding increase of the hydrogen in solid solution in the zirconium alloy matrix up to the hydrogen solubility limit. The temperature increase also leads to an increase in the rod internal pressure. Subsequent cooling during dry storage or re-wetting after transportation causes hydrogen in solid solution in the cladding materials to re-precipitate, but possibly aligned in the radial and axial directions (or simply 'radial') due to the influence of the tensile hoop stress in the cladding caused by the internal rod pressure that is no longer compensated by external means such as the coolant pressure in the reactor. This can result in a mixed, circumferential and radial, hydride orientation, depending on temperature and cladding hoop stress histories. Such a hydride orientation can have significant effect on cladding mechanical properties and failure limits, especially in the lower range of temperatures (<150°C).

4.1.2.6. Hydrogen migration and re-distribution

Hydrogen can migrate from the high temperature to the low temperature region of the cladding as a result of the temperature profile that exists in a storage system. Hydrogen may precipitate in the colder regions in the form of hydrides, and consequently, lead to increased embrittlement, which may affect fuel integrity.

4.2. Degradation mechanisms in stainless steel based claddings

The cladding material used in the manufacture of Advanced Gas-cooled Reactor (AGR) fuel pins is 20Cr/25/Ni, which is a Nb-stabilised stainless steel. During reactor operation, stainless steel components in the 340-520°C temperature range (peak at 420°C) become sensitised, primarily as a result of radiation induced segregation.

4.2.1. Intergranular Stress Corrosion Cracking (IGA)

IGA is the prominent defect mechanism for AGR fuel cladding. Sensitisation, coupled with stress induced at the anti-stacking grooves of the fuel pin, can result in cladding failures for pins stored in moist air conditions or in pond storage environments (with high chloride ion concentrations). Radiation-induced segregation has been found to be the over-riding factor in AGR cladding sensitisation. Fast neutrons generate defects in metals. An atom can be knocked from its normal lattice site into an interstitial location and a vacancy is created. Vacancies are mobile and can be removed from the lattice at sinks. A grain boundary is a good sink. In moving in the lattice, a vacancy needs to exchange positions with another atom.

The diffusion coefficients of iron, chromium and nickel in austenitic steel are such that chromium diffuses fastest. The net effect is to move chromium away from the grain boundary. Nickel diffuses the most slowly of the three main steel constituents, and, as a result, nickel tends to become enriched at grain boundaries. The chromium depletion profile at a grain boundary (width of depleted zone and reduced chromium concentration on the grain boundary itself) is likely to influence the occurrence and rate of intergranular corrosion.

4.2.2. General corrosion

When storage durations become prolonged (i.e., hundreds of years), general waterside corrosion becomes a significant factor, especially where initial in-reactor cladding wall thinning has been significant. This failure mechanism also affects and can be dominant in the fuel element stainless steel structural materials (braces, etc.).

4.3. Magnesium-based (MAGNOX) cladding

4.3.1. Wet storage

In wet storage, Magnox cladding corrodes according to the reaction with water, forming $Mg(OH)_2$ and hydrogen. In principle, if the pH is increased to greater than 11.5, the solubility of the magnesium hydroxide layer is low, effectively forming a passive surface film on the metal. Research work, however, shows that this is not the full story and other factors, such as micro-cracking and the presence of carbon in the oxide layer, result in corrosion rates being up to three times higher than unirradiated material. Cladding corrosion is also influenced by the presence of water impurities, notably chloride, which leads to pitted areas even at low concentrations. Pitting corrosion may cause failure of the cladding during longer storage. Crevice corrosion between the splitter blades and the cladding may also occur. For the reasons outlined above, MAGNOX spent fuel is usually reprocessed within two years of reactor discharge; however, current storage practices show good performance and storage periods >5 years are feasible.

4.3.2. Dry Storage

Since the cladding was developed for CO_2-cooling during reactor operation, dry storage under CO_2 is suitable for long term storage. Corrosion rates in CO_2 are negligible for temperatures below 350°C. Storage in air is feasible when the fuel temperature is below 150°C; the safety case contains restrictions due to uranium hydride ignition if bare metal was present. In the case of dry MAGNOX fuel stored in dry air, fuel deterioration will occur through pitting attack, if relative humidity is >50%, or if there is water ingress.

5. DISCUSSION

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

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

Both the fuel and the environment were monitored in different programmes by using both conventional and special methods. In addition, development work on experimental procedures, such as corrosion monitoring, was undertaken by the participants.

Monitoring and surveillance confirmed that adherence to the specified pool water chemistry was essential to prevent fuel degradation during pool storage for all types of spent fuel. Near-neutral pH conditions with low ion content are satisfactory for zirconium-alloy-clad fuels. Elevated pH is beneficial for Magnox and AGR fuels. For Al-clad fuels, a pH of ~5.5 is used.

Defected fuel, in some cases, was stored in canisters; however, no cladding deterioration or significant fuel losses or changes in defects were detected. In pool components, experience on liner and rack materials indicated no deterioration under proper water chemistry conditions.

The following summaries of wet and dry storage performance reflect the experience gained from the programme participants.

5.1. Spent fuel wet storage performance

The storage of all zirconium-alloy-clad spent fuel is a safe and reliable process. It is licensed in all BEFAST/SPAR-member countries. Positive experience is available for up to 50 years of storage. It is predicted that fuel integrity will be maintained for very long times as long as pool water chemistry is properly controlled. Even when the fuel assembly contains defects that occurred during irradiation, it can also be stored in pools for extended time periods; as shown by long-term monitoring studies on defected fuel; these studies did not show any significant degradation with time.

For extended wet storage, the behaviour of different fuel materials (e.g., non-zirconium-alloy cladding) and pool construction materials (e.g., epoxy lining) may require further investigation. Capacity enhancement (rod consolidation) may require additional work. It may be that pool component performance could, in some cases, be more limiting than fuel assembly performance. It may be desirable to increase efforts on corrosion monitoring and prediction of the long term behaviour of non-zirconium based fuel rod cladding and pool components.

Spent fuel dry storage performance

The storage of zirconium-alloy-clad spent LWR fuel in an inert-gas atmosphere (preferably He) is a widely implemented technology. Today, around 100 dry storage systems have been deployed worldwide. Dry storage of that fuel is licensed in Germany for temperatures <380ºC and in the USA for T < 410ºC (but more typically for T < 400ºC). Experimental data are available up to over 500ºC. To optimize compliance with peak cladding temperature limits, the irradiation history (especially burnup) and cooling time of the fuel must be considered; well-benchmarked heat transfer codes for predicting cladding temperature would be beneficial. Positive experience with dry storage is available for storage periods of more than three decades. Storage in an inert atmosphere of zirconium-alloy-clad spent fuel that defected during irradiation has been licensed in a few countries. Degradation of fuel assemblies due to storage is unlikely. No significant fuel degradation in dry storage has been reported. A few indications of krypton release have been observed, indicating possible individual rod failures, but these suspect fuels have not been examined to determine the cause of the failures.

Defect-free fuel would be expected to have good storage performance, but the possibility of defect propagation in fuel that defected during irradiation would significantly limit the

maximum storage temperature. Dry storage of spent CANDU fuel in air at temperatures up to 160ºC has been licensed in Canada.

Dry storage will also continue to be used for Al-clad and Magnox-type fuels. Such storage has been licensed in the UK (for CO_2 and air environments, depending on the fuel temperature) and in Italy (with nitrogen cover gas). Difficulties are envisaged with dry storage that is implemented after wet storage of irradiated Magnox fuel.

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

6. CONCLUSIONS

The accumulated reporting of the IAEA's BEFAST and SPAR CRPs now exceeds 30 years. The fundamental R&D questions have been answered to the degree that licensing of both wet and dry storage is presently possible in most countries requiring additional spent fuel storage facilities.

For zirconium and stainless steel based LWR clad spent fuels there is >50 years wet storage and >30 years dry storage experience. Performance in storage remains excellent with no generic failure mechanism identified or experienced. In the case of stainless clad AGR spent fuels there is >30 years of wet storage experience. Storage performance is good provided the fuel is stored in the presence of corrosion inhibitor.

A review of degradation mechanisms in both wet and dry storage has not revealed any time limited failure mechanisms in fuel cladding or fuel assembly structures provided that recommended storage environments are maintained. Thus it can be concluded that interim spent fuel is safe in both wet and dry conditions.

The key objectives of spent fuel storage related research and development have changed, since the basic questions of material science have been solved. Currently, work is focused mostly on the amelioration of technology as well as on improvements in storage economy. This shows that spent fuel storage is approaching a mature technology in the back end of fuel cycle.

The areas that have been identified to date as requiring further considerations are:

- Impact of long-term storage on cladding properties in zirconium based alloy materials in relation to fuel retrieval and handling operations;
- Ageing management and issues related to stress corrosion cracking (SCC) in dry storage systems.

Specific conclusions are:

- On-going review and assessment of the results from national plans on the research of materials used in spent fuel storage has been undertaken;
- The results from national research efforts on potential fuel materials deterioration over long-term storage of fuel have been compiled;
- National experience on spent fuel storage has been reported;
- Relevant results from surveillance programmes have been reviewed and assessed;
- Operating experience on spent fuel handling and its potential impact on spent fuel integrity have been shared.

The BEFAST and SPAR CRPs have demonstrated that international co-operation can successfully be accomplished, even with a limited budget, and may provide major benefits to participating countries. Through the IAEA, these results are also made available to developing countries.

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A NEW MODEL ON RADIAL HYDRIDE PRECIPITATION IN ZIRCALOY-4 CLADDINGS UNDER DECAYING STRESS AND TEMPERATURE TRANSIENT

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Abstract

In the context of spent fuel transportation and storage studies, the radial hydride precipitation is considered as a important issue for SRA Zircaloy-4 claddings. The fuel residual thermal power induces a temperature increase within the shipping cask cavity. As a consequence, the zirconium-hydrides contained within irradiated fuel claddings dissolve as solid-solution hydrogen. After transportation and during unloading in the spent fuel storage pool, the fuel rods are cooled down under internal gaseous pressure inducing mainly hoop stress within the cladding. At low hoop stress level, the precipitated hydrides are platelets oriented in the hoop and axial direction. When the hoop stress exceeds a threshold value, the precipitated hydrides rather develop along the radial-axial direction. These so-called radial hydrides strongly embrittle the cladding. Laboratory experimental studies were performed to characterize the degree of radial hydride precipitation at room temperature under constant applied stress after a slow cool-down. The model is successfully compared to reference experimental data. However, considering fuel assemblies transportation, the internal pressurization induced cladding stress decreases during the fuel cool down. There is a need for differential models providing step by step prediction of the radial hydride precipitation during the fuel cool down. A new modeling, describing the influence of an applied stress along with a temperature transient, has been consequently developped and compared to experimental data.

1. INTRODUCTION

The thorough knowledge of precipitation of radial hydrides in the fuel-rod cladding, impeding its fracture thouhgness, and of the key parameters controlling this phenomenon, is needed to conduct safety analysis of wet and dry storages and to evaluate more specifically the consequences of accident scenarios liable to occur in these facilities (dropped spent fuel assembly…) or during shipment performed after a long storage period.

A new thermal-mechanical test, so-called C-shaped ring compression tests (CCT), was developped by IRSN to determine the stress thresholds for radial hydride precipitation during cladding cool down [1, 2]. These mechanical tests develop distributed hoop stress field within the sample section. Under constant compression load, the stress field (covering a broad range of stresses) remains constant at any location of the sample providing an easy approach to characterize the hoop stress thresholds.

A simple theoretical approach was proposed to translate these stress thresholds, first into a simple modeling under constant applied stress during the cool down and second, in a differential model applicable to time-varying applied stress.

In order to study the influence of time-decreasing hoop stress, it is easier to use a mechanical test developping rather spatially uniform stress distribution. For this sake, ring tensile tests (RTT) with machined gage sections were used. Under constant applied load, the stress is close to be uniform in the gage section. This property was used to generate experimental data for modeling validation.

2. HOOP STRESS THERSHOLDS FOR RADIAL HYDRIDE PRECIPITATION

The CCT tests are subjected to a constant load and a heat treatment with typically 24°C per hour cooling down rate [2]. The load is determined to obtain a maximum stress below 230 MPa. Influence of hydrogen contents [H] and maximum heat treatment temperature (T_M) were studied. The stress thresholds were determined using post-test metallographic results at room temperature. The lower stress threshold below which only circumferential hydride precipitate ($\sigma_{\theta\theta,0\%}$) is determined after any test. Differently, the upper stress threshold above which only radial hydride precipitate ($\sigma_{\theta\theta,100\%}$) can only be determined for low hydrogen contents associated to full dissolution of pre-existing hydrides during the heat treatment.

$$\sigma_{\theta\theta,100\%}(MPa) = 110 + 65\left(1 - e^{-\frac{[H](wppm)}{65}}\right) \pm 20 \quad (1)$$

$$\sigma_{\theta\theta,0\%}(MPa) = 0.0200[H](wppm) + 0.3862Min[T_M(°C); T_{SSD}(°C)] + 186.9 \quad (2)$$

T_{SSD} is the dissolution temperature of $[H](wppm)$ during heat-up. It was experimentally determined by Kearns [3] for Zircaloy-4:

$$T_{SSD}(K) = -\frac{4162}{ln\left(\frac{[H](wppm)}{99000}\right)}$$ and inversely, the H solubility is: $$C_{SD}(wppm) = 99000.e^{-\frac{4162}{T_{SSD}(K)}} \quad (3)$$

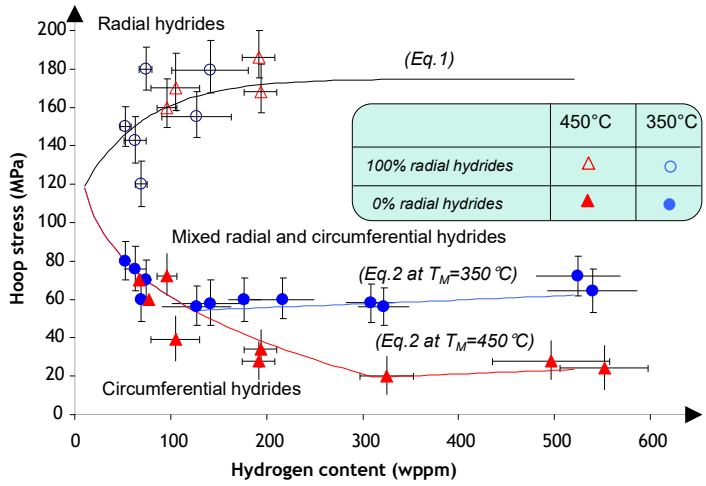

FIG.1. Hoop stress threshold for radial hydride precipitation.

3. MODELING OF RADIAL HYDRIDE PRECIPITATION UNDER CONSTANT STRESS

A linear interpolation model was proposed [2] to determine the radial hydride fraction, fR, after a heat treatment with maximum temperature TM under constant hoop stress, $\sigma_{\theta\theta}$:

$$f_R = \frac{Min([H](wppm); C_{SD}(T_M))}{[H](wppm)} \cdot \frac{\sigma_{\theta\theta} - \sigma_{\theta\theta,0\%}}{\sigma_{\theta\theta,100\%} - \sigma_{\theta\theta,0\%}} \text{ for } \sigma_{\theta\theta,0\%} \leq \sigma_{\theta\theta} \leq \sigma_{\theta\theta,100\%} \quad (4)$$

This model was successfully compared to radial hydride fraction observed after pressurization tests cooled down under constant pressure [2].

4. MODELING OF DIFFERENTIAL RADIAL HYDRIDE PRECIPITATION

The ratio, f, between radial hydrides, dH_R, formed during the precipitation of dH_P hydrogen at T_{SSP} temperature under $\sigma_{\theta\theta}$ hoop stress, can be derived from the hysteresis between dissolution (T_{SSD}) and precipitation (T_{SSP}) temperatures :

$$f(T_{SSP};\sigma_{\theta\theta}) = \frac{dH_R}{dH_P} = \frac{\frac{\partial[f_R(T_{SSD};\sigma_{\theta\theta}).c_{SD}(T_{SSD})]}{\partial T_{SSD}}}{\frac{\partial[c_{SD}(T_{SSD})]}{\partial T_{SSD}}} \quad (5)$$

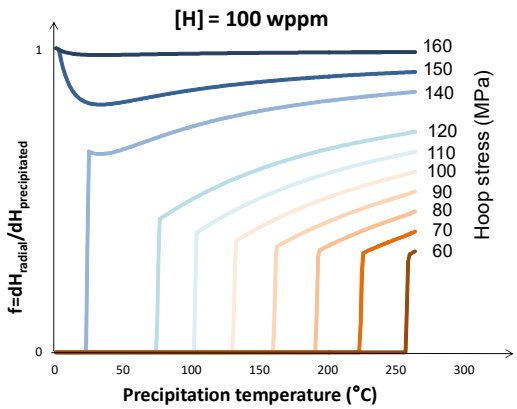

FIG. 2. Influence of hoop stress and precipitation temperature on the differential radial hydride precipitation in a sample containing 100 wppm hydrogen.

Fig. 2 illustrates the typical trend obtained for differential radial hydride precipitation in a sample containing 100 wppm subjected to hoop stresses in the range of radial hydride precipitation.

The proposed differential model can be used to determine the degree of radial hydride precipitation after a heat treatment with decaying stress level.

5. TESTING AND INTERPRETATION OF A THERMAL-MECHANICAL TRANSIENT WITH DECAYING STRESS

Pre-hydrided Zircaloy-4 tubes, with hydrogen content close to 100 wppm, were machined to form Ring Tensile samples with machined gage section (see Fig. 3). The samples were then subjected to a cool-down under applied hoop stress. Two loading conditions applied to the sample were adressed: constant load and decreasing load. In the second situation, the applied load was divided by two after approximately 50 wppm hydrogen precipitation. Post-test metallography was used to determine the radial hydride distribution within the tested samples as illustrated in Fig. 4. Image analysis was used to quantify the degree of radial hydride precipitation.

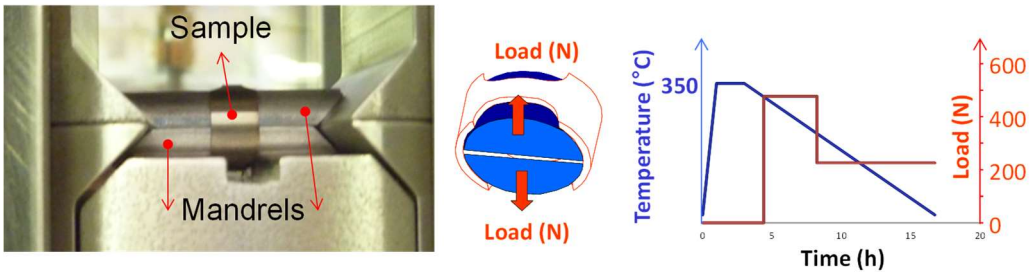

FIG.3. Loading device and applied radial hydride precipitation transient with decaying stress.

The thermal mechanical transient is further modeled using finite element simulation. A three dimensionnal modeling of the ring tensile test is performed and then the radial hydride precipitation along the gage section is modeled using the proposed incremental model. An acceptable consistency between modeling results and experimental results is obtained in spite of a lack of accuracy in the stress determination during ring tensile tests loaded with inserted mandrels.

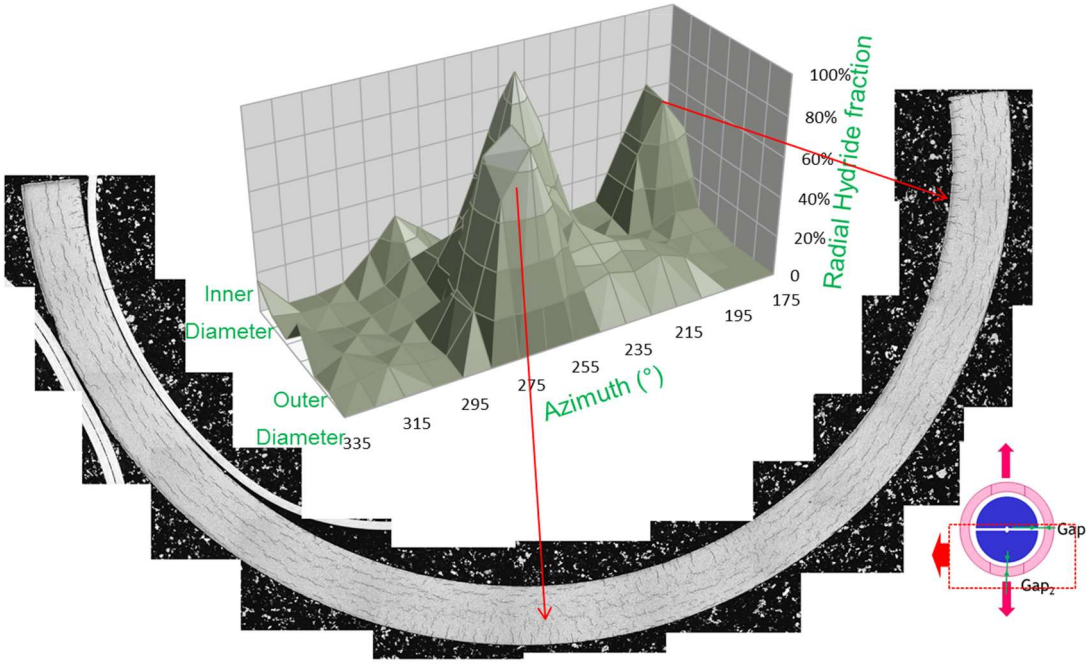

FIG.4. Post-test metallography analysis of RTT after radial hydrides heat treatment.

6. CONCLUSION

A modeling describing differential radial hydride precipitation at constant temperature has been derived from experiments. Validation experiments were performed using RTT with machined gage section using constant or decaying stress during the sample cool down. The comparision between finite element simulations and experimental results qualitatively confirms that the proposed approach is promising. However the stress within the sample

cannot be determined with sufficient accuracy using this mechanical test. Complementary testing is required for full modeling validation.

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EVALUATION OF RADIATION CHARACTERISTICS OF SPENT RBMK-1500 NUCLEAR FUEL STORAGE CASKS DURING VERY LONG TERM STORAGE

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Abstract

The present spent nuclear fuel management concept in Lithuania foresees dry spent nuclear fuel storage in casks for 50 years. During this time the final management concept shall be developed. Different options are under consideration; however, due to various uncertainties in future, there is a risk that these options cannot be implemented in due time, therefore elongation of the spent nuclear fuel storage for a period over 50 years shall be also considered. The existing spent nuclear fuel storage facility at the Ignalina Nuclear Power Plant site was extended for several times and currently is filled up to its final capacity with 20 metal CASTOR® RBMK-1500 and 100 reinforced concrete CONSTOR® RBMK-1500 casks. Modelling of radiation doses on the sidelong, upper and lower surfaces of theses dry storage casks and for certain distance during very long term storage has been performed using SCALE computer codes system. Obtained results showed that a CONSTOR® RBMK-1500 cask has better shielding properties than a CASTOR® RBMK-1500 cask. Material composition and thickness of side walls and bottoms of these casks are different, therefore total equivalent dose rates at casks' side-walls and bottoms differ from 2 to 7 times during storage periods up to 50 years. During a very long term storage period (up to 300 years) the total dose rate is significantly decreasing, the difference of dose rates for different casks is increasing up to 60 times and after such period dose rate caused by neutrons is becoming dominant for both casks.

1. INTRODUCTION

The Ignalina NPP contains two RBMK-1500 water-cooled graphite-moderated channel-type power reactors. Unit 1 was in operation from 1983 till 2004 and Unit 2 was in operation from 1987 until 2009. After the final shutdown of these units the total amount of spent fuel assemblies (SFA) to be stored is about 22 000 assemblies. Dry storage concept was introduced in 1992 when it was decided to build an interim dry spent nuclear fuel (SNF) storage facility at the Ignalina NPP site. In 1999 the first cask was loaded with the SNF and transferred to the dry storage facility. The facility was extended for several times and currently is filled up to its final capacity with 20 metal CASTOR® RBMK-1500 and 100 reinforced concrete CONSTOR® RBMK-1500 casks each containing 51 SFA (6120 SFA in total). For the storage of the remaining SNF (approx. 16 000 SFA) a new dry storage facility at the Ignalina NPP site currently is under construction. SNF in the new facility will be stored in concrete CONSTOR® RBMK1500/M2 casks. These new casks are more capacious and can store up to 91 FA within several baskets of special configuration.

The present SNF management concept in Lithuania foresees SNF dry storage in casks for 50 years. During this time the final SNF management concept shall be developed. Different options are under consideration; however, due to various uncertainties in future, there is a risk that these options cannot be implemented in due time, therefore elongation of the SNF storage for a period over 50 years shall be also considered. Safety analysis reports for CASTOR® RBMK-1500 and CONSTOR® RBMK-1500 and some numerical investigations in research institutes were performed assuming a 50 year storage period [1, 2]. Estimation of radiation characteristics of the existing CASTOR® RBMK-1500 and CONSTOR® RBMK-1500 dry storage casks for a very long term (up to 300 years) has been performed and the obtained results are presented and discussed in this paper.

2. DRY STORAGE CASKS FOR RBMK-1500 SNF

The CASTOR® RBMK-1500 (Fig. 1) has a cylindrical body made of ductile cast iron with a height of more than 4 m, diameter – about 2 m and wall thickness about 0.3 m. The cask is closed with a leak tight bolted double lid system. The mass of a loaded CASTOR® RBMK-1500 cask is about 75 tons.

The sidewall of a CONSTOR® RBMK-1500 cask (Fig. 1) consists of inner and outer steel cylinders. The space between these cylinders is filled with reinforced heavy concrete. The bottom of the cask body is similar – the space between the two steel plates is filled with reinforced heavy concrete. The total thickness of the sidewall is 0.43 m. The leak tight lid system consists of a bolted lid and two welded lids. The mass of a loaded CONSTOR® RBMK-1500 cask is about 88 tons.

The inner overall dimensions of a CONSTOR® RBMK-1500 cask are the same as of a CASTOR® RBMK-1500, but from the outside the metal-concrete cask is bigger. SFA loading into a cask is performed under water at SPH. The loaded cask is closed with lid and after pumping-out of water, drying-out and vacuumization operations, the cask cavity is filled with helium.

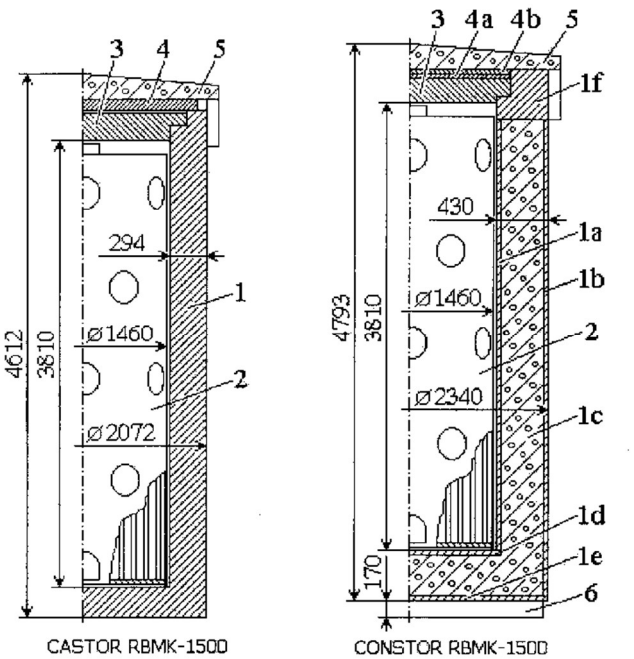

FIG. 1. Cross-sections of the SNF dry storage casks (1 — cask body (1a, 1b, 1d, 1e, 1f-metal parts; 1c-heavy concrete); 2 — basket; 3 — cask lid; 4 (4a, 4b) — guard plates; 5 — concrete cover; 6 — damper).

Reactor RBMK-1500 was design for nuclear fuel with 2.0% Uranium-235 enrichment from the outset. The Ignalina NPP used this type of fuel till June 1995, then the new uranium-erbium fuel (2.4%, 2.6%, 2.8% enrichment) was implemented at the Ignalina NPP. Only intact and 2.0% ^{235}U enriched fuel is allowed for storage in CASTOR® RBMK-1500 and CONSTOR® RBMK-1500 casks. The loaded casks are transferred to the dry storage facility and are placed there on a concrete platform fenced with a shielding concrete wall in the open air.

3. CALCULATION METHOD AND ASSUMPTIONS

Modeling of radiation characteristics for dry storage casks deals with two tasks. Firstly, the characteristics (concentrations of fission products, actinides, neutron and gamma source emissions, etc.) of an irradiated fuel assembly during a very long term storage period shall be estimated. Secondly, when properties of SNF are obtained, dose rate calculations on the surface and at some distance from the casks shall be performed.

Results of the first task, i.e., numerical modelling of spent RBMK-1500 nuclear fuel characteristics, are comprehensively presented in [3]. Modelling was performed using SCALE computer code system TRITON module. According to the results, the initial enrichment of RBMK-1500 fuel does not have significant effect on gamma and neutron emission rates and energies (the results differ by 10–20%). Relatively rapid decrease of gamma emissions in different energy groups is observed during a 50 year period; for neutron emission this period is 100 years. After 300 years of storage, low energy gammas are dominant in the emissions and, contrary, for the neutrons – higher energy neutrons are dominant. Fig. 2 shows activity variations of actinides and fission products in spent RBMK-1500 nuclear fuel with different ^{235}U enrichments. The main actinides with the highest activity values for all fuel enrichments are the same and are as follows: ^{239}Np, ^{237}U, ^{238}Np, ^{241}Pu, ^{242}Am, ^{242}Cm, ^{244}Cm, ^{238}Pu and ^{240}Pu. In case of fission products, ^{137}Cs, ^{137m}Ba, ^{147}Pm, ^{90}Sr, ^{90}Y, ^{85}Kr, ^{125}Sb, ^{121}Sn and ^{154}Eu have the highest activity values. The activity of actinides decreases by two orders of magnitude during a 5 year storage period (Fig. 2, a); afterwards, the activity decreases by one order of magnitude after a 100 year period, and after that the activity decrease is insignificant. In the case of fission products (Fig. 2, b), their activity rapidly decreases approximately by two orders of magnitude during a 10 year storage time; afterwards the activity decreases gradually also by two orders of magnitude during a 10–300 year period. The differences between actinides activities for fuel with 2.0% and 2.8% enrichments are about 15%, and for fission products – about 20%.

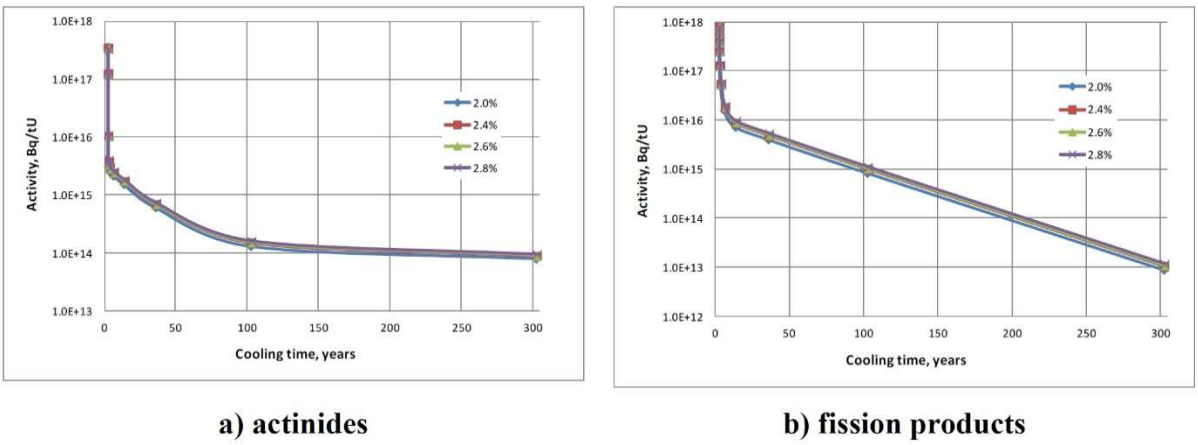

FIG. 2. Activity variations during SNF a very long term storage period [3].

Sequence SAS4 from SCALE 5.1 computer code system was used for solution of the second task, i.e., dose rate calculation on the surface and at some distance from the casks and estimation how these doses change during a very long term storage period. SAS4 calculates radiation doses exterior to a cask using a three-dimensional Monte Carlo method. SAS4 input

data are nuclide composition of an irradiated and cooled fuel assembly; material composition and geometry of a storage cask; temperatures; radiation source characteristics; locations of radiation dose detectors.

The following main assumptions for dose rate calculations were accepted:

- 3-D description of cask geometry in SAS4 code;
- Source emission was modelled as a homogenous cylindrical body which contains 102 half assemblies (with 2.0% ^{235}U enriched fuel) and basket internals;
- Locations of point radiation dose detectors were in the middle of the sidelong surface, in the center of the lid and the bottom. Location distances – 0, 1 and 2 meters;
- Calculations were performed for a storage period up to 300 years.

4. MODELLING RESULTS AND DISCUSSION

Dose rate modelling results on the surface and at some distance for the CASTOR® RBMK-1500 and CONSTOR® RBMK-1500 storage casks during a very long term storage period are presented in Figs 3–5. Dose rate values for a CONSTOR® RBMK-1500 cask in all directions are less than for a CASTOR® RBMK-1500 cask. This is also affirmed by the existing operational experience. For instance, the mean individual dose for the personnel performing cask loading and handling activities at the Ignalina NPP reactor unit was 0.65 mSv per one CASTOR® RBMK-1500 cask and 0.86 mSv per cask for personnel working at the dry storage facility. In case of a CONSTOR® RBMK-1500 cask, the mean individual dose was 0.20 mSv and 0.26 mSv per cask for the personnel at the reactor unit and the dry storage facility, respectively.

The smallest difference of total equivalent dose rate (see Fig. 3) for different storage casks is at the lid points. This is because the lid systems are made from the same material for both casks. Only total thickness of the lid system for a CONSTOR® RBMK-1500 cask is a little bigger.

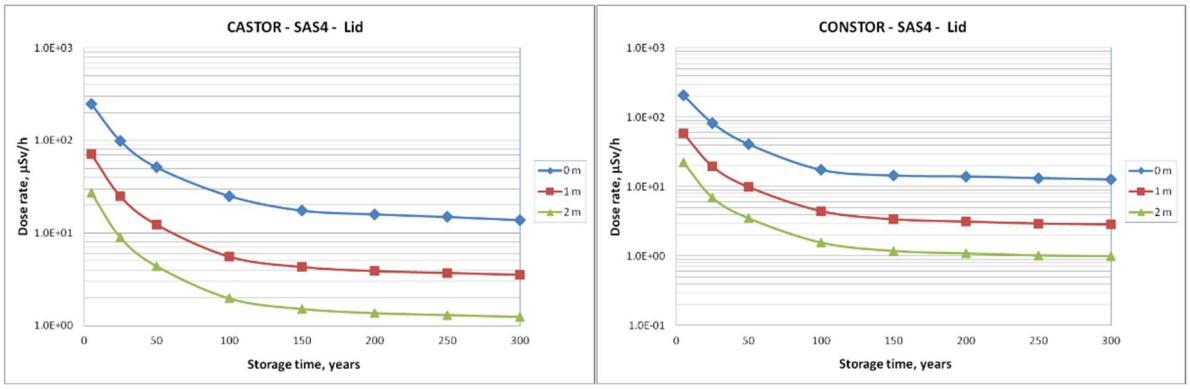

FIG.3. Total dose rate variation during long term storage (cask lid).

Material composition and thickness of the casks side walls and bottoms are different, therefore total equivalent dose rates at cask side walls and bottoms (see Fig. 4 and 5) differ from 2 to 7 times during storage periods up to 50 years; afterwards the difference of dose rates for different casks is increasing up to 60 times. This is determined by material composition of

side walls and bottoms of a CONSTOR® RBMK-1500 cask, which is effectively absorbing the neutrons.

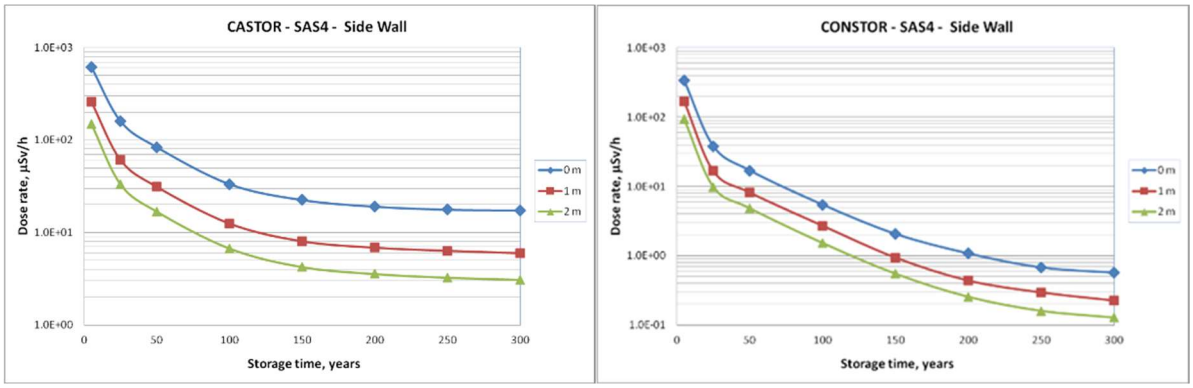

FIG. 4. Total dose rate variation during long term storage (cask side wall).

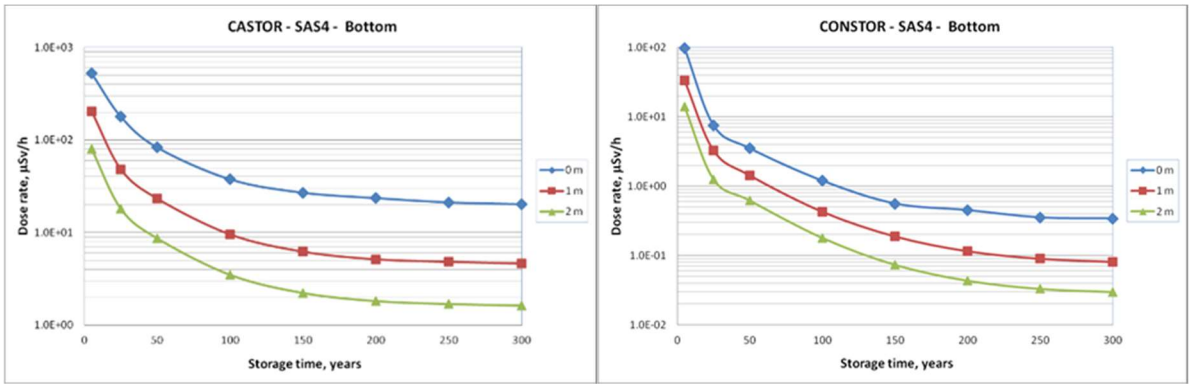

FIG. 5. Total dose rate variation during long term storage (cask bottom).

The total equivalent dose rate is formed by neutrons and gamma radiation. Variations of the percentage of each component during long term storage for CASTOR® RBMK-1500 and CONSTOR® RBMK-1500 casks are presented in Figs 6–8. Influence of neutrons on total equivalent dose rate for both casks is the same at the lid points (Fig. 6) but completely different at the side walls (Fig. 7) and the bottoms (Fig. 8). At the beginning of storage gamma radiation is the dominating component at the side walls and bottoms for both casks.

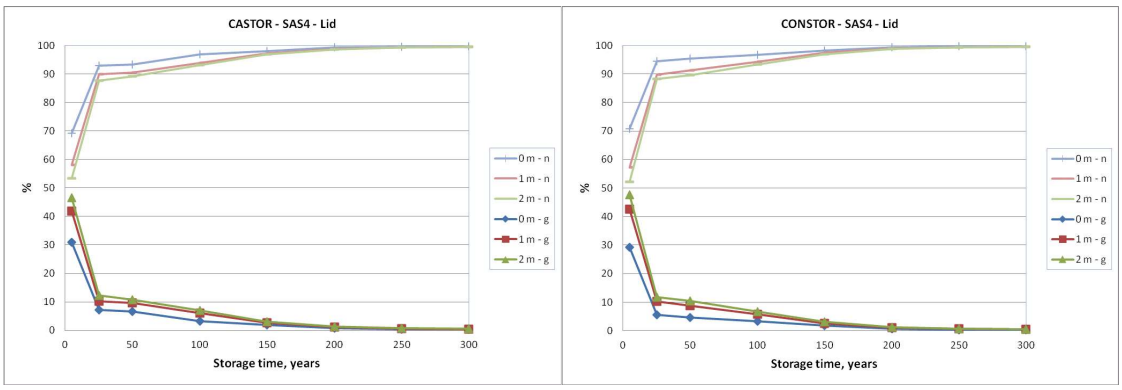

FIG. 6. Gamma (g) and neutron (n) share of total dose rate (cask lid).

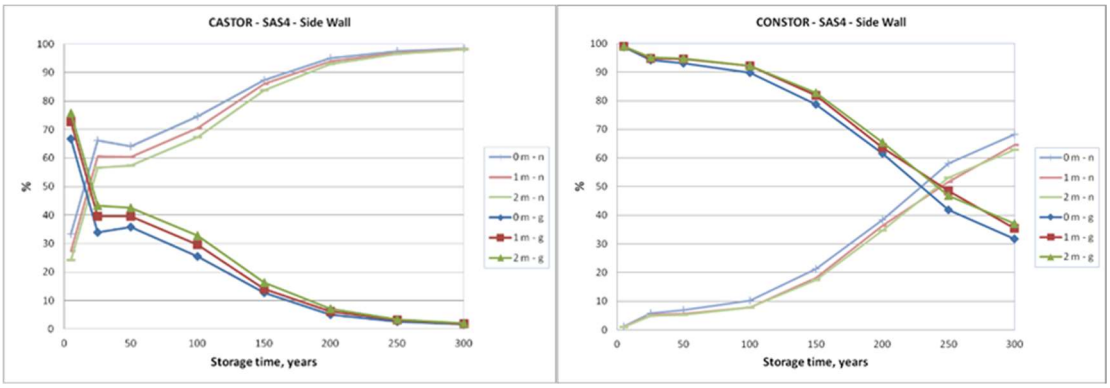

FIG. 7. Gamma (g) and neutron (n) share of total dose rate (cask side wall).

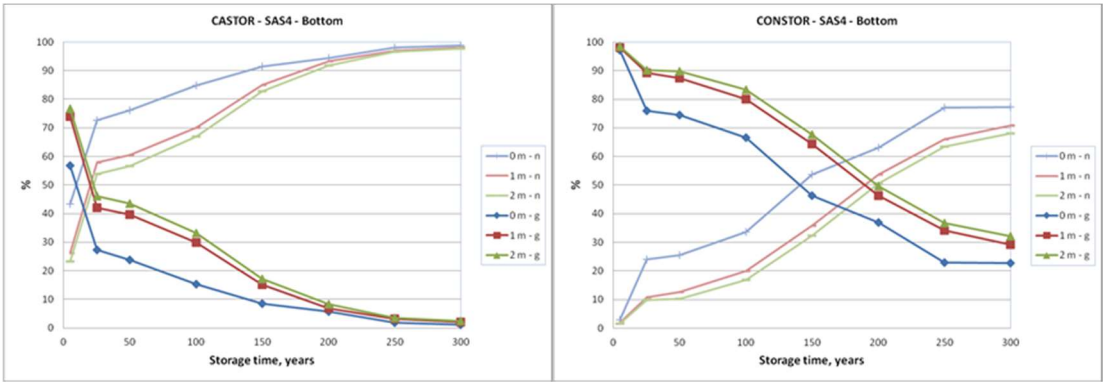

FIG. 8. Gamma (g) and neutron (n) share of total dose rate (cask bottom).

5. CONCLUSIONS

Dose rates on the surface and at some distance for the ductile cast iron CASTOR® RBMK-1500 and heavy concrete CONSTOR® RBMK-1500 casks loaded with RBMK-1500 spent nuclear fuel were calculated using SAS4 sequence of SCALE 5.1 computer code system. These casks are designed for an interim storage (up to 50 years) of spent nuclear fuel at the Ignalina NPP. The dose rate calculations were performed on the sidelong, upper and lower surfaces of the casks and for certain distance at the beginning of SNF storage and after 300

years of very long term storage. The obtained results show that dose rate values on the surface of both casks are less than the design criteria value 1000 μSv/h. Calculations and existing operational experience revealed that a CONSTOR® RBMK-1500 cask has better shielding properties than a CASTOR® RBMK-1500 cask. During 300 years of interim storage, the total equivalent dose rate is significantly decreasing and after such period dose rate caused by neutrons is becoming dominant.

ACKNOWLEDGEMENTS

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CRIEPI'S STUDIES ON THE SCC OF THE CANISTER FOR SPENT NUCLEAR FUEL

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Abstract

To put the concrete cask in practical use in Japan, stress corrosion cracking (SCC) of canister must be coped with. It is required to take measures for one or two of the three factors, i.e. welding residual stress, material, and environment, to cope with the SCC that may result in loss of the containment function of the canister. Prevention of loss of containment due to SCC of a canister was evaluated either by a method of comparing the amount of salt on the canister surface during storage with the minimum amount of salt to initiate rust and SCC or by a method of comparing the wetting time of the canister surface under salty-air field environment with the lifetime of the SCC fracture of the canister material. We examine an application of the zirconia shot peening as the residual stress improvement processing that is considered for prevention of SCC occurrence of the canister weld effectively. For the inspection of the canister integrity during storage, there are several candidate methods. We develop a technique to measure chlorine deposited on the surface of canisters by laser-induced breakdown spectroscopy.

1. INTRODUCTION

Considering the uncertainties of the roles of nuclear power generation in Japan, the necessities of introducing interim storage facilities of spent nuclear fuel at or away from reactor sites will be still increasing. Although the use of metal casks for spent fuel storage has been practically employed to date, the utilization of an alternative storage method, i.e. the concrete cask is also needed from the perspective of cask procurement risk management and economic benefit. Since Japan is an island nation, chloride induced SCC of welded stainless steel MPCs in the concrete cask storage system should be evaluated and tackled. The MPC shall not lose its containment function due to SCC during long-term storage. The SCC occurs when stainless steel with welding residual stress is exposed to a salty air environment.

> "*Measures for one or two of the three factors must be taken, i.e. welding residual stress, and environment, to cope with the SCC that may result in the containment function of the MPC being lost. We evaluated methods to maintain its confinement function of the MPC made of the normal stainless steel SUS304L materials during storage period by comparing the amount of salt on the MPC surface during storage with the minimum amount required to initiate SCC, and by comparing the humid time of the MPC surface under a field environment with the lifetime of the SCC fracture of the MPC material*" [2].

Moreover, an alternative measure against SCC was investigated by reducing the welding residual stress of the conventional stainless steel MPC and demonstrated by SCC tests using mock-up MPC made of SUS304L simulating wall thickness of the commercialized MPC and welding methods.

2. SCC EVALUATION METHOD [1, 2]

In Fig. 1, we propose a scenario of SCC evaluation. The initiation of SCC will start when the surface chloride density exceeded a critical value. Such critical chloride density will be obtained through parametric material testing as a function of chloride density on the surface

of the MPC material. Stainless steels become immune to SCC when the temperature decreases to a certain critical value (TL) below which the anodic process of atmospheric corrosion is negligible as shown in Fig. 2 (a). The SCC initiates and propagates only during a humid period of storage. Humid period is defined by the time when RH around the MPC surface is higher than the critical value (RHL) to deliquesce sea salt particles adhering to the MPC surface. Thus, SCC may grow during the period corresponding to the sum of the shaded areas (Σtcor.) as shown in Fig. 2 (b). Hence, the Crack Growth Depth (CGD) can be obtained by multiplying the Crack Growth Rate (CGR) with humid time. If CGD did not exceed an allowable value to maintain the confinement of the MPC, the storage of MPC can be continued.

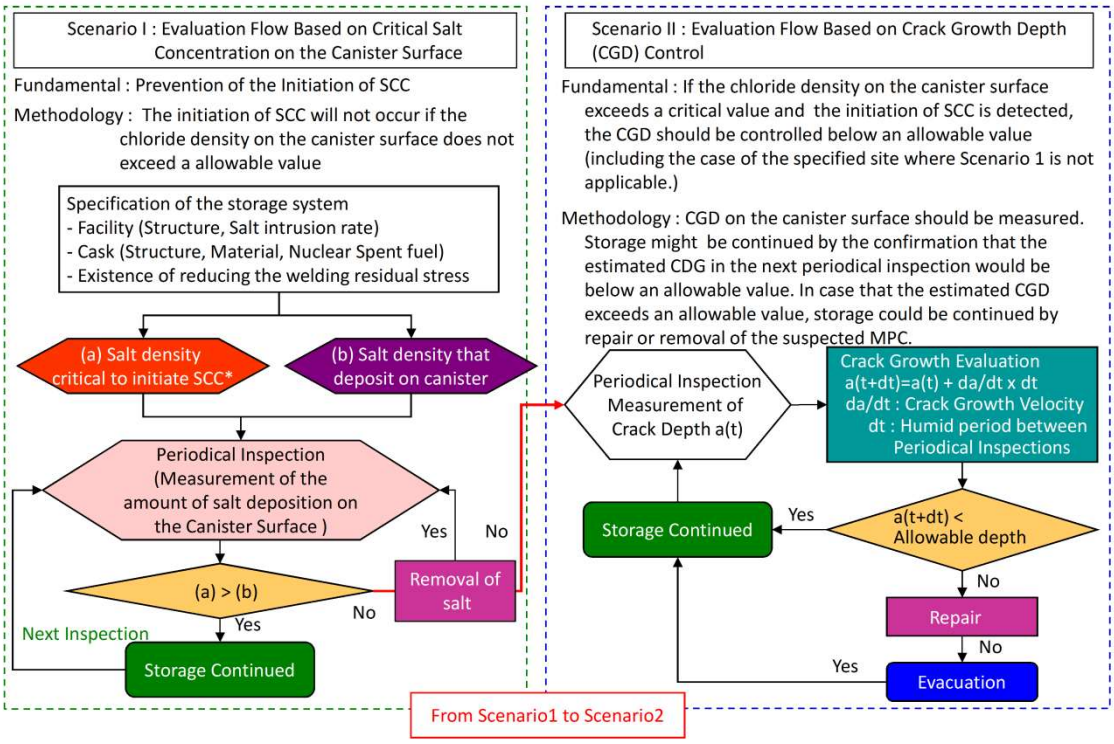
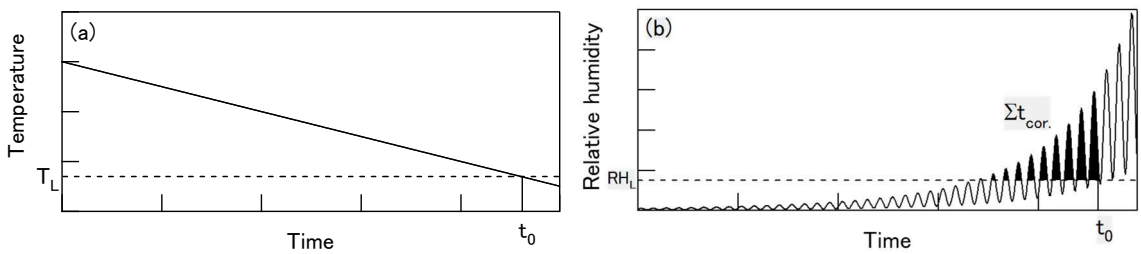


FIG. 1. Methodology of the SCC evaluation.

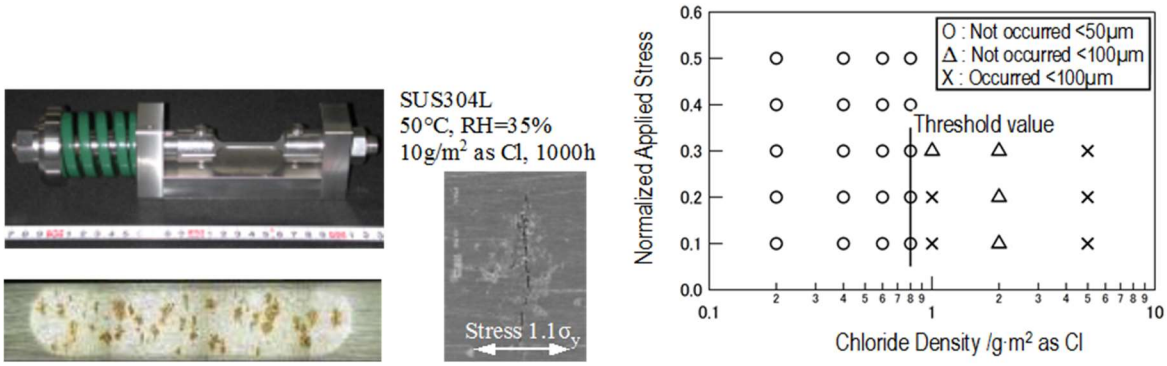


(Temperature profile of canister surface) (Change of RH with time around canister surface)

FIG. 2. Method for evaluating humid period.

3. EVALUATION OF SCC INITIATION CRITERION [1, 3]

To know the SCC initiation criterion, we performed constant-load SCC tests using a spring shown in Fig. 3. We used SUS304L and SUS316L materials of 2 mm thickness as test specimens. Salt was supplied intermittently by using a nozzle which could spray fine particles. The test temperature and relative humidity were determined by taking account of the actual environment based on the domestic climate observation data. The domestic maximum absolute humidity is about 30 g/m^3 and the relative humidity and temperature at which sea salt causes SCC in an atmospheric environment SCC are 35% and 50°C respectively. We set the temperature of 50°C and the relative humidity to 35% in the test. Fig. 3 shows the relation among the SCC test results, stress and deposited salt concentration for each of SUS304L materials. For the SUS304L material, no cracks or significant load stress dependency was observed if the salt concentration was lower than 0.8 g/m^2 as Cl. So, we set the salt concentration limit of SUS304L to 0.8 g/m^2 as Cl. If we take account of the grinder work on the welded part of the actual canister, the salt concentration limit could be lowered to half of the limit for the materials whose surface was wet-polished.



(Test jig and specimens) (Example of the SCC initiation test results)

(SUS304, 5000hr, Hardness 247Hv)

FIG. 3. Results of the SCC initiation test.

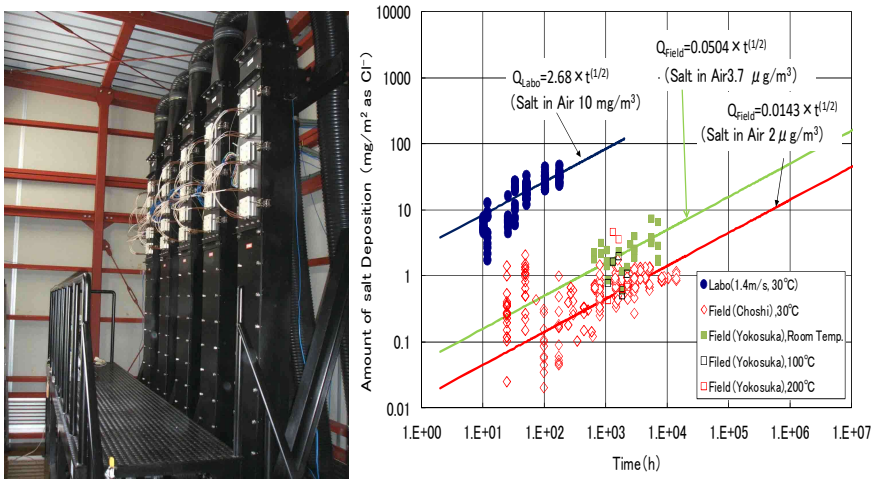
4. EVALUATION OF SALT DEPOSITION ON THE CANISTER SURFACE [3, 4]

We carried out tests to measure salt deposit on a metal surface placed in the air containing salt particles in order to find relationship between salt particle in the air and salt deposit. In the beginning, we have conducted laboratory tests that found influence of specimen temperature and flow rate on salt deposit by spraying salt water in a wind tunnel with a large salt concentration in the air. In order to obtain relationship between salt deposit data of laboratory test and those of field tests and accumulate salt deposit data under many weather conditions as much as possible, we have conducted field tests. For salt deposit tests, we placed stainless steel specimens (SUS304) after cleaning in vertical wind tunnels with certain operating conditions. The sizes of the wind tunnel for each test are different because it is difficult to make the big facilities for the field tests. The specimen temperatures were 30, 100 and 200°C. For the field tests, two locations were selected. One is Choshi area which is typical exposure test centre located at the distance of 4 km from the seashore (3 m above sea level). Another is Yokosuka area of the CRIEPI which is located at the distance of 40 m from the seashore (20 m above sea level). Fig. 4 shows results of salt deposition on the specimens in the salt deposit test device. The results show the amounts of salt deposition in the field tests measurements were less than about 10 mg/m^2 (as Cl). This value is significantly small as

compared with the threshold value (800 mg/m^2) for the SCC initiation of SUS 304L that is a candidate material for the canister. This means the air flows in parallel with the canister surface and the salt deposition less.

To evaluate the amount of salt deposition on the canister, it is important to know the environmental condition. There are several measurement items for the evaluation. Most important item is salt concentration in air, but it is difficult to measure it for the long time in the field. We may evaluate the amount of salt deposition on the canister using the alternative data. Fig. 5 shows the environment data measured in Choshi test facility for the reference.

In the future, measurement on the salt deposit should be obtained at a real site for evaluation.



(Wind tunnel placed in Choshi area) (Results of salt deposition tests)

FIG. 4. Results of salt deposition tests.

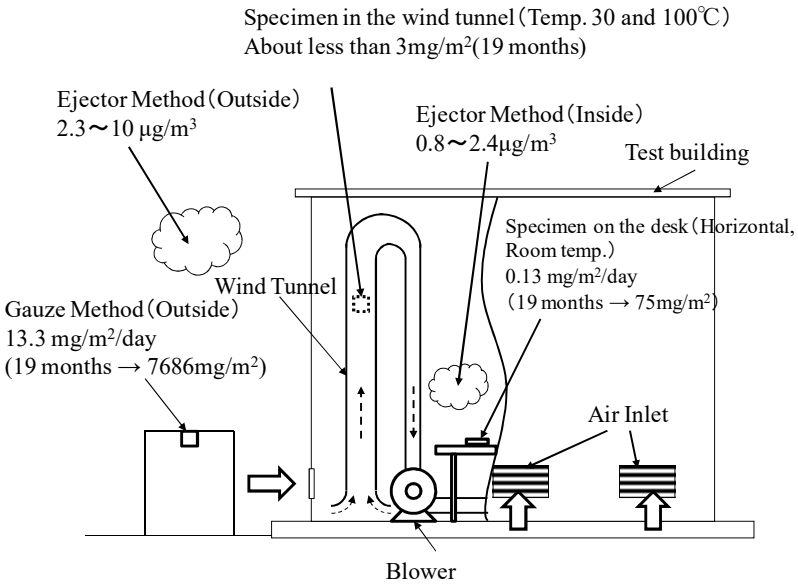
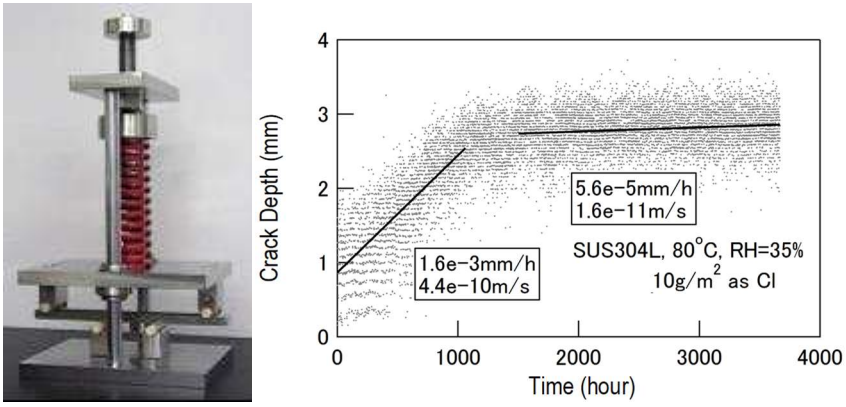


FIG. 5. Results of environmental measurements in Choshi test facility.

5. EVALUATION OF CRACK GROWTH RATE [1, 2]

To know the SCC crack growth rate, we performed crack growth tests using a four-point bend test jig shown in Fig. 6. We used SUS304, SUS304L and SUS316L materials. Bending test specimens with a gage section of 10 mm thick, 20 mm wide, and 220 mm long were machined from the matrix, subsequently subjected to a constant bending load. Salt was supplied intermittently by using a nozzle which could spray fine particles. We set the temperature of 80°C and the relative humidity to 35% in the test.

Fig. 6 shows an example of the test results. The test specimen was placed under conditions of constant temperature and humidity in chambers, namely at 80˚C and with RH=35%. The CGR was measured by the reverse direct current potential drop (RDCPD) method. The potential drop data was converted to crack depth data, assuming the propagation of a half elliptical crack. It emerged that the crack initially propagated fast, whereupon the propagation rate declined. Assuming a certain initial size due to the initial fast crack propagation, the remaining canister wall can be evaluated with slow propagation rate data. We tentatively adopted the CGR as 1×10^{-11} m/s. This CGR value yields 19 mm crack growth for 60 years but should be regarded as a considerably conservative evaluation.



(Test jig and specimens) (Example of the crack growth test results)

FIG. 6. Results of crack growth tests.

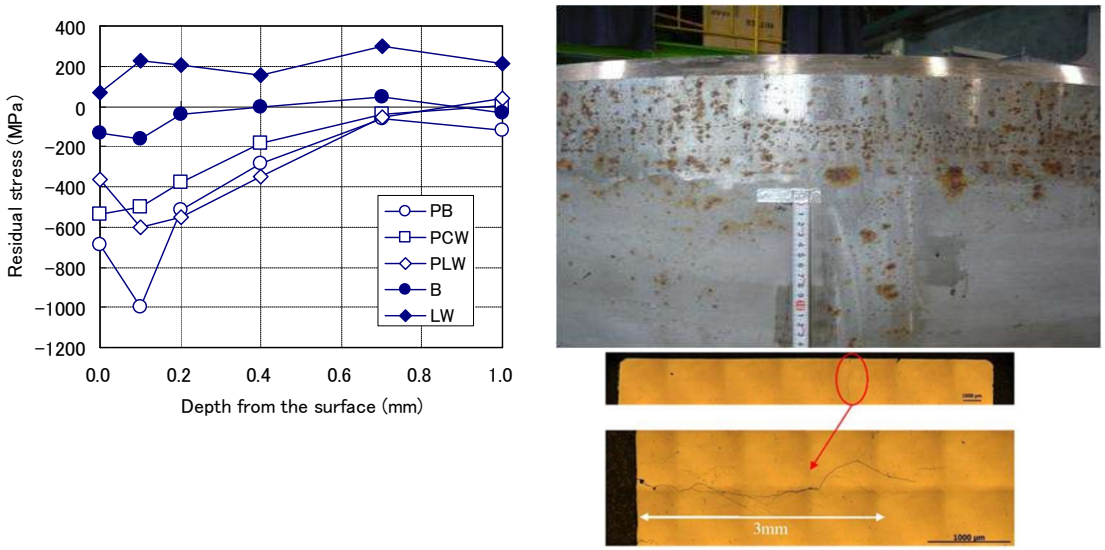
6. SCC TEST USING MOCK-UP MPC MODEL [1, 5]

To verify the effectiveness of the surface treatment techniques to the commercialized MPC, the mock-up bottom part of the MPC (weight 2.1 tons) with full-scale diameter (1836 mm) and wall thickness (12.7 mm) was manufactured made of SUS304L using the laser welding technique. Half of the weld lines were treated by zirconia shot peening (SP) to mitigate the residual tensile stress. Fig. 7 shows the stress distribution in the depth direction along the centre line of the weld metal. In the non-surface-treated weld line (solid marks), the uniform tensile stress (about 200 MPa) was appeared up to the 1 mm depth. On the other hand, in the surface-treated weld line (open marks), the compressed stress was appropriately induced. An accelerated corrosion test was executed with this mock-up MPC, in which the salt concentration on the surface of the weld lines was set to 4 g/m^2 as Cl, which is 5 times of threshold chloride density for SCC initiation of SUS304L as described in section 3. The test condition was set at constant temperatures of 80°C with relative humidity RH=35% over 2000 hours. After the test, the visual inspections and fluorescent penetration test (PT) were executed along all of the weld lines and the suspicious areas for the occurrence of SCC were

detected. As the indication of the existence of the SCC was not visible due to the rust, the suspicious areas were selected by PT indications and cut to small plates. Using these plates, surface crack was observed with SEM, and cross-section of the crack was observed with optical microscope as shown in Fig. 7. As a result, the chloride induced SCC initiation was observed on the as-weld surface and the measured crack depth was 3 mm. As there was no occurrence of the SCC on the surface-treated weld line, it seems that well-qualified surface treatment to mitigate the residual stress exerted on the weld lines should be necessary to prevent the chloride induced SCC initiation of the MPC which will be stored in the salty environment for long-term. Now, we are executing the SCC test with another mock-up MPC. The mock-up is the lid part with full-scale diameter using low plasticity burnishing (LPB) technique. We will check the effectiveness of the LPB technique.

7. MEASUREMENT TECHNOLOGY OF CHORINE DEPOSIT ON CANISTER SURFACE [6, 7]

Periodic measurement to verify that the concentration of chlorine in salt attached on a canister is necessary in the operation of storage to prevent the loss of the confinement function of spent fuels owing to the occurrence of SCC. The applicability evaluation for the measurement under high-temperature and high-radioactivity environments is needed for the measurement of salt attached on the canister that is stored in a concrete body. We are developing candidate method of measuring the concentration of chlorine in salt attached on a canister using Laser-induced breakdown spectroscopy (LIBS) technique. LIBS is a method of measuring the concentration of each element in the target by analysing the emission spectra of plasma produced by the laser irradiation on the target. The experimental setup for chlorine concentration measurement by LIBS is shown in Fig. 8.



(Measured residual radial stress distribution) (Mock-up of SCC test and microscope image)

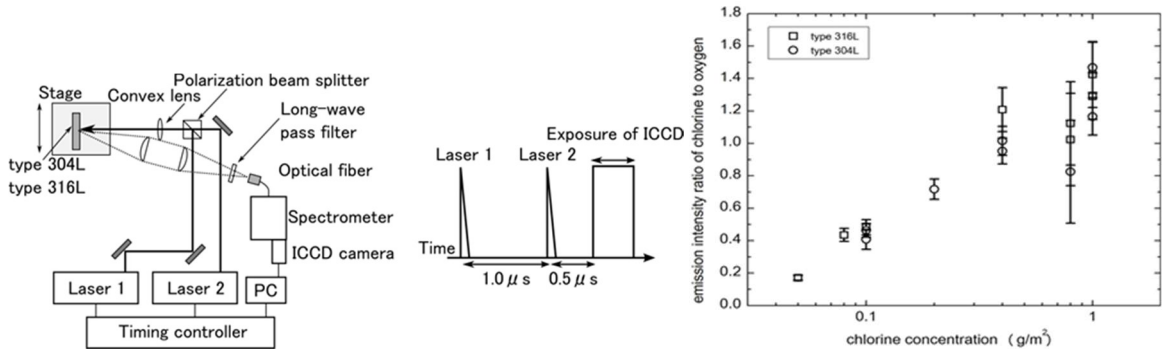
FIG. 7. Results of SCC test using mock-up MPC model.

The pulses of lasers 1 and 2 were superposed using a polarization beam splitter and by rotating the plane of polarization and were focused by a plano-convex lens. One of the laser

pulses was used for producing the plasma, and the other was used for re-exciting the laser-induced plasma. In comparison with the use of one laser pulse, which is called the single-pulse (SP) configuration, the use of two laser pulses, which is called the double-pulse (DP) configuration, is effective for enhancing the emission intensity of the laser-induced plasma, which contributes to the enlargement of the quantitative measurement range. The stainless-steel plate was moved perpendicularly to the laser irradiation by 1 mm per shot to irradiate each laser pulse on the fresh surface of the plate because the salt attached on the plate was removed by laser irradiation. Types 304L and 316L stainless-steel plates, which are candidate canister materials, were used for LIBS. Synthetic seawater was sprayed uniformly on the plate, and the chlorine concentration was controlled by adjusting the number of sprayings.

The chlorine spectra were measured for the samples with a chlorine concentration from 0.05 to 4.0 g/m^2 by using DP measurement (Fig. 8). The chlorine fluorescence intensity normalized by oxygen fluorescence intensity increased monotonously versus the chlorine concentration from 0.05 to 1.0 g/m^2 in the DP measurements. These results show the possibility of the quantitative measurement of the chlorine content on the canister surface.

Furthermore, remote measurement is an important issue in terms of inserting the LIBS devices into the narrow space such as the space between the canister and a concrete body. We developed a compact device unit comprised of optics for LIBS and a drive unit for vertical movement.



(Experimental setup and time relationship of laser pulse) (Emmission intensity ratio by LIBS)

FIG. 8.Experimental setup and emmission intensity ratio by LIBS.

8. DETECTING METHOD OF HELIUM LEAK FROM CANISTER [8]

> "*In the concrete cask, spent fuel is installed in a canister and the lid of the canister is sealed by weld. The canister is filled with helium gas and its containment should be maintained and inspected during storage.*
>
> *Helium gas enhances the function of heat removal from spent fuel. When helium gas leaks, the effect of helium convection is weakened in the canister. Then the temperature on the canister surface changes. By using this principle, the detecting method using information on temperature difference between top and bottom surface has been proposed*" [8].

Helium leak tests were performed using two types of the concrete casks. After the steady state of heat removal tests, the valve of the canister was opened. Fig. 9 shows the relation between the inner pressure in the canister and canister surface temperatures, including inlet air

temperature. The inner pressure decreased from 56 kPa to 5 kPa in 4 days. The temperature difference between the centre of canister bottom (TB) and the centre of canister top (TT) was defined as ΔTBT, and a detecting method of helium leak using ΔTBT was examined. Fig. 9 also shows the relationship between the changes of ΔTBT and that of the inner pressure in the canister. It was found that ΔTBT increased significantly due to the helium leak. Δ TBT increased monotonously to a constant value during helium leak, even if the inlet air temperature decreases.

In order to analytically evaluate the change of the temperature difference during the leakage of helium gas, we are developing the thermal hydraulic analysis method.

This figure has been removed from the proceedings as it has been reproduced elsewhere. Interested readings should refer to ref. [8] and may view the entire paper at the following link: http://www.sciencedirect.com/science/article/pii/S0029549307002506?via%3Dihub

FIG. 9. Temperature changes during helium leak form the canister.

9. CONCLUSION

In Japan, there is no dry storage facility using concrete cask. The CRIEPI is performing the studies on concrete cask to realize the concrete cask storage.

For the practical use of the concrete cask storage, remaining issues are preventing design (monitoring, inspection and countermeasures) and its demonstration of the chloride induced SCC for the MPC. In this study, we showed the following results:

- We proposed the SCC evaluation method including two scenarios. To maintain its confinement function of the MPC during storage period, it is necessary to keep the salt density on the MPC surface not to exceed its critical salt density to initiate SCC or controlling the crack propagation if the salt density exceed the critical value;
- We performed the SCC initiation tests, salt deposition tests and SCC crack growth tests. We showed the exampe data of several critical values. In the future, measurement on the salt deposit should be obtained at a real site for evaluation;
- We performed the SCC test with mockup of MPC using SP surface treatment. We verified its effectiveness;
- We are developing the inspection technologies such as chlorine deposit on the MPC surface by LIBS and helium leak from the MPC by surface temperature measurement.

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TECHNOLOGICAL DEVELOPMENT TO SUPPORT A CHANGE IN THE UNITED KINGDOM'S STRATEGY FOR MANAGEMENT OF SPENT AGR OXIDE FUEL

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Abstract

The United Kingdom's Nuclear Decommissioning Authority (NDA) has announced its preferred strategy for managing spent oxide fuel arising from the EdF Energy fleet of Advanced Gas-cooled Reactors (AGR). This is to reprocess the optimum quantity of fuel at the Thermal Oxide Reprocessing Plant (THORP), Sellafield and place the remaining fuel and all future arisings into interim wet storage in the THORP Receipt and Storage (TR&S) facility pending geological disposal. Storage periods up to 80 years are likely. Sellafield Ltd. is responsible for completing the technical basis for interim storage of AGR fuel and making a safety case to deliver this strategy. The imminent transition from buffer storage of fuel pending reprocessing to interim storage without the contingency of reprocessing of fuel that might fail (defined as penetration of the fuel cladding) during storage requires development of the wet storage technology to deliver a more thorough technical basis. Therefore, Sellafield Ltd. has developed a programme of technical work to increase confidence in the storage regime, demonstrate that corrosion models are applicable to cladding after higher irradiation and to changes in fuel design, characterize failed fuel and develop methods for fuel condition monitoring. The current technical underpinning for interim wet storage of AGR fuel is primarily empirical. It is based on the safe storage in sodium hydroxide dosed pond water of a large quantity of fuel, including some fuel stored for a period of 25 years. Post storage hot cell examination of a sample from this long stored population is currently ongoing as a demonstration of the storage regime. Furthermore, due to the continued evolution of the AGR fuel design and reactor operations, it is also necessary to ensure that the storage experience accrued is applicable to modern fuel discharges that will make up the majority of the storage inventory. Post irradiation examination of modern high burn-up fuel has also been performed to address changes in irradiated cladding properties resulting from changes in reactor operation. This paper summarises the existing technical basis to support the transition from reprocessing to interim storage of AGR fuel, the process of establishing a programme of work to develop the wet storage technology based on identified gaps, and the latest results available from this programme, including the ongoing post storage examination work.

1. INTRODUCTION

Spent fuel discharged from the EdF Energy fleet of seven AGR nuclear power plants (NPPs) is dispatched to Sellafield in Cumbria, UK for management. Sellafield Ltd. is licensed to operate the Sellafield site under contract to the NDA. As the site owner, the NDA is responsible for strategic decision making and has recently announced that its preferred strategy for managing spent AGR fuel is to reprocess the optimum quantity of fuel at the Thermal Oxide Reprocessing Plant (THORP) and place the remaining fuel and all future arisings into interim storage awaiting geological disposal [1]. It is currently assumed that a Geological Disposal Facility (GDF) for spent fuel and high level waste will be available to commence fuel receipts in 2075, meaning that storage periods of up to 80 years are likely (accounting for the time taken to emplace all stored spent fuel in the GDF) [2].

THORP reprocessing operations are programmed to cease in late 2018. After this date, THORP Receipt & Storage (TR&S) will be converted from operating as a reprocessing buffer store to an interim storage facility, principally for AGR fuel. The technical underpinning required to demonstrate that AGR fuel will remain in a safe state during interim storage over a period of decades is more demanding than that required for the much shorter storage periods as a precursor to reprocessing.

2. OVERVIEW OF AGR FUEL LIFECYCLE

An AGR fuel element (Fig. 1) consists of 36 stainless steel clad pins (each containing uranium oxide, UO_2 pellets) of approximately one metre in length. The fuel pin cladding is made from 20Cr/25Ni/Nb-stabilised stainless steel and is the primary containment barrier around the nuclear material. The pins are fixed in a grid at the base, with supporting braces at the centre and top of the element. The arrangement of fuel pins is held within a graphite sleeve, which contributes to the moderation during irradiation. Within the reactor eight elements are joined to form a stringer, which is equivalent to one reactor channel (with the exception of Dungeness B stringers, which only contain seven elements). The elements are numbered from one at the bottom to eight at the top.

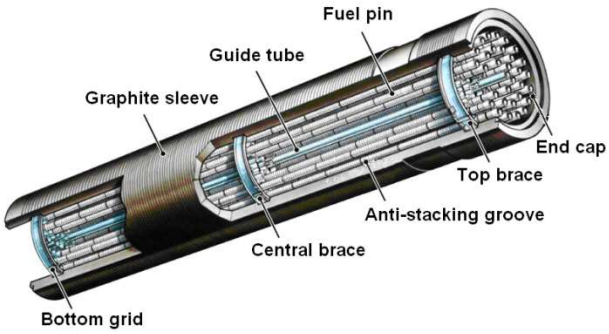

FIG. 1. Features of an AGR fuel element.

During service in the reactor, the materials in the fuel elements undergo a number of physical and chemical changes as result of exposure to irradiation, the carbon dioxide coolant gas and the thermal conditions. The most significant change that occurs to the fuel in the reactor is sensitisation to Intergranular Attack (IGA) corrosion. This affects material operating between 350 – 520ºC, which corresponds to the majority of the cladding and end caps of elements one and two, and the end caps of elements three to five. The principle mechanism that leads to through wall sensitisation is Radiation Induced Segregation (RIS), whereby high energy (fast) neutrons change the material composition at grain boundaries resulting in chromium depletion and nickel enrichment. When stored in demineralised water with chloride concentrations above the limit of detection, these elements have been found to fail by cladding penetration as a result of IGA.

Fuel clad ductility decreases during exposure to fast and thermal neutrons at elevated temperatures as a result of radiation-induced damage and helium embrittlement. Oxidation of the fuel cladding surface – principally forming chromium(III) oxide – in the coolant gas essentially is limited to the upper elements of the fuel stringer that operate at higher temperatures. The internal cladding surface may be affected by chemical attack by corrosive fission products and microbore cracking may occur on the upper elements as a result of thermal expansion of the fuel pellets. The fuel pin internal pressure results from the initial helium filling during fuel manufacturing and fission gases. The pressure at end of reactor life is a function of the fuel irradiation and operating temperature.

On discharge from the reactor, the fuel stringer is disassembled, and the individual elements are transferred to the NPP cooling pond. Here they are stored in open topped skips submerged in demineralised water dosed with 1250 ppm boric acid, neutralised to pH7 with sodium

hydroxide. After typical storage durations of 100 – 200 days, the fuel is transported in shielded transport flasks to Sellafield where it is received by the Fuel Handling Plant (FHP), shown in Fig. 2.

After a short period of further storage in FHP, AGR fuel elements are dismantled. This involves removing the fuel pins from the bottom grid and braces. The fuel pins from three elements are collected and subsequently stored in a stainless steel slotted can (Fig. 3). The remaining stainless steel components and the graphite sleeve are compacted and removed as separated waste material.

FIG. 2. AGR fuel storage ponds at Sellafield. From left to right: Fuel handling plant; AGR storage pond; and Thorp receipt & storage.

FIG. 3. Side view (left) and top view (right) of an AGR slotted can.

Slotted cans of AGR fuel pins are stored in lidded storage containers with a capacity for 20 slotted cans (Fig. 4, left) that allow exchange between the water in the container and the bulk pond water. Such containers are stored in both the FHP ponds and the AGR Storage Pond (Fig. 2), which operate with demineralised water dosed with sodium hydroxide to pH11.4 (colloquially termed caustic dosing). This pond water chemistry is seen to be effective at inhibiting IGA corrosion of the fuel cladding. Containers of AGR slotted cans are also buffer stored in the TR&S ponds (Fig. 2) awaiting reprocessing. Currently the TR&S ponds operate with demineralised water at pH7. They cannot be dosed with sodium hydroxide to pH11.4 whilst Multi Element Bottles (MEBs) incorporating aluminium components for storage of Light Water Reactor fuel (associated with international reprocessing contracts) are stored in the pond because of a materials compatibility issue.

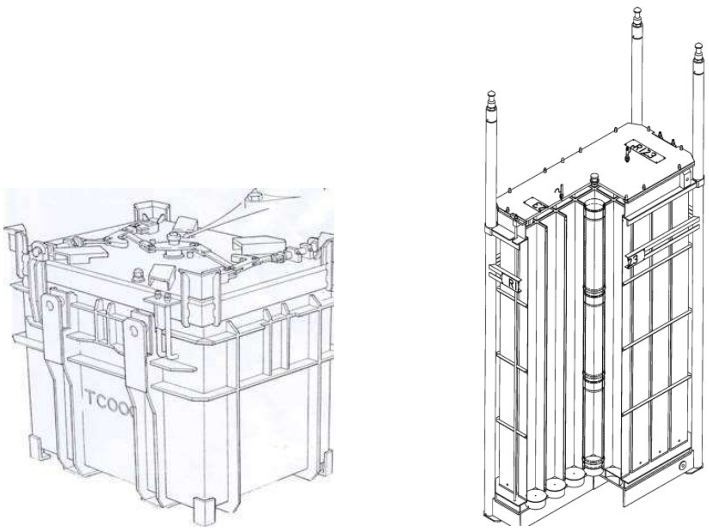

FIG. 4. AGR fuel storage containers: current '20e container' (left) and new '63e rack' (right).

3. OUTLINE OF THE PLANNED INTERIM STORAGE REGIME

Following the cessation of reprocessing of oxide fuel by Thorp, the remaining AGR fuel in storage at that time and all future AGR fuel arisings will be placed in interim storage. Interim storage is defined as covering the period from the cessation of Thorp operations until the completion of AGR fuel receipts from the AGR NPPs, which is expected by 2040. The storage period between 2040 and the consignment of fuel to the GDF is defined as long term storage and could be subject to a change in storage strategy, such as a transition to dry storage if sufficiently compelling reasons (in terms of technology, cost, regulatory requirements, and likely timescales for GDF availability) emerge in the future.

TR&S commenced active operations in 1988 and has been chosen as the most suitable interim storage facility. This is on the basis that the facility was built and continues to meet and exceed modern nuclear standards and is not required for other – potentially conflicting – Sellafield missions. With the introduction of new high density storage containers (Fig. 4), the facility has sufficient capacity, including the necessary flexibility to accommodate reasonable changes in the predicted inventory of AGR and other fuels.

Unlike the safety cases for many nuclear facilities that are based on a combination of technical arguments, operating rules and engineered safety mechanisms; the proposed basis of the safety case for interim storage is primarily based on technical understanding of fuel cladding corrosion behaviour. The principles of the safety case are that the fuel cladding is the primary containment and that the pond structure and other water retaining features are the secondary containment barrier. The introduction of the new '63e rack' high density storage container design, not only increases the facility's capacity but also allows for the container to provide substitute primary containment in the event of cladding failures and the associated release of soluble activity (predominantly caesium) to the pond water.

Therefore, to underpin the safety case, it must be proved that:

- General corrosion of the cladding of all fuel elements is at a sufficiently low rate that the integrity of the fuel is not compromised over the proposed storage duration;

- IGA in pH11.4 caustic dosed pond water of sensitised fuel cladding is either arrested or the rate of attack is reduced to a level that is acceptable for the proposed storage duration.

4. EXISTING TECHNICAL UNDERPINNING

4.1. General corrosion

General corrosion is a uniform attack over the metal surface, leading to loss of the passive oxide surface film and eventually material loss. In the absence of other failure mechanisms, general corrosion becomes the overriding degradation mechanism for the prolonged storage of AGR fuel. A number of studies have been performed using unirradiated cladding material and irradiated brace material to determine the rate of general corrosion in high pH solutions. These all found the rate to be very low or at the limit of detection, so that the cladding thickness losses from general corrosion are expected to be <0.2 µm/yr. This is equivalent to a total thickness loss of <16 µm over an 80 year wet storage period or <5% of the expected cladding thickness after irradiation.

4.2. Original corrosion inhibitor development work

A programme of work was undertaken in the early 1980s to identify a suitable corrosion inhibitor for AGR fuel storage, as post storage examination (PSE) of spent fuel discharged from the prototype Windscale AGR had shown that its cladding was susceptible to broad front pitting[1] and IGA corrosion. A range of inorganic salts were evaluated by corrosion tests including generation of electrochemical polarisation curves, long term tests at open circuit potential and long term tests under potentiostatic control and tests to determine radiation stability. The outcome was that 200 ppm sodium hydroxide (pH12.1) was chosen for further investigation on the basis of its efficiency, stability, compatibility with the Site Ion Exchange Plant (SIXEP) (used to treat the pond water purge) and the operational experience with Magnox fuel storage ponds.

Initially, whole irradiated fuel pins taken from elements one and three were tested by immersion in solutions containing 1–100 ppm chloride at 60–80ºC as a control, and 10 ppm chloride and 200 ppm hydroxide at 60ºC. The three control pins all failed after between 1900–8400 hours of exposure to these conditions, whereas the pins in solution including hydroxide did not exhibit any signs of attack after 8760 hours (1 year). However, it is noted that the control conditions were more aggressive than the inhibited solution.

Subsequent inhibitor testing work was performed with irradiated braces on the basis that these were fabricated at the time from 20Cr/25Ni/Ti-stabilised stainless steel, which is more susceptible to sensitisation than the niobium stabilised stainless steel cladding material, so incorporates some conservatism. Brace material is also easier to work with as it has not been in direct contact with the fuel pellets. The work investigated the minimum concentration of caustic required for inhibition based on the chloride concentration, temperature and brace irradiation. Analysis of the results from the different tests, including braces from fuel of burn-up up to 21.1 GWd/tU, predicted that for the FHP maximum specification chloride

[1] Broad front pitting has only been seen on cold-worked cladding irradiated in the Windscale AGR and not annealed cladding as used for the commercial AGRs, except for one incident in 1986 where annealed cladding was exposed to pond water with a very high chloride concentration and algaecides.

concentration of 0.5 ppm, a hydroxide concentration of 90 ppm (pH11.7) would be required for inhibition at the 95% upper confidence interval.

This work provided the basis for introducing caustic dosing of the AGR storage pond (AGRSP) in 1989. However, the pH of AGRSP was limited to pH 11.5 in order for the pond purge to be compatible with the SIXEP process. It should be noted that the pre-treatment used in order to induce corrosion (galvanostatically polarising the samples in 1000 ppm chloride solution for 30 hours) is very aggressive and induced effects not representative of pond storage conditions. Therefore, the conclusions drawn are believed to be overly conservative for pond stored fuel in sodium hydroxide.

4.3. Post storage examination

PSE work was carried out regularly between 1977 and 1993 on both Windscale AGR and commercial AGR fuel following periods of storage in Sellafield ponds. All fuel examined was subjected to: visual examination; leak testing; macroscopy; and metallography consisting of optical microscopy of transverse and longitudinal cladding sections. In some instances, this was supplemented by additional techniques required to address specific technical issues, such as the use of Strauss testing (similar to Ref. [3]) to determine the extent of cladding sensitisation, and mechanical testing of braces and fuel pins.

The main conclusions from the PSE work may be summarised:

- Annealed cladding, as used for commercial AGR fuel, demonstrated superior corrosion resistance when compared to early cold worked and TiN cladding variants;
- Where AGR fuel with irradiation in excess of 15 GWd/t, with standard cladding, had been stored in demineralised water with a chloride concentration >1 ppm in the AGRSP it was found that cladding failures of the lower elements (of a reactor stringer) had occurred by IGA (Fig. 5). Approximately 50% of the pins from elements one and two, and occasional element 3 pins, had cladding penetrations;
- No fuel that had been stored in caustic dosed pond water had cladding failures and, in general, exhibited much improved corrosion resistance. There was evidence of IGA corrosion to a depth of 40 µm on some samples (Fig. 5) and deeper intergranular cracking was observed on five out of 164 samples. The fuel examined had been stored for up to five years.

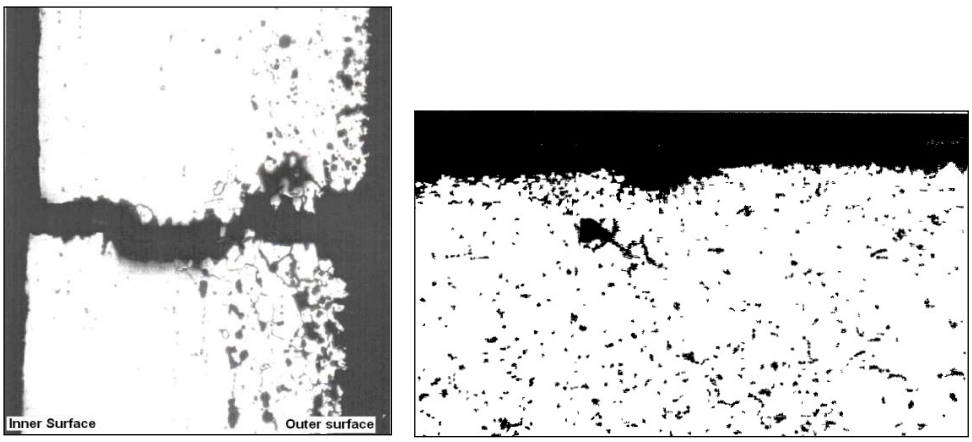

FIG. 5. Image of an IGA cladding failure (200x) [UKAEA] (left); some limited IGA on the external surface of cladding stored in sodium hydroxide (325x) [AEAT] (right).

4.4. Recent corrosion testing of irradiated brace material

Between 2008 and 2010, a series of corrosion trials were performed on irradiated AGR brace material. The purpose of this work was to demonstrate that pH11.4 sodium hydroxide was sufficient to inhibit propagating corrosion on fuel cladding that has received higher levels of irradiation, and hence is potentially more sensitised, than the material tested in the original corrosion inhibitor development work and early PSE studies. Zero resistance ammetry was used to monitor the galvanic current between the irradiated brace sample and a piece of unirradiated AGR cladding material, and the potential difference between the brace sample and 304L stainless steel pseudo reference electrode. This technique was sufficiently sensitive that the aggressive pre-treatment used in the original work (Section 4.2) was not required.

For brace material taken from fuel with an irradiation of approximately 38 GWd/t, it was shown that propagating corrosion could be arrested at pH11.0 and 2 ppm chloride at 50ºC. Testing to evaluate the impact of degradation in pond water quality was also performed. This showed that corrosion was not initiated over a 32 day period in a solution of pH9 and 2 ppm chloride. Assuming that brace material is bounding for cladding material's corrosion behaviour, this indicates that there is a safety margin within the current caustic dosing target of pH11.4.

Metallography and Strauss testing were used to confirm that the samples tested were heavily sensitised. This showed that the extent of chromium depletion had not increased when compared to less highly irradiated cladding analysed in the 1980s, which indicates that sensitisation may saturate with increasing irradiation.

4.5. Lead-time container studies

A lead-time container study is currently running in TR&S, which utilises fuel containers that are ullaged so that the container liquor composition may be controlled and not affected by the bulk pond water. Regular samples of the container liquor are taken to enable trending of fuel cladding corrosion products and indicators of cladding penetration, principally caesium radioisotopes, in addition to the pH and chloride concentration for control of the study conditions. The fuel chosen for the study includes the typical mix of elements seen in normal operations, of which approximately 25% are likely to be susceptible to IGA as a result of RIS.

The principle results, demonstrating the effectiveness of sodium hydroxide as a corrosion inhibitor for AGR fuel cladding are:

- A container of AGR fuel that has been stored in pH11.4 caustic dosed pond water conditions since 1989 has been monitored and confirmed to remain intact;
- AGR fuel exposed to pH9 (compared to the interim storage target of pH11.4) and 0.4 ppm chloride remains intact for at least 1000 days. This is a realistic upper chloride concentration in a fault condition arising from considerable damage to the pond hall roof (based on operational experience). In demineralised water (~ pH6) the fuel would be expected to fail within 450 days of the start of such a chloride excursion;
- The effect of more severe chloride excursions coincident with under-dosing to pH9, rather than the target for interim storage of pH11.4, has also been assessed for chloride concentrations in the range 1–2.5 ppm. No evidence of fuel failure is apparent for at least 1250 days at 1 ppm and at least 400 days for 2.5 ppm chloride. For the container at 2.5 ppm chloride, the concentration was increased to 4 ppm after 400 days and the

tests are currently ongoing. For comparison, in demineralised water at these chloride concentrations the fuel cladding would be expected to fail between 100–200 days.

4.6. Operating experience

Bulk quantities of AGR fuel have been stored in caustic dosed pond water in FHP and the AGRSP since 1986 and 1989, respectively. There has been no evidence of detectable cladding failures of fuel stored in these ponds. Fuel condition is monitored in the AGRSP by activity release modelling. This makes a comparison between the measured activity releases, primarily from a limited quantity of historic failed fuel, with the predicted activity release based on a number of parameters. If the two are in good agreement, this indicates there are no additional fuel failures. As additional reassurance, fuel stored in the AGRSP for approximately 15 years is routinely transferred to TR&S to be marshalled for reprocessing. The TR&S ponds normally operate with very low pond water activity, and release from tonne quantities of failed fuel would have been detected by the routine pond water sampling.

5. DEVELOPMENT AND STATUS OF THE CURRENT TECHNICAL WORK PROGRAMME

Sellafield Ltd has developed a conceptual basis of the safety case — essentially a safety case strategy — for interim storage of AGR fuel in TR&S in conjunction with dialogue with the regulators in the United Kingdom (the Office for Nuclear Regulation and the Environment Agency) and other stakeholders. This included preliminary assessment against the relevant safety and environmental principles, and how these may be satisfied by the relevant strategic, substantiation and technical aspects of the change from storage of AGR fuel prior to reprocessing to interim storage of AGR fuel. This process identified a number of gaps in the technical underpinning that need to be addressed. These gaps have been incorporated into a programme of technical work. The scope and status of this programme, including the latest available results, are described over the following sub-sections.

5.1. Increase confidence in the efficacy of sodium hydroxide dosed pond storage

While the existing technical underpinning (as summarised in Section 4) gives good confidence that AGR fuel can be stored successfully in pH11.4 sodium hydroxide dosed ponds, there are still some remaining uncertainties over the longer term efficacy of these storage conditions, such as whether IGA is absolutely inhibited or merely retarded to such a rate that has not become manifested over the current typical storage periods.

Therefore, PSE of a slotted can of nominally intact fuel, stored in pH11.4 sodium hydroxide dosed ponds since October 1989 is currently ongoing as a demonstration of the storage regime over a 25 year period. This work aims to examine the cladding in detail to determine whether there is any evidence of ongoing corrosion that has not yet resulted in cladding failure but could in the future. The slotted can chosen for examination includes the fuel pins from elements 1, 2 and 6 (the number representing the axial position in the fuel stringer), thereby enabling fuel from the lower temperature region of the reactor (expected to be affected by RIS) and fuel from the high temperature, high power region (not expected to be sensitised) to be studied.

The scope of examination work includes:

- Visual examination of all pins to assign the element position in reactor and general condition;

- Gamma scanning a selection of pins to confirm the element positions assigned visually;
- Macroscopy to give a low power surface characterisation looking for evidence of corrosion or other damage / degradation;
- Drop testing to give a simple confirmation of mechanical integrity with forces representative of fuel handling operations;
- Metallography of transverse and longitudinal sections of cladding using light optical microscopy to look for evidence of corrosion;
- Transmission electron microscopy of a limited number of samples to confirm fine microstructure and grain boundary chromium depletion profiles indicative of susceptibility to corrosion.

At the time of writing (April 2015), this work is ongoing so only early results are reported and it should be noted that these may not represent the final conclusions. The initial findings are:

(1) Visual examination (Fig. 6) enabled assignment of the fuel pins to the three elements present within the slotted can based on the appearance of the carbonaceous deposit, which was confirmed by the gamma scan results;
(2) No corrosion or other damage to the slotted can was noted visually during unloading of the fuel pins;
(3) Macroscopy of fuel pins from each element did not reveal any signs of corrosion damage (Fig. 6). Occasional minor abrasion marks and loss of the carbonaceous deposit over no more than 10% of the cladding surface were noted;
(4) Metallography of sections of fuel cladding taken from an element 1 pin in the regions most likely to be affected by corrosion do not show any degradation of the cladding condition beyond that typically observed during post irradiation examination (Fig. 6). Further examination after etching revealed a microstructure consistent with sensitisation, thereby confirming that susceptible material had been studied.

FIG.6. From left to right: General condition of element 1 pin; cladding surface condition; and metallography of longitudinal section showing heat transfer rib at an anti-stacking groove (fuel pellet is at bottom of image) [NNL].

5.2. Storage of fuel with higher irradiation

Since the original technical development work associated with wet storage of AGR fuel, the operation of the AGR NPPs has evolved, especially in terms of increasing fuel irradiation and increasing reactor dwell times. The effect of evolution of reactor operating practices, especially increased cladding irradiation, has been partly assessed by the recent inhibitor testing with irradiated brace samples (Section 4.4).

To support the conclusions from this work, and to assess the effect of increased time at temperature on the cladding microstructure, post irradiation examination of recently discharged fuel with higher irradiation has been carried out. The scope is to characterise samples of cladding from a selection of different element positions, by light optical microscopy, scanning and transmission electron microscopy to determine the overall microstructures, distribution and composition of precipitates, and quantify the elemental compositions at grain boundaries. At the time of writing, the results from this work were going through the governance route prior to being issued and it is hoped that the findings may be shared at the conference.

It is also intended to perform PSE on a slotted can of fuel with higher irradiation that has been stored in the target interim storage conditions for a period of 10–15 years to evaluate whether any degradation during storage is apparent and to provide a comparison with the 25 year stored fuel, but with lower irradiation, currently being examined (Section 5.1).

5.3. Characterise failed fuel

It is important to confirm that fuel subject to IGA cladding failures (either if the small population of historic failed fuel cannot be reprocessed or further failures occur during storage) does not have its structural integrity compromised so that the ability to handle fuel is not threatened. Current evidence that the structural integrity of IGA failed fuel is not compromised is based on:

- Experience of dismantling historic failed AGR fuel elements without an increase in the incidence of pin breakages;
- Successful experience of retrieving and transporting slotted cans of historic failed fuel pins from the AGRSP to THORP for reprocessing.

Preparations to commence PSE of 30 year stored IGA failed fuel are currently underway to examine how failed fuel evolves with time and determine whether there is evidence of: additional or larger cladding penetrations; fuel pellet degradation; corrosion from the inner cladding surface; or loss of structural integrity.

5.4. Fuel condition monitoring

It would be beneficial to have a means of monitoring the condition of AGR fuel cladding during interim storage. In particular, a method of detecting the initiation of IGA in sensitised fuel before clad penetration occurs is desirable so that corrective action / contingency measures may be implemented. It should also provide confirmation and reassurance that the pond water chemistry is achieving its objective of inhibiting IGA and is expected to contribute to ongoing understanding and confidence in the storage regime.

To provide a diverse monitoring regime, complementary techniques are being evaluated. Some techniques are already being used successfully or could be implemented easily. However, development work is required for corrosion sensing and remote non-destructive testing techniques. It is planned that a selection will be made from the candidate techniques that have been studied or are under development, which include:

- Electrochemical noise sensing to detect the initiation of IGA on a representative inactive sensitised cladding simulant;

- High definition camera, remote visual inspection of irradiated sensitised brace material;
- Ultrasonic measurements of cladding thickness in-pond;
- Eddy current testing of fuel pins;
- Potential drop / field signature method technique for detecting changes in electrical resistance of cladding or simulant subject to material loss as a result of corrosion.

6. CONCLUSIONS

Post closure of the THORP reprocessing facility, AGR fuel will be consigned to interim wet storage in the TR&S facility in pH11.4 sodium hydroxide dosed pond water at Sellafield for up to 80 years. There is substantial evidence that the planned storage regime will be effective at maintaining the fuel condition without cladding failures, based on operating experience and existing research work. Further work is ongoing to develop the wet storage technology based on identified gaps, and the latest results available from this programme further strengthen the technical basis for the declared strategy.

ACKNOWLEDGEMENTS

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The NDA has funded some areas of research on fuel condition monitoring.

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DEVELOPMENT OF SCC RESISTANT CANISTER FOR SPENT FUEL STORAGE AND TRANSPORT

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Abstract

During the long-term operation of austenitic stainless steel canister, the possibility of chloride induces stress corrosion cracking (SCC). One of the measures to prevent the SCC is to convert surface tensile stress to compressive stress. It is confirmed that both zirconia peening and ball burnishing methods can prevent SCC. Compressive stress layers where the zirconia peening or the ball burnishing were applied are about 800 μm and 1500 μm respectively. SCC tests were carried out with boiling aqueous magnesium chloride solution in order to evaluate the treated surface performance. SCC was observed on an as-welded specimen and a grinded specimen. However, treated surface by zirconia peening or burnishing did not show any SCC. In addition to SCC, pitting is also a big concern for long-term storage, so that the pitting test was conducted and confirmed that the predicted maximum pitting depth for 60 years storage period is shallower than that of compressive stress layers obtained by zirconia peening or burnishing. As these methods can be expected to prevent SCC for long storage period, a SCC resistant canister is proposed based on the results.

1. INTRODUCTION

During the long-term operation of austenitic stainless steel canister, the possibility of SCC begins to be pointed out [1, 2]. However, the concrete cask system (Fig. 1) has an advantage in terms of economy and is a major interim storage system in the United States.

In Japan, a concrete cask system is being to be recognized that it has the merits of small amount waste generation and economy comparing to a metal cask system, and studies of concrete cask which is suitable for Japanese conditions have been carried out by Central Research Institute of Electric Power Industry (CRIEPI) [3, 4]. In Japan, Interim Storage Facility (ISF) will likely be installed at coastal area because the transport of spent fuel is carried out by ship. A concrete cask system is using natural convection of the cooling air, so that salty air contacts canister surface directly. Salt particles in the cooling air adhere to the surface of the canister which may possible to induce the SCC and may lose its containment function as the worst case. We have to solve this issue for practical use of the concrete cask system in Japan.

2. SCC RESISTANT CANISTER

Three factors, stress, environment and material are necessary to induce SCC that is shown in Fig. 2. We have to note that if one of them can be eliminated, SCC does not occur. As mentioned later, austenite stainless steel Type 304L or Type 316L is planning to use for a canister shell material, we cannot eliminate the factor of material. In addition, it is difficult to eliminate the factor of environment because the ISF will likely be installed at coastal area and concreate cask system is using natural convection of the air for cooling. SCC resistant canister is one, which has compressive residual stress on the outer surface of a canister to eliminate the factor of stress.

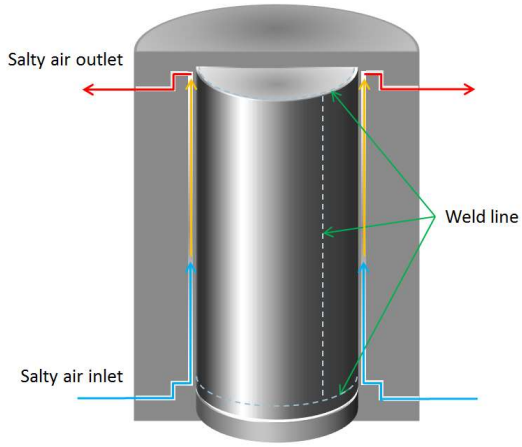


FIG. 1. Concrete cask storage module.

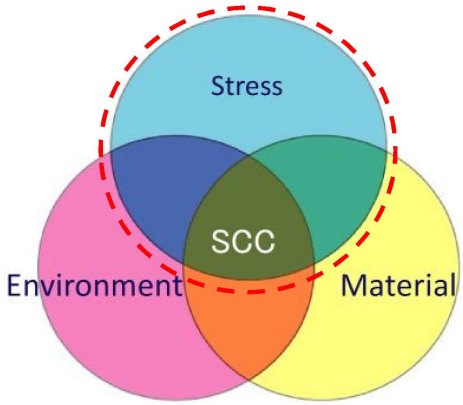


FIG. 2. Three factors to induce SCC.

2.1. Specifications for SCC resistant canister

2.1.1. Material and welding method

Type 304L and Type 316L stainless steel are selected for canister shell materials. Tensile residual stress in the weld is considered to be main cause of SCC, so that a laser welding method was chosen, because it generates smaller weld and consequently smaller residual stress area. The canister shell thickness is 12.7 mm ($^1/_2$ in.).

2.1.2. Surface treatment method

The tensile residual stress exists at and around the welded parts of the canister shell. As surface treatment methods to make the surface tensile residual stress compressive, a zirconia peening method and a ball burnishing method were selected. The zirconia peening is a method to blow zirconia ball shown in Fig. 3 on the canister shell by high-speed air. Since zirconia ball have big bulk density, high hardness and excellent toughness, it can give deep compressive residual stress layer on the canister shell. The ball burnishing (Fig. 4) is a method to make plastic deformation on the surface by pressing a hard rolling ball at the tip of the head. The process is well-known as the method for improving the fatigue strength and making material surface smooth. In order to evaluate their ability, two levels of process pressure (soft and hard mode) were used.

FIG. 3. Zirconia ball.

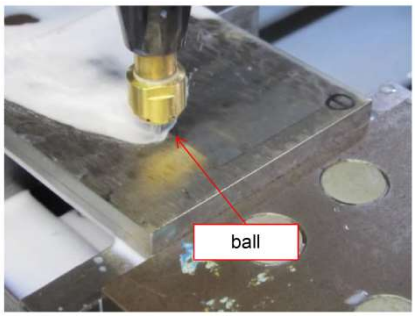


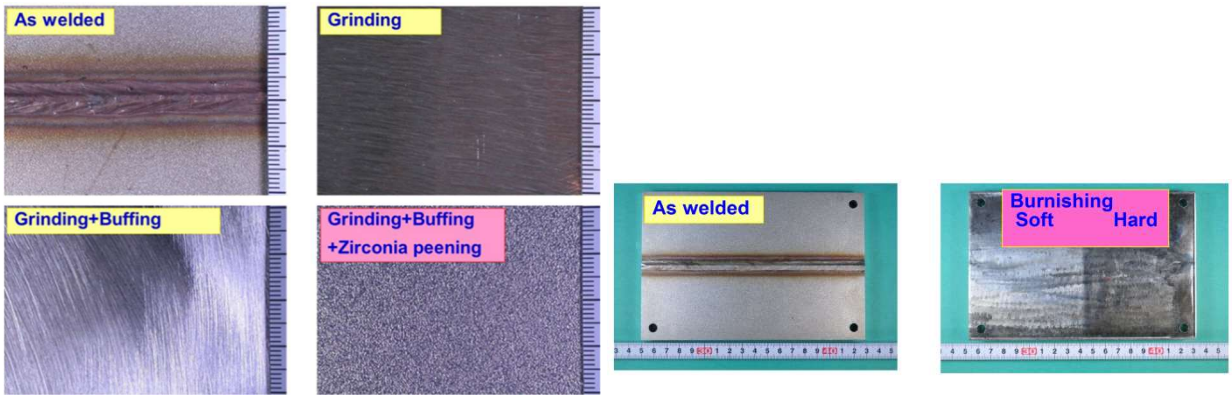
FIG. 4. Ball burnishing method.

3. EVALUATION OF COMPRESSIVE STRESS LAYER

3.1. Residual stress distribution

3.1.1. Test specimen

Surface appearances of specimens subjected to various fabricate process (As weld, Grinding, Grinding + Buffing, Grinding + Buffing + Zirconia peening or Burnishing (soft and hard mode) are shown in Fig. 5. All specimens have weld by laser welding at the centre of their body.



(Type 304L) *(Type 316L)*

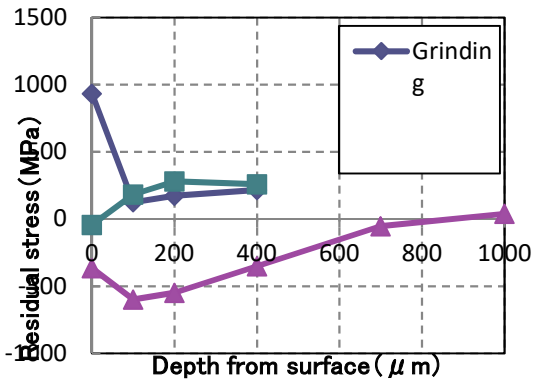
FIG. 5. Surface appearance of test specimens.

3.1.2. Stress distribution in Zirconia peening specimen

Fig. 6 shows residual stress distribution in the thickness direction measured by X-ray diffraction method. The residual stress is tensile to 400 µm depth in the grinder treatment specimen. The stress of surface to 100 µm depth area is decreased by additional buffing. Further zirconia peening changes the surface stress state to compressive. The compression depth reaches to 800 µm.

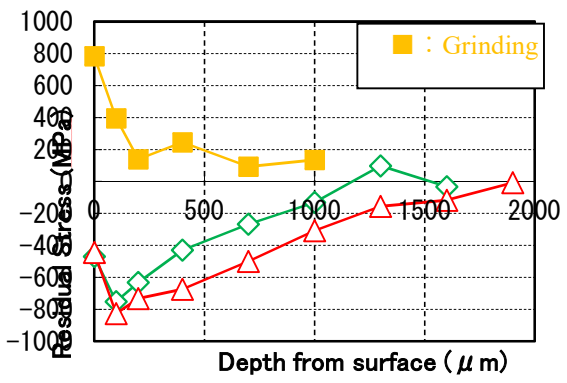
3.1.3. Stress distribution in Burnishing specimen

Residual stress distribution in the thickness direction measured by X-ray diffraction is shown in Fig. 7. Grinding process generates tensile residual stress whose depth is 1000 µm. The compressive stress depth is 1800 µm by hard burnishing and 1200 µm by soft burnishing, depended on the process pressures. Those depths of compressive stress by the ball burnishing method are deeper than that of zirconia peening.



(Type 304L)

FIG. 6. Residual stress distribution in the thickness direction (Zirconia peening).



(Type 316L)

FIG. 7. Residual stress distribution in the thickness direction (Ball burnishing).

3.2. SCC test

3.2.1. Zirconia peening

The Type 304L specimens were evaluated by SCC test method (JIS G 0576) using 42% boiling magnesium chloride aqueous solution. Fig. 8 shows the test results. Since the tensile residual stress was strong along the weld line direction, SCC was generated in the direction perpendicular to the weld line in the as-welded specimen. The cracks existed in 30mm width from the weld line. In the grinded and buffed specimens, SCC occurred on the whole surface area of each specimen. In zirconia peened specimen, SCC did not occur at all. From these results, it was confirmed that zirconia peening was very effective method for preventing SCC.

FIG. 8. Visualized cracks of SCC test specimens (Type 304L).

3.2.2. Burnishing

The Type 316L specimens were evaluated by SCC test (JIS G 0576). Fig. 9 shows the test results. Same as zirconia peening, strong tensile stress remained in the welding direction of the as-welded specimen, SCC occurred in the direction perpendicular to the weld line in 20 mm width from the weld line. In the burnished specimens, SCC could not be found regardless to the process pressure (soft and hard burnishing). From these result, it was confirmed that the burnishing method was also very effective to preventing SCC.

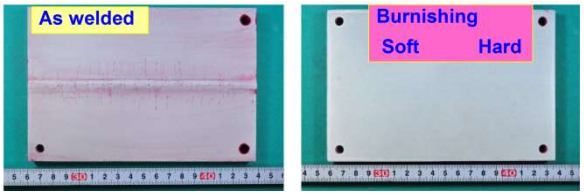


FIG. 9. Visualized cracks of SCC test specimens (Type 316L).

4. EVALUATION OF LONG-TERM SCC RESISTANCE WITH PITTING

Not only SCC but pitting should be considered during long-term storage. Since SCC may grow from the bottom of the pit where the residual stress is tensile as shown in Fig. 10, the pitting test was conducted using the specimens whose surface were treated by zirconia peening or ball burnishing.

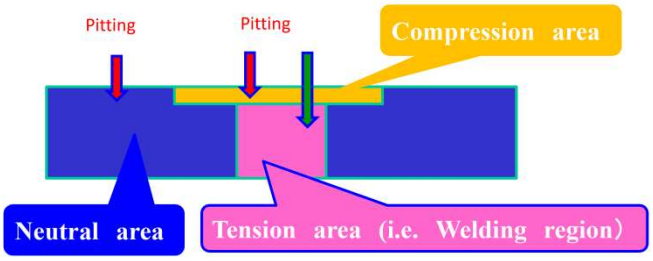


FIG. 10. Pitting test.

4.1. Prepareation of the specimen

Zirconia peened and burnished specimens of Type 316L stainless steel were prepared for the test.

4.2. Test conditions

The 5 μL of artificial seawater was dropped on the 100 × 100 size of specimens (Fig. 11), and it was dried in φ3 mm area. The salt condensation is about 14 g/m^2. The dropped areas are weld material, weld boundary, heat affected zone (5mm from the weld boundary) and base metal. 10 portions were tested at each position. Tests environment was 50ºC and 35% relative humidity. The pitting depth was measurement by microscope at 100 h, 300 h, 1000 h and 3000 h.

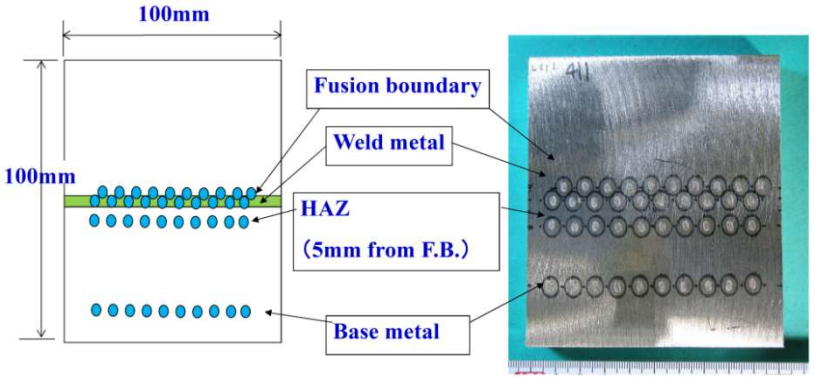


FIG.11. Pitting test specimen.

4.3. Maximum pitting depth

The relationship between the test period and the maximum pit depth are shown in Fig. 12. It is recognized that differences by the surface treatment and tested position were small and negligible.

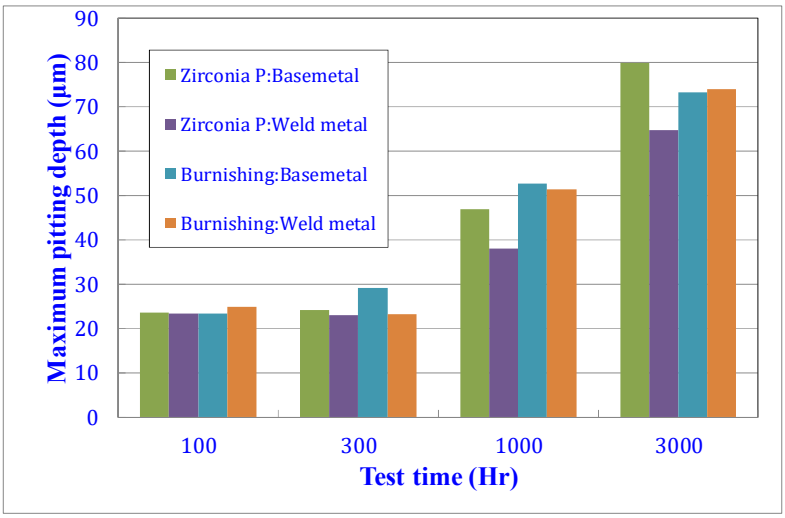


FIG. 12. Measured maximum pitting depth (Type 316L).

4.4. Estimation of pitting depth for 60 years

Pitting depth during 60 years of storage period is estimated under the following assumptions.

(1) The pitting growth is linear based on the growth rate calculated from the data of 3000 h in Fig. 12;
(2) Exposure time to corrosive environment is the period that relative humidity becomes more than 15% [4].

The estimated pitting depth for 60 years storage period is shown in the centre columns Table 1, and the compression stress depth obtained by surface treatments in the right columns. It is confirmed that pitting originated SCC does not occur during 60 years because the maximum pitting depth is less than that of compression depth obtained by the zirconia peening or ball burnishing.

TABLE 1. ESTIMATED MAXIMUM PITTING DEPTH (60YEARS)

	Pitting depth for 60 Years, (μm)	Depth of compressive stress area, (μm)
316L-Zirconia peening	401	800
316L-Burnishing	402	1500

5. CONCLUSIONS

The conclusions are as follows:

(1) Both the zirconia peening and the ball burnishing methods are effective for the SCC resistant canister;
(2) The Zirconia peening and the ball burnishing can make deep compressive residual stress layer;
(3) Pitting originated SCC is not occurring during 60 years, because the maximum pitting depth is less than that of compression stress layer.

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HYDRIDE EFFECT ON CLADDING BEHAVIOUR FOR SPENT FUEL STORAGE AND TRANSPORT CONDITIONS

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Abstract

Morphology and distribution of hydrogen picked-up during in-reactor operation could change due to the evolution of fuel conditions (pressure, temperature, etc.) from pool to dry storage. The possible dissolution and re-precipitation of the hydrides in the radial direction could provoke an embrittlement of the cladding. As the clad behaviour is a key factor during spent fuel dry storage and transport, Spanish organizations CSN, ENRESA and ENUSA have carried out research and development programs to characterize the performance of fuel under these conditions. The mechanical and fracture behaviour of fresh cladding electrochemically charged at different Hydrogen concentrations has been studied using ring compression tests (RCT). The tests have been performed at different strain rates and temperatures representatives of dry storage and transport conditions. Additionally a finite element model, based on the experimental RCT load vs. displacement curves has been developed to calculate the fracture energy as a function of the hydrogen concentration. The results show that samples with homogeneous and circumferential hydrogen distribution, do not present brittle behaviour. Rupture is produced at displacements higher than 3 mm. For radial hydride reoriented samples brittle fracture are obtained for low displacement values at 20°C and 135°C.

1. INTRODUCTION

The strategy followed in Spain for the spent fuel management set in the Radioactive Waste General Plan, which is currently in its 6th revision (approved in 2006 by the Spanish Government) is the open cycle. The spent fuel is on-site stored in spent fuel pool and in Interim Dry Storage when the pool capacity is over. Additionally, a Centralized Interim Spent Fuel Storage Facility has been approved and it is under design. The installation is based in a vault system, where the fuel assemblies are stored in capsules located in wells and cooled by natural convection. This installation is expected to be operative in 2018; therefore, spent fuel transports will be performed soon in Spain.

The safety functions (subcriticality, confinement, retrievability, heat removal and shielding) should be assured during storage and transport. Spanish Nuclear Authority (Consejo de Seguridad Nuclear-CSN) has published Safety Instructions regarding spent fuel storage and is developing specific regulations for spent fuel transportation, all of them consistent with the US-NRC regulations, 10 CFR 71 and 10 CFR 72. Cladding is the first barrier for the fissile material, and their properties and behaviour under dry storage and transport conditions (temperature and stress) are crucial to assure the requirements indicated in the standards:

> "*The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage*" [1].

"'Normal conditions of transport': The geometric form of the package contents would not be substantially altered" [2].

Under normal storage conditions, the major degradation mechanisms of Spent Nuclear Fuel (SNF) cladding that have been hypothesized to lead to failure include creep and hydrogen induced mechanisms [3]. During fuel irradiation hydrogen is absorbed in the cladding and, when hydrogen concentration is high enough, it precipitates as platelets of zirconium hydrides, that produces an embrittlement of the cladding [4]. The temperature and stress evolution during dry storage, may produce a hydride re-precipitation in the radial direction, as a consequence of the temperature reduction under stress. In those conditions, the clad could become brittle.

During transport the fuel rod integrity should be maintained considering the stresses transmitted to the fuel assembly and to the rods, as the bending loads supported by the rod material between grids, the pinch loads transmitted by the grid to the rod at the rod/grid contact point and vibration loads. Additionally, it should be taken into account that the transport can be performed at lower temperatures, with lower cladding ductility as temperature decreases.

Therefore, Spanish organizations, CSN, ENRESA and ENUSA have carried out several test programs to characterize the behaviour of spent fuel rod under dry storage and transport conditions. In the past, the programs were focused on creep behaviour of PWR and BWR fuel clad material [5, 6]. In this paper the results of a project focused on the effect of the hydrogen content and morphology in the cladding mechanical behaviour are presented.

2. OBJECTIVE

The aim of this project is to know the mechanical behaviour of the cladding material in storage and transport conditions. The effect of the hydrides, the mechanical and fracture properties of the cladding as a function of the hydrogen content, hydrogen morphology and testing temperature, and the parameters that could affect to these characteristics, could affect the material behaviour.

3. EXPERIMENTAL RESULTS

The material used for this work is un-irradiated PWR cladding (ZIRLO™ alloy [7]) in Stress Relief condition with 9.5 mm outer diameter and 0.57 mm wall thickness. The material had been pre-hydrided with different hydrogen content (representative of irradiated material and above) and different hydrides morphology (circumferential and radial hydrides). After that, it has been mechanically tested and analyzed. A FEM analytical model and a failure criterion have been developed.

Tests have been performed at high temperatures, representatives of dry storage conditions, and lower temperatures representatives of transport conditions. Additionally, room temperature has also been analyzed, to know the behaviour of the fuel after extended periods of storage and for theoretical transport after long time storage.

3.1. Cladding hydriding

Hydrogen was introduced via cathodic charge technique in KOH aqueous solutions. Samples with controlled amounts of hydrogen (150 to 2000 ppm) were obtained by this process. Further details about the charging method and sample preparation are given in [8].

After a thermal treatment, a homogeneous distribution with hydrides oriented along the hoop direction of the samples was obtained (Fig. 1).

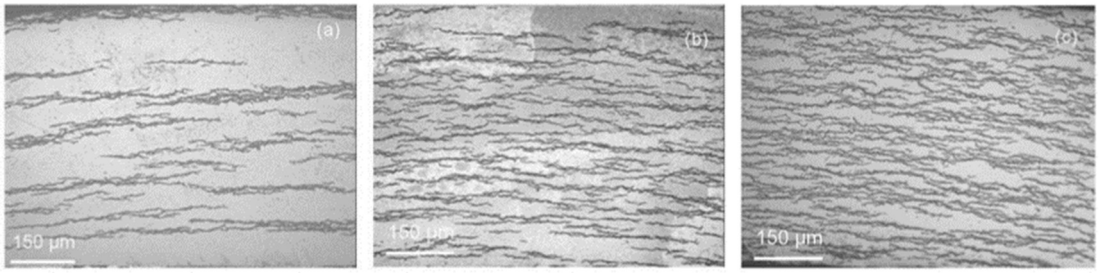

FIG. 1. Samples with 150 (a), 500 (b) and 1200 (c) ppm of hydrogen.

3.2. Hydrides reorientation

Samples with homogeneous hydride distribution and concentrations of 150, 500 and 1200 ppm of hydrogen were subjected to a thermomechanical treatment to produce the dissolution and reorientation of the hydrides [8]. The population of radial hydrides is a function of the hydrogen concentration and the hoop stress applied during cooling down. Fig. 2 shows samples with 500 ppm of hydrogen subjected to different reorientation hoop stresses: (a) 60 MPa, (b) 90 MPa, (c) 120 MPa and (d) 140 MPa.

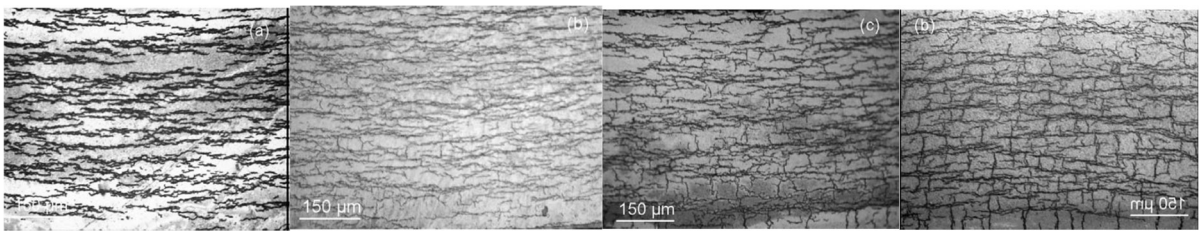

FIG. 2. Samples with 500 ppm of hydrogen subjected to hoop stress values of 60 MPa (a), 90 MPa (b), 120 MPa (c) and 140 MPa.

The hydride reorientation factor Fn (reoriented hydride percentage) has been calculated according to ASTM B811 [9]. For hoop stresses lower than 60 MPa, the reoriented percentage is lower than 3% for all the hydrogen concentrations. A relevant increase is produced for 90 MPa, and this tendency continues for higher hoop stresses. It seems that the threshold hoop stress for hydride reorientation should be around 90 MPa. This value is coherent with other authors' results [10] and with the limit imposed by the NRC to avoid radial hydrides in the Interim Staff Guidance ISG-11 [11].

Fig. 3 shows samples with (a) 150, (b) 500 and (c) 1200 ppm of hydrogen. All of them were subjected to a hoop stress of 90 MPa.

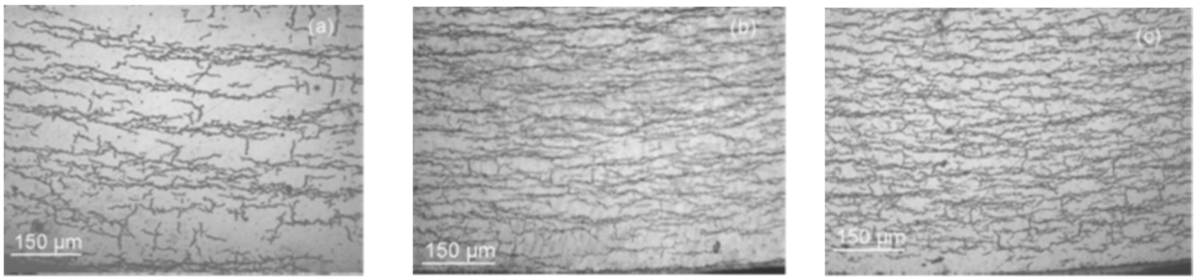

FIG. 3. Samples with (a) 150, (b) 500 and (c) 1200 ppm of hydrogen and a 90 MPa reorientation hoop stress.

The analysis of samples with different hydride concentration indicates that the fraction of reoriented hydrides is greater in the sample with 150 ppm of hydrogen, and the radial hydrides are longer than those of 500 and 1200 ppm samples (Figs. 3b, 3c).

3.3. Ring compression tests

The sample evolution and the load-displacement representative curve for RCT are represented in Fig. 4.

This figure has been removed from the proceedings as it has been reproduced elsewhere. Interested readings should refer to ref. [12] and may view the entire paper at the following link: http://docplayer.net/1737340-Journal-of-nuclear-materials.html

FIG. 4. Representative RCT evolution.

The RCT performed could be divided into two groups, representative of different irradiated material condition: samples with circumferential hydrides and samples with radial hydrides.
Samples with circumferential hydrides tests were carried out at different hydrogen concentrations (0 to 2000 ppm); different temperatures (20, 135 and 300°C) and displacement rates of 0.5 and 100 mm/min.

The results obtained for all the temperatures indicate displacements higher than 2 mm previous to the point of crack initiation. Besides, while for higher hydrogen concentrations, the displacement where cracks are initiated is lower, for some other material properties, as Yield Strength and Young Modulus, do not show a straight dependence of the hydrogen content (Fig. 5).

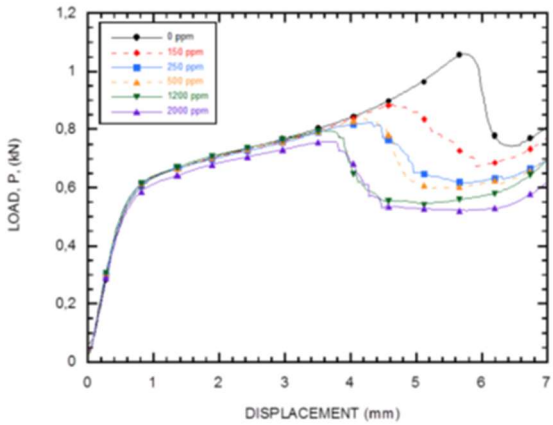

FIG. 5. Load-Displacement curve for different hydrogen concentration, 20ºC and 0.5 mm/min.

For these samples with circumferential hydrides, a 2D finite element model was developed with ABAQUS v 6.7–5 commercial code to simulate RCT results.

> "*The results of the numerical modelling versus the experimental results (one example in Fig. 6) show that the whole load vs. displacement curve is correctly predicted, including the downloading after the maximum load*" [11].

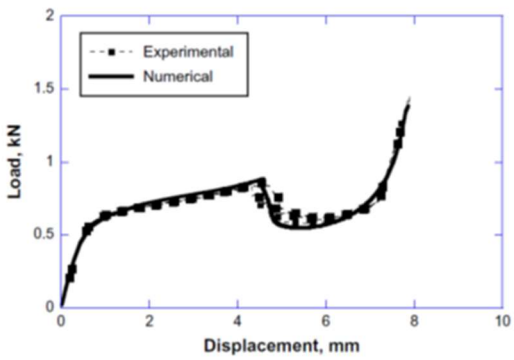

FIG. 6. Experimental vs numerical Load-Displacement curve.

With this model, the mechanical properties and the fracture energy are calculated based on the experimental load vs. displacement curves.

Additionally, to simulate the impact loadings corresponding to cask drop transport accident conditions, some tests have been performed at very high rate of displacement, 3 m/min. For all the temperatures, the obtained load-displacement curves are similar to those obtained at lower displacement rates with higher Yield Strength and Young Modulus, however the rupture of the material is produced at lower displacement values (3 mm vs 4 mm).

Samples with radial hydrides tests were performed for 150, 500 and 1200 ppm of hydrogen that have been reoriented with hoop stresses of 60, 90, 120 and 140 MPa. The test temperatures were 20, 135 and 300ºC, and the displacement rate has been 0.5 mm/min.

At the highest temperature, 300ºC, there is no remarkable effect of radial hydrides on material behaviour (Fig. 7).

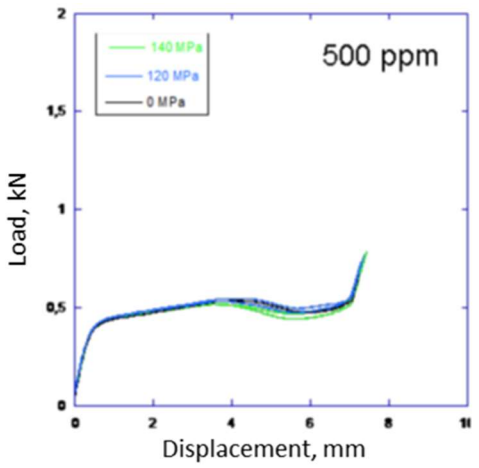

FIG. 7. Load-displacement curve for different reorientation hoop stress (300ºC-500 ppm H).

At 135ºC, it seems that there is no effect of the radial hydrides for samples with higher hydrogen concentrations (500, 1200 ppm). However, for samples with 150 ppm hydrogen, there is an important effect of the radial hydrides, as the rupture of the sample is produced at very low displacement values when the hoop stress used for the hydride reorientation is 140 MPa (Fig. 8). This result is related with the reoriented hydride characteristics indicated before, higher Fn and higher hydride lengths, and is coherent with data published by other authors [13].

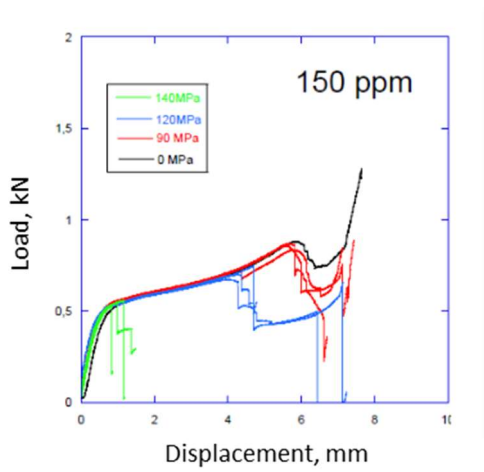

FIG. 8. Load (kN)-displacement (mm) curve for different reorientation hoop stress (135ºC-150 ppm H).

For the test performed at lower temperature, 20ºC, there is an important effect of the radial hydrides. For reorientation hoop stresses equal or higher than 90 MPa, the sample rupture is produced at displacement values lower than for the sample with no radial hydrides. For higher stresses during reorientation, the rupture is produced in the elastic part of the test (Fig. 9).

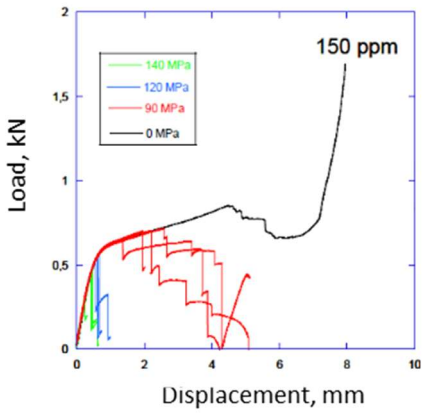

FIG. 9. Load (kN) -displacement (mm) curve for different reorientation hoop stress (20ºC-150 ppm H).

3.4. Fractographic analysis

A detailed fractographic analysis to identify the fracture micro-mechanisms has been performed.

The fracture surfaces were observed by Scanning Electron Microscopy (SEM).

The results are coherent with the RCT data. For 1200 and 2000 ppm samples, there are different behaviours depending on temperature. For 20 and 135ºC fracture mechanism is ductile to some extent, with crystallographic planes and local orientations which suggest a brittle fracture in some parts of the sample. For 300ºC a ductile behaviour at the microscopic level is observed everywhere.

3.5. Failure criteria

To assess the experimental results, three different failure criteria has been considered: (1) the offset strain criteria proposed by Billone [13] $\delta p/D0 < 2\%$; (2) SED at maximum load <25 N/mm^2; and (3) plastic deformation $\varepsilon p < 3\%$. If any of these conditions is fulfilled, the rupture has been considered brittle.

Applying these criteria for the radial hydrides samples, the following could be concluded:
The behaviour of the samples tested at 135 and 300ºC is ductile, even for the maximum hoop stress used for the reorientation. Only for one case, 150 ppm hydrogen and 140 MPa reorientation hoop stress, it could be considered in the ductile-brittle transition region.

The samples tested at 20ºC, are ductile only for reorientation hoop stresses lower than 90 MPa, for any Hydrogen concentration.

4. CONCLUSIONS

The Ring Compression Test has been used to obtain the fracture properties of hydrided non-irradiated ZIRLO™ cladding.

A finite element model has been developed to simulate the ring compression test. A very good agreement between the simulations and the experimental results has been obtained.

For homogeneous and circumferential hydride distribution, the behaviour of the material is always ductile; even for the highest hydrogen concentration (2000 ppm) and the lowest temperature (20ºC).

For samples with radial reoriented hydrides, the hydrogen concentration, the reoriented hydride percentage (related with the hoop stress), and the test temperature are key parameters that affect the material behaviour. At room temperature and 150 ppm of hydrogen, the rupture is produced at low displacements for every reorientation stresses, and there is a better behaviour for 500 and 1200 ppm hydrogen.

5. DISCUSSION

The different R&D projects performed during years for the Spanish organizations, CSN, ENRESA & ENUSA, provide interesting data regarding the cladding material behaviour during dry storage and transport. Creep tests performed in the past with irradiated ZIRLO samples result in very good creep behaviour. In the only test that led to failure among the test matrix performed, 17% strain was needed (at constant inner pressure) for cladding rupture.
The results of the project presented here are also very important related to the spent fuel dry storage and transport. It should be marked that for circumferential hydrides, the cladding material behaviour is always ductile, even at room temperatures. Therefore, any condition to avoid the radial precipitation of the hydrides during storage will assure a good behaviour of the cladding. For material with radial hydrides, the temperature and the hydrogen content are crucial; at low temperature the material behaviour is brittle. Additionally, for lower hydrogen content, the percentage of radial hydrides is higher, and the material is more brittle, this could be related with the radial-hydride-continuity factor (RHCF) [13].

All of these results are coherent with other authors published data for non-irradiated material.

ACKNOWLEDGEMENTS

A special acknowledgement to the Material Science Department, Polytechnic University of Madrid for a very good project development.

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INVESTIGATIONS OF STRESS CORROSION CRACKING OF SPENT FUEL DRY STORAGE CANISTERS USED FOR LONG-TERM STORAGE

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Abstract

The United States (US) Department of Energy (DOE) and the US Nuclear Regulatory Commission are working to develop a better understanding about the environment to which the inventory of dry storage canisters is exposed and how that might contribute to the corrosion performance of the canisters over time. The canisters are mostly made of 304 stainless steel that is susceptible to stress corrosion cracking (SCC), and canister penetration by SCC due to deliquescence of salts in dust on the canister surfaces is considered to be the most likely mode of canister failure. The DOE and the Electric Power Research Institute have teamed to sample and characterize dust deposited on the surface of in-service spent nuclear fuel storage canisters at three sites. While all three sites were near salt water or brackish water, the deposits collected from the tops and sides of the canisters show considerable variability based on the geographic location of the storage site. Samples from the sites more distant from the open ocean contained chloride-poor salt assemblages typical of continental dusts, while samples from the near-ocean sites were enriched in chloride-rich marine salts. Future work will evaluate how the composition and abundance of salt species affects corrosion of 304SS over time. Other tests will be performed to characterize the weld and manufacturing residual stresses in the canister shell; high through-wall tensile stresses are an important criterion for canister penetration by SCC. This work will improve understanding of canister performance through time as a function of the geographic location and the chemical environment on the canister surface.

1. INTRODUCTION

In the United States (US), spent nuclear fuel (SNF) is likely to remain in interim dry storage until a permanent disposal solution has been developed and placed into operation. Current US policy dictates that a permanent disposal solution should be available in 2048 [1]. Until a permanent repository is opened, SNF will accumulate and remain in dry storage for longer than originally planned. There are currently about 70,000 metric tonnes of commercial used fuel in storage in the United States, of which 80% is in used fuel pools and about 20% is in dry cask storage systems. This amount increases by about 2000 metric tonnes annually [2]. At the end of 2013, the US had SNF stored in 1850 dry storage casks, located at over 60 different sites across the US [3]. Commercial spent nuclear fuel is stored throughout the US at both coastal and inland locations (Fig. 1).

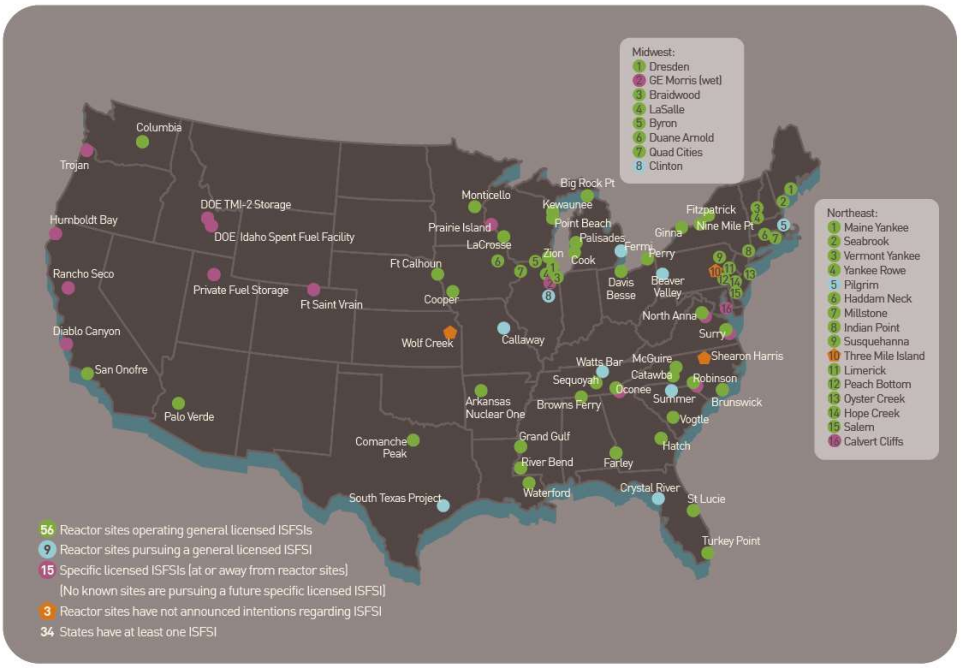

FIG.1. US Independent Spent Fuel Storage Installations.
http://www.nrc.gov/waste/spent-fuel-storage.html. downloaded 3/18/2015

In the United States, the predominant storage system consists of a welded austenitic stainless steel (typically 304) container. This container represents the primary confinement boundary for the spent fuel. The containers are stored either vertically or horizontally within an overpack, the role of which is to provide neutron shielding, physical protection, and passive cooling for the spent fuel container. As seen in Fig. 2 below, the overpacks are concrete or steel, and have space between the stainless steel canister and the overpack to allow for passive cooling by convective air flow. Air enters the overpack via inlet vents near the bottom and exits through outlet vents near the top. These vents allow for continuous convective cooling using ambient air to remove heat from the system. This fail-safe, zero energy, convective cooling has the disadvantage of allowing dust, salts, and other air contaminants to contact and interact with the stainless steel canister.

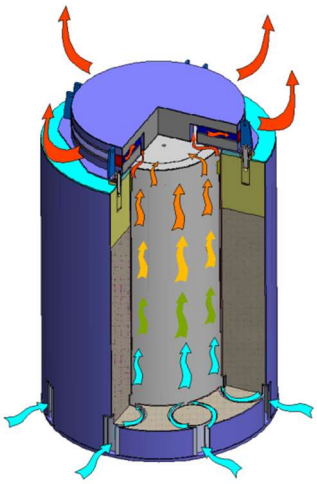

FIG. 2. Diagram of typical vertical (right) dry storage canister systems. Printed with permission from Holtec International.

Prolonged exposure to the unfiltered air stream described above will result in the deposition of atmospheric dust on the container surface. A portion of this dust will likely be chloride rich salts, potentially creating an environment conducive to stress corrosion cracking (SCC) of austenitic stainless steels. As a result, understanding the SCC related to interim storage containers has been identified as a high priority data gap by DOE and other government and private entities [4–7]. The US Department of Energy (DOE) and the US Nuclear Regulatory Commission (NRC) are evaluating the risk for SCC in the US dry storage canister inventory. This paper summarizes work done to characterize the canister material properties that affect susceptibility to SCC (e.g., weld residual stresses), and to understand the diversity and composition of the dust deposited on the canisters at different geographic locations. This paper also summarizes future testing funded by the DOE in this area.

2. STRESS CORROSION CRACKING

The US DOE and NRC are evaluating potential failure mechanisms for stainless steel canisters during long term dry storage. Stress corrosion cracking is a plausible degradation mechanism for the container with the potential to result in a penetration, which typically initiates in areas of localized corrosion (e.g., pitting). Verification of the presence of SCC requires that a detailed inspection of the container surface be performed. To date, the technology for such inspections has not been demonstrated, and no detailed inspections have been conducted. There are several factors that make implementing such inspections difficult. First, in-situ inspections of the canister surface within its overpack are difficult due to limited entry points and narrow annuli between the canister and inner wall of the overpack. Second, high radiation levels exist at the surface of the container, hindering the extraction of the canister from the overpack and implementation of traditional SCC inspection techniques. These factors, coupled with the inherent difficulty in detecting small stress corrosion cracks, have prevented the detailed surveillance of fielded spent fuel canisters for the presence of SCC. Because of these limitations and the lack of data, the US DOE Used Fuel Disposition (UFD) Program is currently funding work to understand the mechanisms for SCC and the anticipated environmental conditions on canisters in the current inventory.

Over time, the passive cooling system described above will result in the deposition of atmospheric dusts on the surface of the canister. A portion of these dusts will be salts which will eventually deliquesce, to form an aqueous brine on the canister surface. As austenitic stainless steels are particularly susceptible to SCC in chloride-bearing brines, there is a risk of cracking in situations where this brine layer is chloride-rich. Preliminary data indicate a difference in the deposited salt compositions and concentrations in “marine” and “near-marine” environments. The dust deposited in marine environments will likely contain sea-salt aerosols that are high in chloride.

In order for stress corrosion cracking to occur, three conditions must be met, as shown in Fig. 3. There must be:

(1) A susceptible material: Austenitic stainless steels such as 304SS, are highly susceptible to SCC in chloride-bearing aqueous environments. This susceptibility can be exacerbated due to microstructural changes that accompany high heat input processes, such as welding. Results of some of these studies will be discussed in this paper;
(2) An aggressive (chloride-rich) environment: Samples of the dust and salts from the surface of in-service canisters at marine and near-marine locations have been

collected. In addition, canister surface temperatures have been collected. The results of these analyses will be discussed in this paper;

(3) Sufficient tensile stress: A full-diameter scale cylindrical mock-up has been manufactured and residual stresses near the welds and heat affected zones will be measured. The plan for this testing will be summarized in this paper.

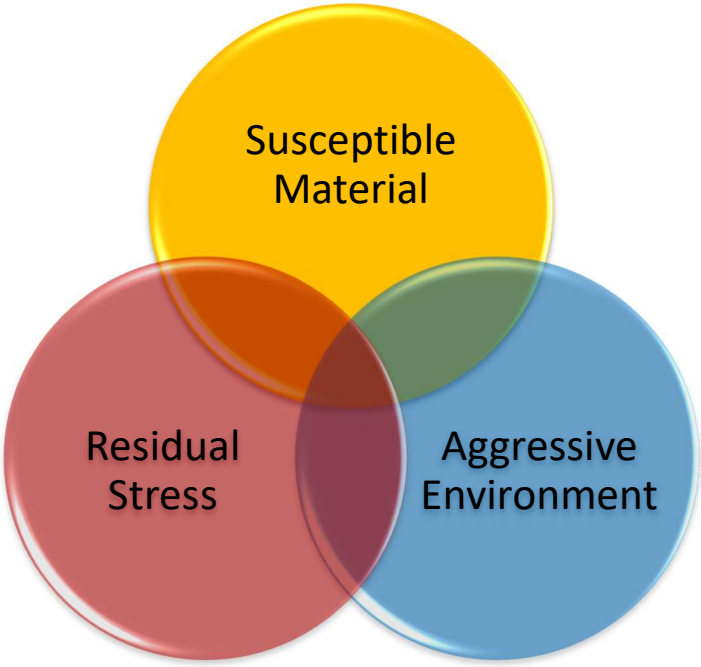

FIG.3. For stress corrosion cracking to occur, you must have the right material, environment, and stress.

2.1. Susceptible material

As noted previously, there is ample evidence, both from industry and from experimental studies, that 304SS is susceptible to SCC under atmospheric conditions if chloride-containing salts are present [8].

2.2. Aggressive environment

Since it has been confirmed that 304SS is susceptible to localized corrosion and SCC, the next step was to investigate if currently in-service canisters could be exposed to a corrosive environment. Sandia National Laboratories and the Electric Power Research Institute (EPRI) have partnered to sample and analyze dust deposited on in-service storage canisters to increase the data set and knowledge of the environments on the dry storage canisters.

2.2.1. Storage container environmental conditions

Because the canisters are shielded from the weather by the overpack, the only mechanism for corrosive conditions to persist on the canister surface is by deliquescence of salts that settle on the metal canister from air convecting through the overpack. Deliquescence can occur when a critical combination of temperature and absolute humidity are achieved at the container surface, the magnitude of which is a function of the salts that are present. The relative humidity (RH) is controlled by the local temperature on the canister surface and the absolute

humidity in the air entering the overpack, and the threshold RH value for deliquescence is a function of the salt composition present. Because of the importance of canister surface temperature, the US DOE has been funding work to better understand the surface temperatures of dry storage canisters with different fuel heat loadings and how those temperatures change with age, outside temperature fluctuations, and at different locations on the canister [9]. The temperature on the canister surface will vary with position and is a function of the thermal output of the fuel within the canister, the temperature of the air stream being pulled into the overpack to cool the canister, and the location of the surface within the overpack itself. This is demonstrated in Fig. 4 which presents a model of the surface temperatures on a horizontal storage canister located at the Calvert Cliffs Independent Spent Fuel Storage Installation (ISFSI) [9].

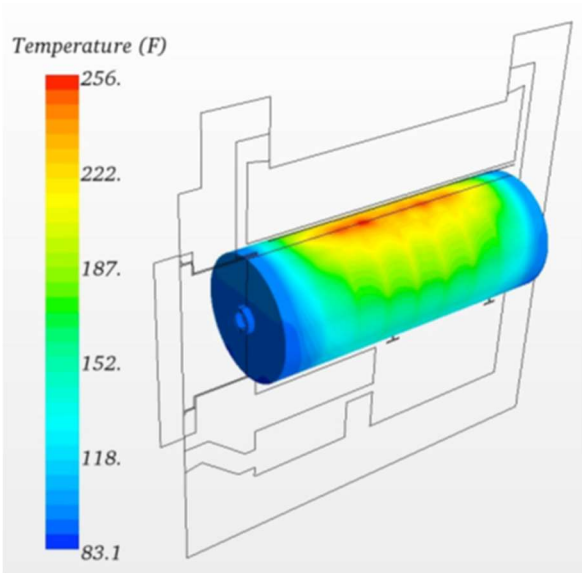

FIG. 4. Example of canister temperature model results for a vertical canister stored in a concrete housing [9].

2.2.2. In-service samples

From 2012 to 2014, dust samples were taken from the tops and sides of canisters at three different ISFSI sites. The first was Calvert Cliffs, which is located in the state of Maryland on the Chesapeake Bay, a brackish water site with little wave action. This facility utilizes horizontal storage systems. The second site sampled was Hope Creek, which is located in the state of New Jersey, 0.25 miles from the tidal Delaware River. This is also a brackish water body with little wave action. The final site that has been sampled was Diablo Canyon, which is located 0.3 miles from the California coast near a rocky coastline with constantly breaking ocean waves. Both Hope Creek and Diablo Canyon use vertical dry storage systems. The design and construction of the sampling systems and the acquisition of dust samples were performed by a team of personnel from each reactor site and from the vendor that supplied each storage system.

Dust samples were taken from two canisters at each of the three sites. At each location, two types of samples were collected. Dry samples of the dust were collected by using an abrasive Scotch-Brite™ pad; in the case of Calvert Cliffs, the pad was backed by filter paper and connected to a vacuum line (see Fig. 5 for an illustration); while at the other two sites, the pad was simply backed with a steel coupon, and the dust collected via static attraction to the

Scotch-Brite™ pad. Neither of the dry sampling techniques was characterized for their extraction efficiency.

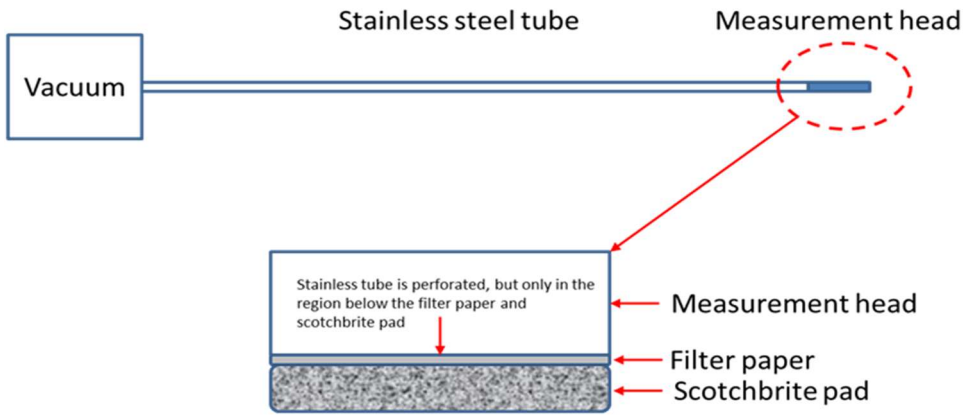

FIG. 5. Schematic of the dry sampling tool used to collect dust on canister surfaces (Diagram credit: D. Enos).

The second sampling method was a wet method, which used a SaltSmart™ sampling device to collect water-soluble surface contaminants. The SaltSmart™ device quantitatively collects salts from a known surface area, allowing quantification of the mass loading of the soluble salts on the canister surface to be measured. Following collection, the SaltSmart™ sensor was disassembled and the collected salts leached out for chemical analysis. Fig. 6 shows a disassembled Scotch-Brite™ sampling device. At the Calvert Cliffs site, the door was removed and replaced with a shield plug that allowed access to the sides of the container. Sampling was done by hand, by inserting long probes through the open door of the overpack. Unfortunately, there was not sufficient space to allow the tooling to deploy at the top of the container, hindering measurement at that location. Therefore, only a single salt-smart measurement was obtained. At the other two sites, sampling tools were inserted through the outlet vents and directed down into the annulus between the canister and the overpack. Following insertion into the overpack, the samplers were pneumatically pushed onto the canister surface in order to ensure good contact. Although the SaltSmart™ sensors could quantitatively measure salt loads if better contact was made, obtaining constant contact was difficult. Moreover, the sensors did not operate as reliably at temperatures over 80ºC, limiting their use, especially at the Diablo Canyon site. The Scotch-Brite™ pads were used to extract dust samples for mineralogical characterization but could not be used to estimate salt loads or surface concentrations, because the efficiency of collection is unknown, and because the actual area sampled is unknown.

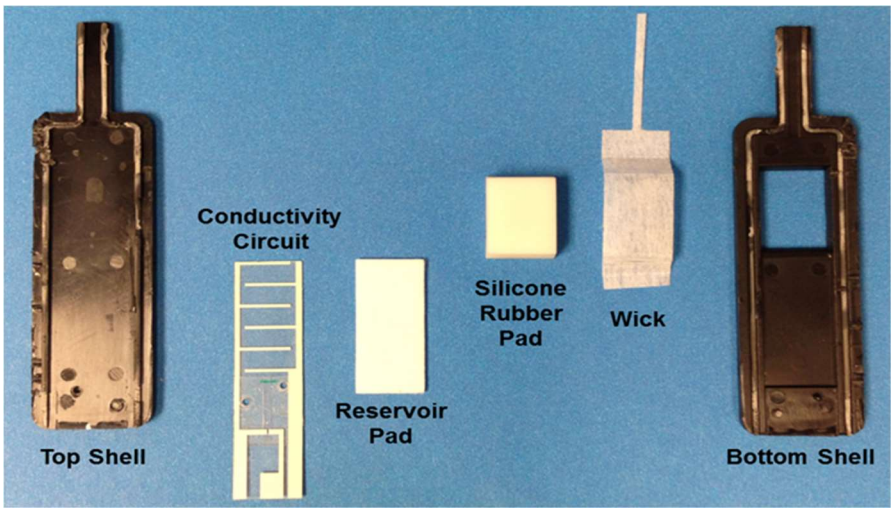

FIG. 6. Wet sampling device taken apart to show the inner workings. This device is approximately 80 mm long.

Samples were analyzed using several techniques. Dry pad samples were characterized using Scanning Electron Microscopy to identify organic matter and dust and salt mineralogy, and by X-ray fluorescence analysis to obtain semi-quantitative chemical analysis and element ratios for chemicals heavier than sodium. Finally, leachate from the Scotch-Brite™ and SaltSmart™ pads was analyzed to determine the chemical composition of the soluble salts present.

2.2.3. In-service dust analysis results

At all sites, there was a greater concentration of dust on the tops of the canisters and less on the sides. Dust was collected from all sampled locations on the canister surfaces and consisted largely of aluminosilicates and quartz, and metal particles (stainless steel and carbon steel) which are presumed to be generated from the canister manufacturing process. The soluble salts present on the canisters differed from storage site to storage site, and the compositions can mostly be differentiated on the basis of marine and near-marine locations.

2.2.4. Near-marine sites: Calvert Cliffs and Hope Creek

These sites were adjacent to brackish, calm water with prevailing inland winds. Dust on the vertical sides of the canisters was very sparse, with the top surfaces having a much heavier dust load. The mineralogical composition of the dust as determined by scanning electron microscopy was similar at the two sites; consisting mostly of quartz, feldspar and clays. Sulphates and nitrates dominated the soluble salts, with $CaSO_4$ being especially common. At both sites, NaCl was the only chloride salt present, and was observed as rare, isolated particles

The chemical analyses of the soluble salt and dust leachates from the two sites were also similar. At Calvert Cliffs the soluble salts were low in Na^+ and Cl^- and high in Ca^{2+} and SO_4^{2-} [10]. The Hope Creek soluble salts were mostly sulphates and nitrates, mostly of calcium. As with Hope Creek, there was very little chloride found [8].

2.2.5. Marine sites: Diablo Canyon

Diablo Canyon is located on the Pacific coast, and the dry storage site is one-half kilometre from breaking waves. At Diablo Canyon, sea-salt aggregates consisting mostly of NaCl and $MgSO_4$ with a little K and Ca were common dust components. The observed dust and salt loads on the dry pads from the vertical canister sides were light, and the SaltSmart™ sensors

measured low levels of chloride; however, high canister surface temperatures may have affected the performance of the SaltSmart™ devices, and the results may be questionable. Heavier salt loads were observed on the dry pads from the tops of the Diablo Canyon canisters, but because the tops were very hot (above 100ºC), SaltSmart™ measurements were not taken, and salt loads cannot be estimated. Detailed sampling results are documented in Ref. [10]. The abundance of chloride-rich sea-salts in the dust indicates that upon deliquescence, an environment that could support chloride-induced stress corrosion cracking will be present.

2.3. Residual stress

The third requirement for the initiation and propagation of a stress corrosion crack is the presence of a sufficiently large tensile stress. Based upon modelling, such stresses are believed to be present through the canister wall at welded regions. Stress distributions in actual storage canisters have never been measured, so to obtain data on weld residual stresses, a full-diameter canister mock-up has been made using the same materials, welding procedures, and manufacturer as a frequently-used commercial storage canister. The canister has an outer diameter of 1.7 meters, a wall thickness of 1.58 cm, and an overall length of 3.66 meters. The parent material is 304SS and the weld filler material is 308SS. Because we were more interested in residual stresses in the longitudinal and circumferential welds (where penetration could result in local penetration of the cask wall), a lid and closure welds were not fabricated. The base plate was also excluded, for cost and logistical reasons. The residual stresses will be measured at the circumferential and longitudinal welds and weld-repairs, heat affected zones, and in the base metal of the canister remote from the welds. Techniques that will be used include incremental deep-hole drilling, incremental centre-hole drilling, contour mapping, and X-ray diffraction techniques. All data will be shared with the NRC and EPRI and published in a US DOE UFD report. After residual stress testing, the mock-up will be cut into pieces and shared with partner organizations for further stress and corrosion experiments and testing. The end goal is to understand what areas of the canister SNF might be most susceptible to chloride-induced stress corrosion cracking, and to characterize stresses and material properties in base metal, weld, and heat-affected zones. This information will be used to inform decisions for inspection intervals, to identify locations on the canister that might be of greatest risk, to provide realistic sample properties and tensile stresses for corrosion testing, and to validate existing models for weld residual stress estimation.

3. CONCLUSION AND NEXT STEPS

Chloride induced stress corrosion cracking resulting from the deliquescence of atmospherically deposited salts could potentially result in the local penetration of the canister wall in existing SNF dry storage systems. Therefore, the US NRC is developing protocols for inspections of individual canisters while in storage. The research and development being performed by the DOE UFD will be used to help determine the need for on-site canister inspections, potential inspection start times and inspection intervals, the geographic locations around the country at highest risk for SCC, and specific locations on the canister that might be more susceptible. This work will provide a greater understanding of how canisters will age over the current licensing times and beyond to better inform regulators about the performance of long-term interim storage and subsequent aging management.

The DOE UFD program continues to identify research needs and to develop testing programs to evaluate the risk of SNF storage canister failure by SCC. Current experimental work evaluates the effect of limited chloride surface loads on the rate and extent of localized corrosion (pitting and crevice corrosion), which is a precursor for SCC. Additional testing is

planned that will evaluate pit and stress corrosion crack growth rates as a function of temperature, relative humidity, chloride surface loading, and crack depth.

Testing is also being carried out to understand the environment on storage canister surfaces. These tests evaluate the stability of different salts and brines on the heated surfaces of the storage canisters. Initial testing has been done with ammonium salts, which are the dominant components of inland salts, and which degas rapidly both in the solid state (prior to deliquescence), and after deliquescence. Additional testing will assess the stability of ammonium-bearing salts and brines containing NaCl or $MgCl_2$, conditions which might occur if road salts aerosols were incidentally deposited on canisters at inland sites. Finally, the long-term stability of heated sea-salt assemblages will be evaluated. These tests will increase our understanding of how different salts are retained on the canister surface, how the salt/brine chemistry changes with time, and how particle-gas conversion reactions, acid degassing, and decomposition processes can potentially affect the corrosiveness of deliquescent brines on the metal surface.

Lastly, to obtain data on weld residual stresses, a full-diameter canister mock-up has been made and the residual stresses will be measured at the circumferential and longitudinal welds and weld-repairs, heat affected zones, and in the base metal of the canister remote from the welds.

ACKNOWLEDGEMENTS

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AN INTEGRATED APPROACH TO CLOSING THE TECHNICAL DATA GAP FOR HIGH BURNUP SPENT FUEL PERFORMANCE DURING NORMAL CONDITIONS OF TRANSPORT (NCT)

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Abstract

This paper will cover the technical issues associated with addressing the technical gap of high burnup spent fuel performance when subjected to Normal Conditions of Transport (NCT). The theme of the paper will show how integration of engineering tools (e.g., testing and analysis), coupled with understanding of the basic science phenomena germane to this problem, is used to demonstrate understanding of how high burnup fuel will perform during Normal Conditions of Transport.

1. INTRODUCTION

The United States Department of Energy (DOE), Office of Nuclear Energy (NE) initiated a program in 2009 to develop the technical basis to support the licensing for long term storage and subsequent transportation of high burnup spent fuel. An initial focus of this program was development of a technical data gap analysis that identified the data gaps that needed to be addressed to support licensing. Given the number of gaps that were identified, a prioritization of gaps was developed in order to identify the high priority gaps where a focus would result in high impact in terms licensing support [1, 2].

From the gap analysis and prioritization work, performance of high burnup fuel during transport was identified as high priority. This paper summarizes the work that has subsequently been done to address the issue and to understand performance characteristics of high burnup fuel during Normal Conditions of Transport (NCT) operations. The work that has been done includes laboratory and field testing, as well as modelling and simulation. Further, this work addresses material property issues associated with cladding strength and ductility, cladding and fuel interaction related to the fuel rod response to cyclic loadings, and loading functions on the fuel generated from shock and vibration conditions that are representative of NCT. This integrated approach builds confidence in our overall understanding of high burnup spent fuel system response to transportation loadings.

It is impractical and unnecessary to conduct a complete and comprehensive R&D effort that addresses all aspects of fuel behaviour during NCT. Fuel operational histories, fuel mechanical properties, storage histories and environments, fuel retrieval from storage to transport systems, and thermal and mechanical loadings during NCT all play important roles

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in response characteristics of high burnup fuel during transportation. The question becomes: 'When is enough research and development enough?'

The first order answer is when there is sufficient confidence in the understanding of fuel behaviour under these various conditions.

The U.S. Department of Energy, Office of Nuclear Energy (DOE/NE), is supporting the development of the technical basis to support extended licensing of LWR spent fuel for storage and subsequent transportation. The DOE supports and manages the work described herein. This paper will provide a template for closing the technical gap associated with the transport of high burnup spent fuel under NCT.

2. FUNCTIONAL CRITERIA

Storage and transport integrity of high burn up fuel has been identified as a high priority technical gap that needs to be addressed. How and why did this particular gap become such a significant issue?

In the United States, storage of spent fuel in dry systems began in the early 1980's. These early designs were licensed for a period of 20 years with the option for extending the licensing based on performance. A demonstration project was conducted in the late 1980's to assess fuel performance in these dry storage systems. Results of this demonstration were satisfactory as no significant spent fuel degradation was observed after the demonstration storage period [3, 4].

One important limiting factor in this demonstration was that only low burnup fuel (i.e., <45 GWd/MTU) was placed in the demonstration cask. Over the last 30 years, the U.S. reactor fleet has been realizing increased efficiencies and fuel burnups >45 GWd/MTU is now routine. This was recognized fairly early in the dry storage era by the U.S. Nuclear Regulatory Commission (NRC). It issued a guidance document [5] that provided guidance for assessing spent fuel with burnups >45 GWd/MTU for transportation and storage operations. Since the issuance of this guidance document, the NRC has published 3 revisions to up-date the state of knowledge [6]. Importantly, ISG-11.3 provides threshold thermal criterion that limits fuel cladding temperatures to 400ºC during transfer, drying, and storage operations. This limits creep as well as the amount of hydrides in the cladding that will go into solution and then precipitate back as a radial hydride as the cladding cools. Radial hydrides decrease ductility and embrittle the cladding, making it more susceptible to fracture during storage and transportation operations.

Importantly for transportation, ISG-11.3 states that:

> *"Reviews of the transportation of commercial and non-commercial spent fuel will be handled on a case-by-case basis. Depending on the concentration of hydrogen in the cladding and the size, distribution, orientation, and location of zirconium hydrides, the spent fuel cladding with decreasing storage temperatures loses fracture toughness such that the stresses on the cladding when applied at high strain rates could cause fracture of the cladding. Therefore, staff is continuing to review the data on the mechanical and fracture toughness properties of commercial spent fuel cladding in order to develop further guidance to assist the applicant in meeting the requirements of 10CFR71.55, 10CFR71.43(f), and 10CFR71.51 for transportation".*

The lack of data relative to the performance of high burnup spent fuel during storage and transportation operations is what led to the identification of this issue as being a high priority to address in the DOE program.

3. INTEGRATED ENGINEERING APPROACH TO ADDRESS THIS TECHNICAL ISSUE

Experimental data collection and computer modelling and analyses have both distinct and complementary roles necessary to developing the understanding of spent fuel behaviour during NCT and in developing the technical basis to corroborate this understanding. Neither approach, followed exclusively, will be able to develop this demonstrated level of understanding within a reasonable timeframe or under finite funding realities. Rather, the use of experimental data, coupled with modelling and analysis, is necessary to leverage the benefits and advantages of each approach. Therefore, planning for experimentation and computer analyses must be done with both approaches in mind.

Experimental data provide the foundational understanding of material properties and material/system responses to specific applied environments. Design of tests with storage/transport applications provides specific insights into material behaviour that may affect the safety functions of storage/transport systems. This data is used to answer fundamental questions, provide guidance for more refined tests, and to use as benchmarks for validating computer modelling and analyses.

Computer modelling and simulation serves an important role in this process. Preliminary analyses can be used to help guide and define the test design. Parametric studies can identify parameters that drive a particular response and tests can then be designed to capture the response experimentally. Preliminary analyses can also provide guidance for locating data collection devices such as strain gages, accelerometers, and thermocouples.

In order to have confidence in the analyses, experimental data provides the critical function of being able to benchmark the modelling and analyses of a specific problem. Once a code is benchmarked, analyses can be performed with confidence that the results properly capture the response in a real situation. Modelling and analysis is fundamentally cheaper and more flexible than experimental work. Once the benchmarking is performed, analyses can be used to extend the technical basis work to a much broader set of problems. For example, the combination of fuel types (e.g., PWR and BWR), fuel cladding (e.g., Zr-2, Zr-4, Zirlo, M5), fuel operational histories (e.g., high burnup v. low burnup, duty cycles), time in pool storage, transfer operations from wet to dry storage, storage configuration and system design, and subsequent transfer of the fuel after storage to a unique transport system all have an effect on the integrity of the fuel and its response to NCT travel conditions. Once a code is benchmarked to an applicable data set, multiple analyses of many different scenarios and systems can be conducted with confidence that the corresponding results properly represent the response of the specific fuel being analyzed.

A caution needs to be stated in that it is important to recognize that benchmarked codes are only valid for the range of physics being considered. That is, when testing is done under specific conditions to record specific responses, one must be careful to not assume a code benchmarked to this data will properly capture response to conditions that may be orders of magnitude different from that tested. One important example is that of strain rate. Depending on how fast a test article is tested, response of that article to an input load can vary dramatically and any resultant modelling and analysis may incorrectly estimate true values.

4. CLADDING MATERIAL PROPERTIES

Fundamental to the issues identified in ISG-11.3 are the effects of hydride reorientation on the material properties of the cladding. As burnups increase, more hydrogen is diffused into the cladding in the form of circumferential hydrides. During drying operations in the transfer from wet to dry storage, cladding surface temperatures increase substantially, allowing the hydride to go back into solution. As the cladding cools the hydrides precipitate back out. Depending on the internal hoop stress and resultant strain in the cladding, the hydrides will reorient to a radial direction, thereby decreasing cladding ductility and increasing brittle behaviour. One of the DOE experimental programs focused on these characteristics is conducted at Argonne National Laboratories. Testing protocols have been specifically designed for cladding that has been defueled. Results of these ring compression tests (RCTs) have provided important information concerning the effects of radial hydrides on material properties. In particular, "…a DBTT (Ductile to Brittle Transition Temperature) of ≤20ºC can be achieved for peak drying storage hoop stresses ≤110 MPa for high burnup Zry-4 and ≤90 MPa high burnup ZIRLO™ and M5® cladding". [7]. This indicates that even high burnup fuel with these cladding types may have a ductile mechanical response under these conditions.

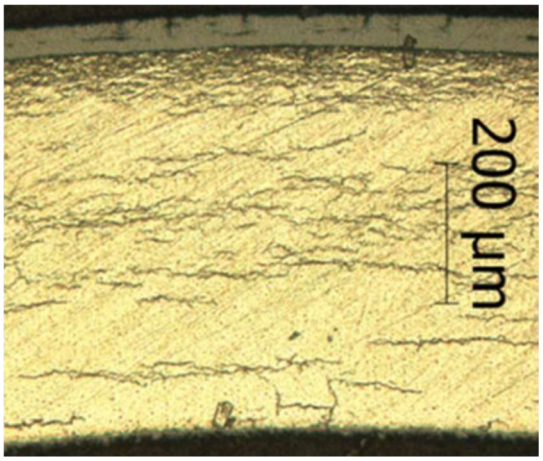

FIG. 1. High burnup ZIRLO™ cladding.

Fig. 1 is a micrograph showing high burnup ZIRLO™ cladding with associated hydride characteristics at 80 MPa peak hoop stress and 400º C. The ZIRLO™ cladding shows much less radial hydrides than Zry-4 and subsequent less radial hydride connectivity across the cross-section of the cladding, an important feature in maintaining ductility.

Fig. 2 shows the ZIRLO™ response to mechanical loadings as a function of cladding hoop stress. As stated, for hoop stresses <90 MPa, the cladding remains ductile across a range of hydrogen concentrations and temperatures.

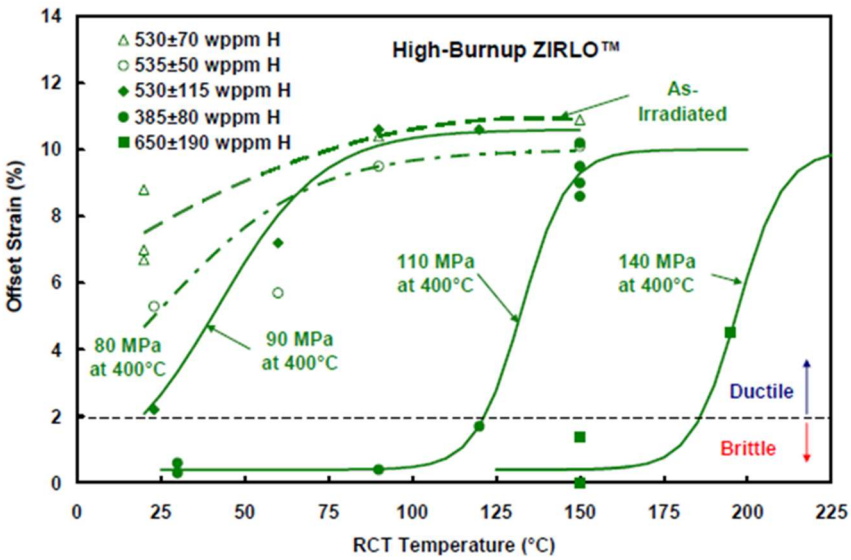

FIG. 2. Ductility data and DBTT for high burnup ZIRLO™.

This experimental program provides important insights into high burnup cladding material behaviour in a way that quantifies engineering judgment. Although important features of the cladding discussed point to a positive ductile response to mechanical loadings, it is important to note that the data set is relatively small and restricted to a limited number of cladding types, burnups, duty cycle histories, etc. While the results are encouraging, extrapolating cladding response across the range of fuel types and operational histories with confidence is not possible. However, given these results, coupled with the work that is presented in the following two sections, the trends look promising that high burnup fuel in general will maintain its integrity during normal conditions of transport.

5. FUEL RESPONSE CHARACTERISTICS

It is important to note that the experiments discussed in the previous section were on high burnup cladding that had been defueled. The response of the cladding in these experiments provides data on the ductility and DBTT values of the cladding itself but does not represent the way fuel will respond to mechanical vibrational and shock loadings that represent NCT. These types of loadings, as defined in regulations worldwide [8], provide the basis by which safety of the fuel can be evaluated during transportation. Since these loadings are in general bounding, fuel that will maintain its integrity during NCT will also remain intact during storage.

A program jointly funded by the DOE and the NRC at Oak Ridge National Laboratory (ORNL) is gathering data on the response of high burnup intact fuel subjected to cyclic vibration loadings [9]. Importantly, these tests are designed to capture the pellet-clad interaction (PCI) and pellet-pellet interactions that can significantly impact the mechanical response of the fuel rod. Fig. 3 shows the pellet-pellet and pellet-clad contact bonding in one of the fuel segments.

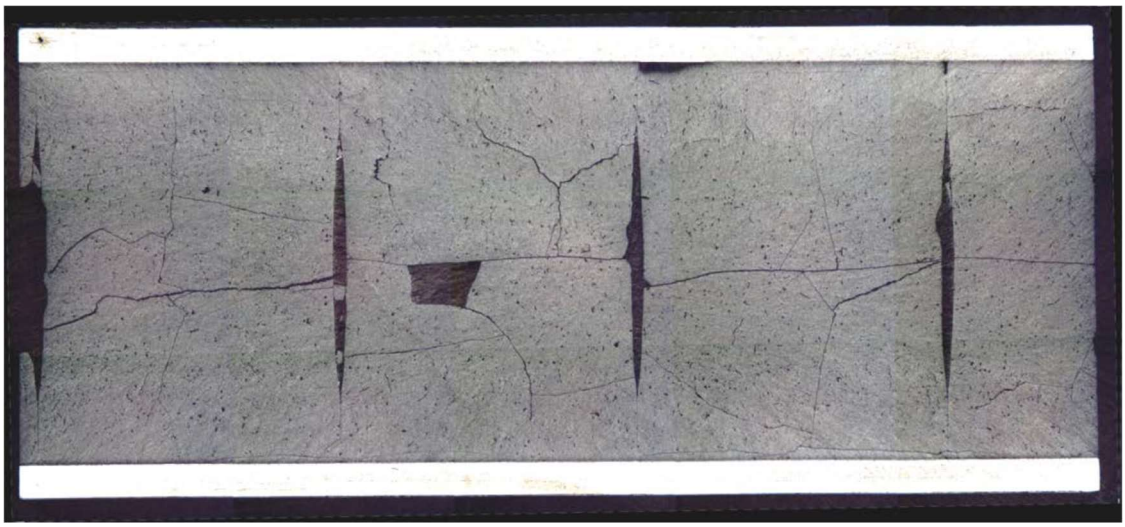

FIG. 3. Cross-section view of high burnup fuel and contact with Zr-4 cladding [10].

In a hot-cell, pure bending cyclic loading was applied to spent fuel high burnup rod segments. Initial results of the tests indicate that spent fuel is much stiffer than previously assumed and that the mechanical loading applied is resisted by the pellet to pellet bounding as well as the pellet to clad interaction. This stiffness relates to increased strength of the fuel to resist applied mechanical loadings. Fig. 4 shows the test configuration out-of-hot cell with the rod specimen in the centre of the fixture and the three linear variable differential transformer measurement gauges in the front.

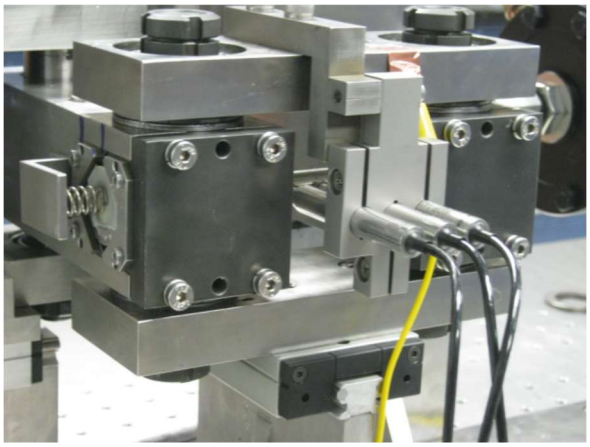

FIG. 4. Test apparatus for measuring spent fuel response to cyclic loadings [10].

Table 1 provides an overview of the effect of bonding on fuel rod stiffness. The effect is substantial, as can be seen.

TABLE 1. FLEXURAL RIGIDITY COMPARISON BETWEEN DIFFERENT BONDING AND DEBONDING CASES [9]

Interface bonding conditions	Flexural rigidity *EI* ($Nm^{2)}$	Reduction from perfect bond (%)	Reduction from only pellet-pellet debond to further pellet-clad debond (%)	Increase from with gaps to without gaps (%)
Perfect bond	44			
Pellet-pellet interface with gap debond, pellet-clad interface bonded	32	27		
Pellet-pellet interface without gap debond, pellet-clad interface bonded	40	9		20
Pellet-pellet interface with gap debond, pellet-clad interface debonded	16.5	62.5	48	
Pellet-pellet interface without gap debond, pellet-clad interface debonded	26.5	40	34	38

The effect on the fuel bonding, as described in [9], is:

> *"As the Young's modulus of the interface material increases, the fuel rod system stiffness increases accordingly. As debonding of the pellet-clad interface advances, the impact of the interface material property change is limited. The immediate consequences of interface debonding are (1) a shift in load-carrying capacity from the fuel pellets to the clad and (2) a reduction in composite rod system flexural rigidity. Therefore, the flexural rigidity of the fuel rod and the bending moment resistance capacity between the clad and fuel pellets are highly dependent on the interface bonding efficiency at the pellet-pellet and pellet-clad interfaces."*

Important questions remain. What is the extent and continuity of the bonding of the fuel to itself as well as the cladding? What is the bonding consistency in lower burnup regions of the fuel and in lower burnup fuels? While the initial results are very encouraging, more work needs to be done to better understand the bonding characteristics inherent in all types of spent fuel.

Developing data to investigate the impact of hydrides in the fuel cladding on mechanical behaviour (RCT tests), coupled with the cyclic mechanical tests to real spent fuel, provides an overall understanding of fuel response to a range of mechanical loadings. While the database is not complete, the database as it exists suggests that high burnup fuel with hydrides will

maintain its integrity under a wide range of loadings over a long span of fatigue cycles. The final parameter that needs to be investigated is the actual loads that will be applied to the fuel during NCT. If the loads are bounded by the mechanical tests that have been described, confidence is increased in the ability of high burnup spent fuel to maintain its integrity during NCT will be increased.

6. APPLIED LOADINGS REPRESENTING NCT TO SPENT FUEL

There has been much research conducted over the past six decades on how casks respond to transport loading conditions; both from NCT and from Hypothetical Accident Conditions. However, there has been little research or data collected and reported in open literature that investigates the loadings that are applied to the fuel during normal transport. To have confidence in judging the ability of high burnup fuel to withstand these applied loadings, it is critical to measure these loads on the fuel during transport.

The DOE program has conducted two important tests to measure the applied accelerations and strains resulting to NCT loadings [11, 12]. During NCT, shock and vibration loadings will be seen by the fuel. NUREG/CR-0128 [13] provides the range of shock and vibration (e.g., amplitude and frequencies) that should be used to evaluate fuel response to NCT loadings. This data was obtained from a 700 mile truck transport from Mercury, Nevada to Albuquerque, New Mexico. The data was collected on the outside of the cask, not on the fuel. While this is not a direct measure of the energy transmitted to the fuel, it accepted by the NRC as satisfactory for evaluating fuel response. This shock and vibration data was then reduced to a loading input file that was input into a shaker table at Sandia National Laboratories. The description of the test that follows and results can be reviewed in detail in the shaker table test report [11].

One reason there is little data in this area is that it is not possible to instrument spent fuel in any practical way during transport. As such, many compromises were made in the construction of the assembly due to budget, schedule, operational, and ALARA constraints. The fuel assembly that was used for this test was a surrogate 17 × 17 PWR. All but three of the rods that make up the assembly are made of hollow copper. Three of the rods are standard Zr-4. All of the rods were filled with lead rope to simulate the fuel. To capture a representative dynamic response, the important parameters are mass (each individual fuel rod and the entire assembly), rod stiffness (Moment of Inertia and Elastic Modulus), and the configuration of the assembly in a representative cask basket that properly represents mass and support configuration for the assembly.

Fig. 5 shows a picture of the surrogate assembly within the transportation basket (the top is removed) and instrumentation attached. Note the copper rods with two Zr-4 rods in the top-middle and top-corner positions. The third Zr-4 rod is in the bottom-corner position and cannot be seen in this photo.

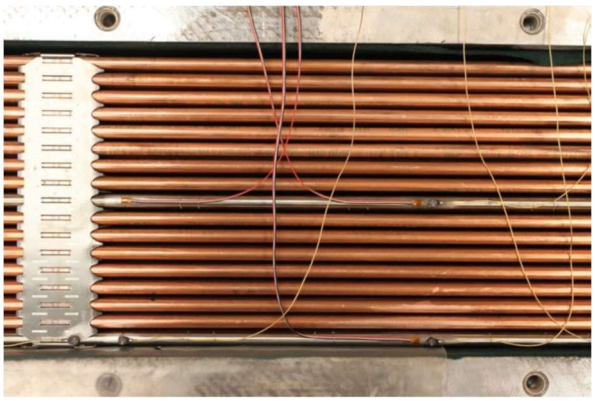

FIG. 5. Surrogate 17 × 17 assembly.

The mass of the surrogate assembly is almost exactly the same as a standard 17 × 17 PWR; 656 kg vs. 637 kg, respectively. Both the modelers and experimentalists involved agreed that matching the mass of the surrogate assembly to a standard PWR assembly was the most important parameter to properly simulate a representative dynamic response. This assembly was then placed inside an aluminium basket of the same mass as a NAC- LWT cask basket and that provides the exact same assembly support and clearance conditions. Given these test article and loading parameters, it is felt that the NCT shaker table test properly represents NCT loadings on spent fuel as well as response of the surrogate to these loadings.

The results of the tests showed a large margin of safety relative elastic yield strains. The maximum strain measured on the Zr-4 rods was 213 με. This value is compared to measured yield strain values of 7000-9000 με for irradiated and unirradiated fuel [14]. While this test incorporated many compromises from a true over-the-road load input to real spent fuel, the margin of safety relative to elastic properties provides confidence to move forward with more exacting tests that better represent the real condition.

To provide additional data to validate the shaker table results, an over-the-road test was performed with the same surrogate assembly to ascertain whether or not the shaker table test properly represented the load inputs expected for a true over-the-road test [12]. This was a 60 kilometre test that traversed over dirt roads, rough paved roads, potholes, highways, and railroad crossings to simulate a range of road conditions and speeds. The maximum strain on the Zr-4 rods in the surrogate assembly during this test was 143 με. While compromises and design parameters affected the attenuation of loads at certain frequencies, the two tests present similar results and provide confidence that the imparted loads to the surrogate assembly and cladding properly represent loadings expected from NCT for truck/highway shipments.

Fig. 6 shows a representation of applied strains from the two tests, compared with the preliminary analysis and the measured elastic stress/strain properties for irradiated and unirradiated spent fuel that was provided by [14]. Again, this comparison shows the large margin available for this test.

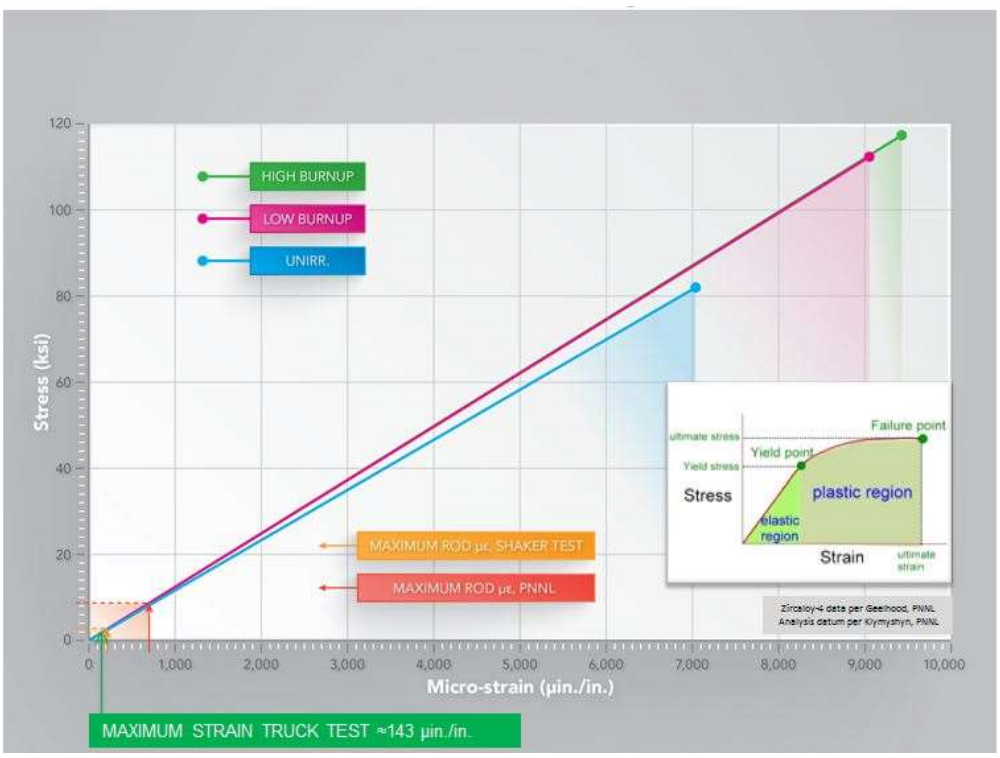

FIG. 6. Comparison of applied strains to measured elastic properties of Zr-4.

While the results of these tests are promising and provide confidence that high burnup fuel will maintain its integrity during NCT, there are many questions regarding the transfer of these test results to real high burnup spent fuel subjected to real NCT loadings. What are the effects of using lead instead of a more representative surrogate pellet? What are the differences in response characteristics of using lead rope instead of discrete pellets in the dynamic response? How does the lack of bonding effect the overall test response? How would a more representative transportation cask affect the fuel response?

7. CONCLUSIONS AND FURTHER WORK

A set of tests and results were presented to inform the behaviour of high burnup spent fuel to transportation loadings. While looking at each one of the tests in isolation raises questions about the fidelity of making conclusions relative to spent fuel response to these loadings, looking at the results in an integrated way increases confidence that high burnup fuel will maintain its integrity during NCT. Is the data set enough to make a definitive conclusion? Not quite.

There is on-going work to further investigate the consistency of the pellet-pellet and pellet-clad bonding. This effect is extremely significant, and it is important to understand the continuity of these bonds along the length of the rod and across different fuel types and burnups. The 30 cm drop requirement under the NCT regulations has not been addressed. This may result in a loading that exceeds the loading that has been applied to the cyclic tests or the shaker table and over-the-road tests. This will be addressed initially through modelling and analysis. A shaker table test will be conducted in 2015 to measure strains on the surrogate assembly resulting from rail NCT loadings. As part of this test, a lead rope will be used in one rod, lead pellets will be used in a second rod, and molybdenum pellets will be used in a third rod. This will allow direct strain comparisons related to the questions that have been raised.

Given the assessment of the dataset that has been accumulated to date, it is expected that these next tests will provide further confirmation that applied shock and vibration loads for NCT truck and rail conditions are low. Further, the resultant maximum measured strains on the surrogate rods are well below measured elastic properties of irradiated spent fuel.

8. ACKNOWLEDGEMENTS

While three authors are listed in this paper, the work is clearly supported by a large team of experts. M. Billone from Argonne National Laboratories has been the lead figure in the United States for the development of the Ring Compression Tests and resultant data related to hydride effects and DBTT. J.-A. Wang, H. Wang, H. Jiang, B. Bevard and R. Howard from Oak Ridge National Laboratories have designed, developed, and conducted the cyclic testing at Oak Ridge that has shown for the first time the strong dependence of rod response to mechanical loadings due to the pellet-clad and pellet-pellet bonding. P. McConnell, G. Flores, R. Wauneka, G. Koenig, D. Ammerman and J. Bignell from Sandia National Laboratories all supported the shaker table and over-the-road tests for the Normal Conditions of Transport loadings. N. Klymyshyn, S. Sanborn and H. Adkins from Pacific Northwest National Laboratories provided the preliminary modelling and finite element analyses that help in the design of the test and placement of the instrumentation.

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POSTER SESSIONS

SESSION 1

PERFORMANCE ASSESMENT OF SPENT FUEL ASSEMBLIES REMOVED FROM SHUT-DOWN RBMK REACTOR CORES FOR REBURNING IN OPERATING POWER UNITS

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1. INTRODUCTION

Now 11 power units with the RBMK-1000 reactors are in operation at three NPPs in Russia. In 2018 – 2035, the design lifetime of the RBMK reactors will expire. In case of step-by-step decommissioning, such fuel can be burnt in the RBMK units which are still in operation.

TABLE 1. SCHEDULED DECOMMISSIONING OF POWER UNITS WITH RBMK-1000 REACTORS [1]

	Unit 1	Unit 2	Unit 3	Unit 4
Leningrad NPP	2018	2020	2024	2025
Kursk NPP	2020	2023	2028	2030
Leningrad NPP	2027	2030	2035	-

The fuel of shut down for decommissioning reactor has some residual performance and its reburning will minimize fresh fuel needs. The amount of SFAs which have more than a half of their residual performance is too large to be reburnt at the operating units of the Leningrad NPP even if some SFAs are rejected as non-conforming by any reason. According to the analyses [2, 3], the reburning of the fuel removed from the shutdown reactors of the Leningrad NPP, in case of its decommissioning, will allow to save up to 1850 fresh fuel assemblies depending of the reburning mode.

2. SFA HANDLING PROCEDURE

The procedure for handling the SFAs during their preparation for reburning includes the SFA storage in the reactor pool, the transportation to the operating unit, the SFA preparation for loading into the core and the SFA health checking (Fig. 1).

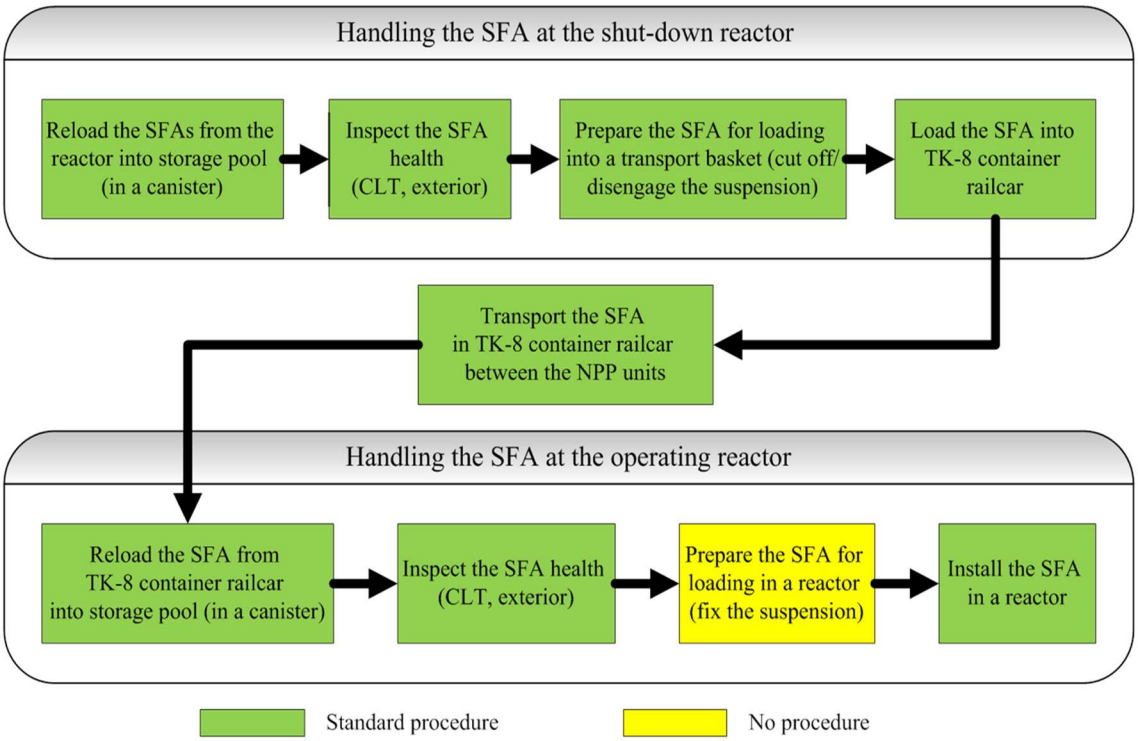
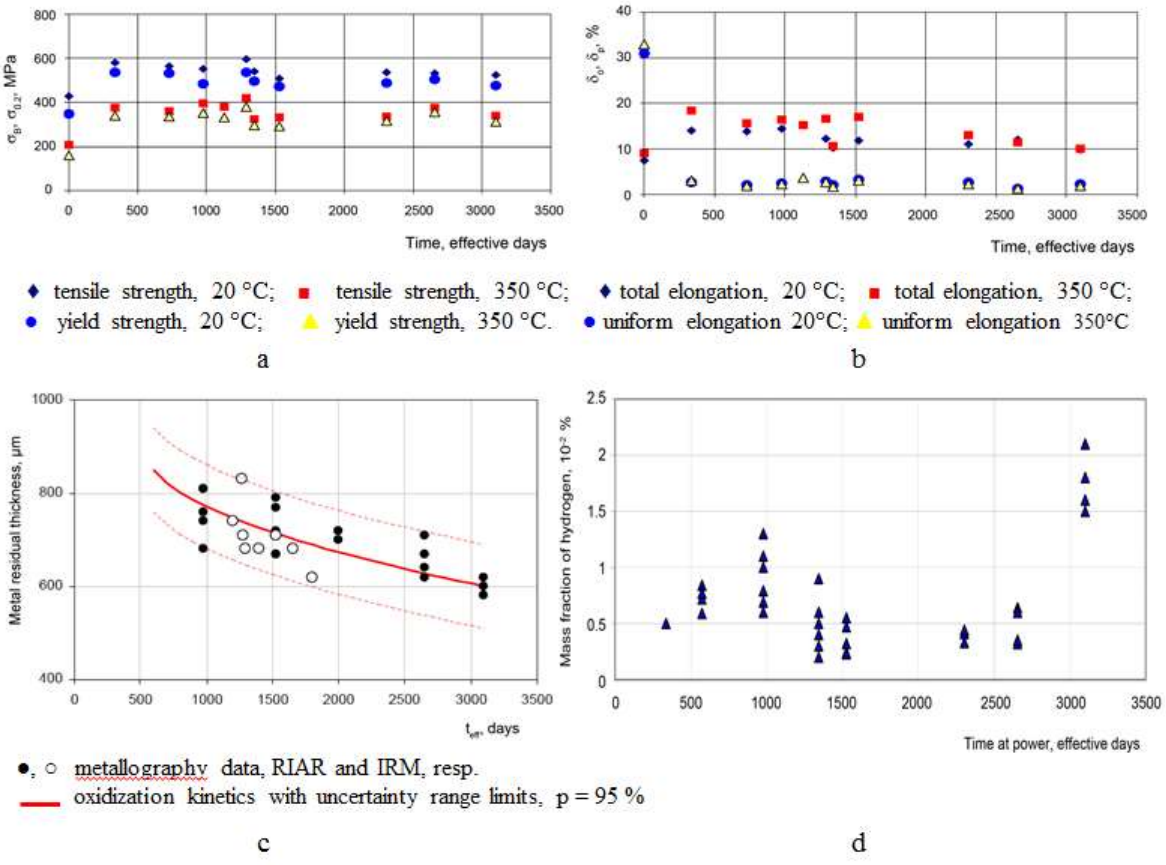


FIG.1. Flow chart of SFA handling.

3. SFA EFFICIENCY ASSESSMENT AND PROCEDURE SAFETY ANALYSIS

The results of the SFAs post irradiation examination in a wide range of their operation period were analyzed in order to assess the performance of candidate SFAs [4, 5]. The analysis of the corrosion characteristics and mechanical properties of the fuel rod claddings, state of the spacer grids and the elements of the SFA skeleton, and the SFA geometry has confirmed the efficiency of RBMK-1000 SFAs further burning after their irradiation for 1100 effective days (Fig. 2). The analysis of the examinations of RBMK-1000 SFA after long-term storage in the reactor pools [6] has not revealed any serious changes in the SNF health.

The standard equipment should be used to transport the SFAs between the units, but it has to be modified in order to minimize any risk of the SFA damage during its loading in / unloading out of the transport cask. The analysis of the modified procedure for transport the SFAs between the units of one and the same NPP has proved its safety under normal and accident conditions of operation. The main restrictions to the modified procedure are similar to those imposed to the standard one, i.e. the velocity of the container railcar shall not exceed 5 km/h and the acceleration during transportation shall not exceed 4g.



a, b – mechanical properties of the fuel rod cladding [3]; c – thickness of the fuel rod cladding below the spacer grid [2]; d – hydrogen content in the fuel rod cladding [3]

FIG. 2. Results of some analyses of RBMK-1000 SFA health.

4. SFA REQUIREMENTS

4.1. Requirements to fuel burnup:

Max 20 MWday/kgU or period of operation at rated power, max – 1100 eff. days (about 4 years).

4.2. Requirements to SFA cooling period:

At least one year of cooling in the core or in the reactor storage pool.

4.3. Requirements to SFA integrity:

All SFAs subject to further burning shall be leak tight. The cladding leak testing shall be performed for all SFA after their removal from the shutdown reactor and after their transportation to the operating power unit.

4.4. Requirements to SFA elements health:

Each SFA shall be visually inspected in compliance with the following rejection criteria:

- Any damage, displacement or absence of the spacer grid;
- Any dents or damage (incl. Corrosion) of the fuel rod claddings;
- No fuel rod plug;
- Any foreign object between the fuel rods;
- Any change of the fa geometry (distance between the fuel rod bundles, dimensions of the spacer grid, etc.) impeding the implementation of process operation during the SFA transport or irradiation.

All SFAs shall be twice examined: after their removal from the shutdown reactor and after their transportation to the operating power unit.

5. CONCLUSION

The nuclear fuel of the shutdown RBMK-1000 reactor has a good residual performance that can be used during the step-by-step NPP decommissioning. The reburning of the fuel removed from the shutdown reactors of the Leningrad NPP will allow saving up to 1850 fresh fuel assemblies. The feasibility to continue SFA operation is confirmed with the examinations of the fuel assemblies with burnup up to the maximum design value. At present the procedure for SFA reburning is being developed and the Leningrad NPP equipment is being prepared for further burning of the SFAs removed from the shutdown reactors.

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DEVELOPING REGIONAL SPENT FUEL STRATEGIES

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Abstract

Despite well-developed repository programs in Sweden, Finland and France, and encouraging progress in several other countries including Canada, the vast majority of states — including the U.S. — continue to struggle developing and implementing timely plans to safely and securely dispose of spent fuel and high level waste produced by nuclear power programs. Spent fuel continues to accumulate in cooling pools with limited storage capacity. On-site dry cask storage is a longer-term, but still temporary, solution and reprocessing generates significant waste streams that require geologic disposal. The lack of spent fuel management options strains the credibility of the nuclear community and undercuts public and political acceptance for current and future nuclear activities. It is unethical, irresponsible and ultimately unsustainable to push the problem onto future generations. It is also a proliferation and security issue. The decisions that countries make on the back end of the fuel cycle (activities that could include dry cask storage, shipping spent fuel to away-from-reactor storage or to reprocessing plants and, eventually, the final disposal of spent fuel or high level waste) can include sensitive nuclear technologies such as plutonium separation. Drawing on two years of work funded by the Hewlett and MacArthur Foundations, the principal investigators offer recommendations for developing spent fuel pathways that address broader fuel cycle concerns. Key findings include: many countries face storage and disposal siting challenges. This is particularly true in countries such as Japan and South Korea, where the need is urgent and growing, cooperative networks and regional frameworks for storage and disposal could be a productive way to address these problems in Asia and elsewhere. Such partnerships can enhance regional transparency and flexibility as well as improving global security and fortifying non-proliferation, the possibility of multinational options should not be used as an excuse for countries to neglect domestic responsibilities; all countries must have a national spent fuel management program. Conversely, national programs should not oppose concerted exploration of multinational approaches; geological disposal can be both a business venture and a public service. Even one success story could be a game changer, prompting regional waste management solutions in other parts of the world. Many nations with spent fuel management issues state that sustained U.S. engagement in regional fuel cycle/spent fuel initiatives is necessary. Many productive avenues of discussion can be pursued immediately, that promote cooperation well before the difficult phase of repository siting.

1. INTRODUCTION

Thirty-one countries plus Taiwan operate nuclear power plants; sensitive fuel cycle capabilities are concentrated in fewer than ten of those countries. Globally, reactors generate about 10 500 metric tons of spent fuel each year. More than 270 000 metric tons of spent fuel is held in storage worldwide, most at reactor sites. About 90 percent of that spent fuel is in storage pools, the remainder is in dry casks. In addition, the amount of separated plutonium continues to grow internationally because it cannot be consumed as quickly as it is produced; decisions about reprocessing are often disconnected from use of the recovered products.

Geological disposal is the internationally accepted strategy for permanently isolating spent nuclear fuel and high-level waste. Political and technical difficulties have so far delayed and, in many cases, prevented the construction and operation of any commercial repositories, though programs in several countries are making significant progress. Meanwhile, spent fuel continues to accumulate in cooling pools with limited storage capacity. On-site dry cask storage is a mid-term but still temporary solution, and reprocessing and recycling still generate significant waste streams that require permanent disposal. Both storage and disposal programs are needed: storage provides the interim solution until the repository is operating;

the repository program provides a permanent solution and the necessary public and political assurance that the storage site will not become a de facto repository.

Spent nuclear fuel needs to be stored for extended periods of time until the fuel has cooled sufficiently before repositories can begin accepting it — perhaps 40 years or more. In addition, increasingly high burn-up fuel and large dry storage canisters make moving spent fuel stored in such canisters problematic in the short to medium term because the allowable thermal and radiation limits for transportation can be substantially lower than the limits for storage, requiring an extended period of aging before the canisters have cooled down enough to be moved. In most cases it will be decades before repositories are open and able to accept waste in sufficient quantities to begin to significantly draw down the inventory. The longer spent fuel is stored, the less self-protecting it becomes as the strongly radiating shorter lived isotopes that provide the protection continue to decay, and thus as it becomes less radioactive, the greater security risk it presents. Therefore, high priority should be given to spent fuel storage options in program planning. Consideration should be given to siting interim storage facilities in locations where final disposal might also be feasible. Co-location with nuclear power plants has also proven to be a sound strategy in some cases.

Despite the well-developed repository programs in Sweden, Finland and France, and encouraging progress towards siting national repositories in the UK and Canada, the vast majority of countries continue to struggle with developing and implementing plans to safely and securely dispose of the waste produced by nuclear power programs. The lack of spent fuel management options — both interim storage and final disposal — strains the credibility of the nuclear community and undercuts public and political acceptance for all nuclear activities. Continuing along this path is ultimately unsustainable.

The United States has no effective program for managing and ultimately disposing of the spent fuel and high-level radioactive waste produced by the nation's nuclear power plants. Thirty years after the Nuclear Waste Policy Act was signed into law mandating the operation of a geologic repository by 31 January 1998, there is no active U.S. repository program though there is recent movement to restart the program. Years of work and billions of dollars have proven unsuccessful to date in meeting the nation's spent fuel management and disposal needs. This has a direct impact on the prospects for nuclear power domestically as well as the U.S. ability to lead and influence fuel cycle developments in the rest of the world; the U.S. is unable to lead by example.

For all countries, spent fuel presents both a waste management issue and a proliferation and security issue. The decisions that are made on the back end of the fuel cycle (activities that could include dry cask storage of spent fuel, shipping of spent fuel to away-from-reactor storage or to a reprocessing plant and, eventually, the final disposal of the spent fuel or the reprocessed high level waste) can include sensitive nuclear technologies such as plutonium separation. Meanwhile inventories of spent fuel continue to accumulate at reactor sites across the globe. Failure to manage these problems threatens national, regional and international security, exacerbates nonproliferation risks, jeopardizes the future of nuclear power and impacts serious efforts to deal with climate change.

2. COOPERATIVE APPROACHES

Cooperative approaches to spent fuel management offer a number of potential benefits. They could: enhance global security; promote economies of scale; enable spent fuel to be consolidated more quickly at storage facilities and perhaps allow earlier access to disposal facilities; lower environmental impacts; provide a wider choice of geological conditions;

present a business opportunity for the host country; provide an incentive for countries to forego enrichment and reprocessing; enhance regional transparency and confidence; enable the integration of engineering, hard science and social science research; create 'breathing room' (i.e. interim storage) while disposal options are developed; and demonstrate safe operation which helps build trust. If combined with a nuclear security regime, they could also enhance security and non-proliferation.

Many countries facing storage and final disposal siting challenges (geologic, financial and political) have strong interests in cooperating. In particular, about half of the countries currently operating nuclear power plants have fewer than five reactors. This means they do not accumulate spent fuel very quickly. It also means that spent fuel management becomes proportionately a very expensive component of their nuclear programs. Access to a disposal pathway that limits responsibility for spent fuel management may make more sense financially and, in most cases, politically. Even countries with larger programs may find economic, security and even political benefits in multinational approaches.

Spent fuel management in many European nations is considered a 'European challenge' that for some countries might be best solved by multinational cooperation. Notably, the European Council adopted a legally binding and enforceable European Commission proposal on 19 July 2011 which read in part: "Radioactive waste shall be disposed of in the Member State in which it was generated, unless agreements are concluded between Member States to use disposal facilities in one of them" [1]. Other regions, including several countries in Asia, may choose to take the same approach although it should be noted that the complexities of managing liability and other legal arrangements are minimized to a significant extent in the EU and Euratom due to an existing common understanding of basic concepts and an agreed regulatory framework.

The creation of cooperative networks and regional frameworks for storage and disposal would be a productive way to pursue spent fuel management, particularly in Japan, South Korea and Taiwan where storage space shortages in spent fuel pools are becoming acute. There are no such groupings or approaches at present in Asia. A regional approach could present a 'win-win-win' path forward (globally, in Asia and in the US), allowing for the safe and effective expansion of nuclear power while simultaneously reducing security and non-proliferation concerns and helping to solve the spent fuel/high level waste dilemma.

However, sensitivity to the mutual impacts of national and multinational initiatives is vital. The possibility of multinational options arising should not be used as an excuse for countries to neglect domestic responsibilities; all countries must have a vibrant national program for spent fuel management. By the same token, national programs should not oppose multinational approaches. International organizations such as the IAEA and the NEA should develop and promote consistent and even-handed views on both approaches.

3. TOWARD SOLUTIONS

For waste management programs to have best chance of success, it is extremely important to recognize that a national problem exists and for that recognition to translate into a national commitment to provide a solution in a reasonable amount of time. The problem cannot just be put off for future generations to deal with when the current generation is continuing to create the waste as it benefits from nuclear power. Several countries have had a national debate that led to a decision in principal which provided the foundation for, and clear national incentive to, address the problem. As the Blue Ribbon Commission on America's Nuclear Future (BRC) observed: "This generation has a fundamental ethical obligation to avoid burdening

future generations with the entire task of finding a safe permanent solution for managing hazardous nuclear materials they had no part in creating...while also preserving their energy options" [2]. However, trying to find a host community should not be the first step in any national or regional spent fuel management strategy. Rather, it should be the eventual outgrowth of an adaptive, staged approach. For regional approaches, there are many types of cooperation short of disposal (e.g. research and development on materials and aging issues common to all parties, joint canister construction, transportation methods) that could be helpful starting points for a dialogue. Achieving a consensus view on the potential benefits and drawbacks of a range of cooperation including consideration of an ultimate multinational facility is a key prerequisite in this process.

A consent-based staged, adaptive approach has allowed successful national programs to move forward toward a consistent set of goals (that is, safe storage and timely disposal of HLW/SNF) while preserving the flexibility necessary to adjust to both changing conditions and new opportunities throughout the lengthy program period. This has also been recommended by the BRC for the US program.

Understanding that siting spent fuel storage and disposal facilities is extremely difficult, creative approaches to community engagement, such as the design of the Dutch Central Organisation for Radioactive Waste's Highly Radioactive Waste Treatment and Storage Building (HABOG) which is orange and will be repainted in progressively lighter shades until it is white in about 100 years to reflect the reduction in the waste's heat production, should be publicized widely and adapted to local conditions if possible [3].

Viewing the management of commercial spent fuel as a contribution to national security is a powerful motivator — and has driven US and Russian threat reduction programs converting research reactors using HEU fuel to LEU and removing HEU from as many sites and countries as possible — but this argument has not been sufficiently emphasized with commercial spent fuel. According to a 2007 US-Russian National Academies workshop, "arrangements that would provide assured return of spent nuclear fuel could provide a much more powerful incentive for countries to rely on international nuclear fuel supply than would assured supply of fresh fuel, because assured take-back could mean that countries would not need to incur the cost and uncertainty of trying to establish their own repositories for spent nuclear fuel or nuclear waste" [4].

To be sure, large challenges remain for regional/international storage, disposal and take-back initiatives but the alternative — requiring each nuclear power nation to be self-sufficient at the back-end — is equally challenging. Serious consideration of factors such as security, non-proliferation and cost require the attention of all nuclear nations, particularly countries with small nuclear programs and limited resources but also countries with large nuclear programs.

Economics is not necessarily one of those challenges. While spent fuel disposal appears to be quite expensive, it is in fact a small fraction of the overall cost of electricity generated by nuclear power — roughly 5% of the total cost to the consumer. Therefore, even significant rises in the cost of waste management do not present a substantial penalty to overall delivered costs, particularly when compared to the potential costs of failure to provide a workable disposal solution. Indeed, geological disposal can, in principle, be both a business venture and a public service. And while no viable business ventures have been initiated yet, the business case for spent fuel management is not the problem. What is needed is a driver (i.e. country/countries and an initiative) to move things forward.

Several developments may serve as game changers for timely waste and broader fuel cycle management solutions in other parts of the world, including:

- *The first operating national repository for HLW/SNF*. This might help to build confidence and overcome political resistance in other parts of the world, although it should be noted that licensing and operation of the U.S. Waste Isolation Pilot Plant (WIPP), the first custom-built deep repository, albeit for what would be considered intermediate level waste from military programs rather than HLW, did not have this effect;
- *One or more countries providing a spent fuel take-back service*. This could be an effective solution for many nuclear waste generators (from a security, safety and financial perspective) if an appropriate supplier country was willing to offer this service. Russia is promising to take back spent fuel as part of its nuclear reactor deals with Iran, Turkey and Vietnam. Providing that the country can ensure state-of-the-art competence and facilities to store and/or dispose of the spent fuel, the international community should support any and all such offers, recognizing that they enhance regional and global safety and security;
- A renewed multinational decision to minimize the impacts of climate change and a fundamental commitment to nuclear power as a key element in achieving that goal.

4. CONCLUSIONS

The lack of a repository program strains the credibility of the nuclear community and undercuts public and political acceptance for all nuclear activities.

States must recognize that a national waste management problem exists and translate that recognition into a national commitment to provide a solution. Both storage and disposal programs for HLW/SNF are needed. Consideration should be given to siting interim storage facilities in locations where final disposal might also be feasible.

Spent fuel disposal appears to be quite expensive but is in fact a small fraction of the overall cost of electricity generated by nuclear power. Geological disposal can, in principle, be both a business venture and a public service.

The almost non-existent back-end market potentially offers opportunities for leveraging the provision of a spent fuel disposal pathway in return for a commitment not to pursue sensitive fuel capabilities such as enrichment and reprocessing.

If reprocessing is pursued as part of a closed fuel cycle, the stockpiling of plutonium should be avoided, and recycling should be undertaken as soon as possible afterward to reuse the separated plutonium promptly to the maximum extent.

A consent-based staged, adaptive approach to spent fuel management is proven global best practice. Trying to find a host community should not be the first step in any national or regional spent fuel management strategy, rather the eventual outgrowth of an adaptive, staged approach. Types of cooperation short of disposal (such as cooperative research and development and joint canister construction) can be helpful starting points for discussions aimed at regional approaches to spent fuel management.

Viewing the management of commercial spent fuel as a contribution to national security would be a powerful motivator.

Many countries facing storage and final disposal siting challenges (geologic and political) have an interest in regional approaches to spent fuel management. New nuclear entrants with small power programs or plans for such programs have a strong interest in regional approaches to spent fuel storage and disposal given that spent fuel management is proportionately a very expensive component of their nuclear programs.

Large challenges remain for multinational storage and disposal initiatives but the alternative – requiring each nuclear power nation to be self-sufficient at the back-end – is equally challenging. Sensitivity to the mutual impacts of national and multinational initiatives is vital.

Given the periods of time required for spent nuclear fuel to cool sufficiently and before repositories are open and able to accept waste, high priority should be given to spent fuel storage options in program planning.

It is important that a clear analysis demonstrate, as quantitatively as is feasible, the differences between once-through fuel cycle programs and reprocessing/recycling programs with regard to national security and non-proliferation, as well as their impact on waste management in terms of such variables as costs, volumes, physical forms, toxicity of wastes and long term containment.

Addressing non-technical issues when dealing with the public and their elected representatives at all levels of government has been shown to provide substantial benefits and has helped to make the siting of waste management facilities successful in several countries. Examples of 'added value' include generating feelings of pride in the disposal mission (and spreading that message through site tours and speaking engagements) and providing a public service as well as contributing to regional and international security.

Innovative solutions to spent fuel management in other parts of the world should be publicized. Discussing efforts that have been unsuccessful, and explaining why, can also serve a valuable educational purpose. Lessons learned should not be limited simply to the nuclear industry; important lessons can be drawn from other industries that require the use of controversial facilities.

There are many technical and non-technical topics that can be studied collaboratively by regional partnerships that avoid the third rail of finding host sites, including:

- Long-term performance of wet and dry storage, long-term performance of SNF inside dry casks and different cask types;
- Spent fuel transportation;
- Security aspects of spent fuel management;
- Siting strategies and criteria for (national) storage and disposal facilities;
- Comparison of direct disposal with (advanced) reprocessing strategies focusing on resulting waste inventories' characteristics;
- Public communication and dialogue.

The establishment of a 'virtual multinational laboratory' to support collaborative regional/international work on spent fuel management challenges might be one way to foster technical and institutional cooperation.

The need for action on creating spent fuel management pathways is urgent and first steps must be undertaken now. These steps should be small but concrete, with the initial emphasis on confidence-building measures.

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This paper reflects the status of ongoing work by the Nuclear Threat Initiative, funded by a MacArthur Foundation grant, to explore ways in which further international cooperation on challenges at the back end fuel cycle can be mutually beneficial to cooperating parties.

DEVELOPING A SPENT FUEL MANAGEMENT STRATEGY FOR A NEWCOMER NUCLEAR POWER COUNTRY

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Abstract

It is important to ensure that consideration of spent fuel and radioactive waste management is not postponed during the early phases of using nuclear technologies in new nuclear power nations. This is true, in particular, for the challenge of planning for final disposal of long-lived wastes from nuclear power production. Newcomer countries have an opportunity to plan from the outset an integrated back-end programme that considers all relevant facilities and activities required for waste production, treatment, storage, transport, packaging and disposal. The approach we adopt is to consider a series of key questions related to all of these waste management activities, for each activity and for each waste stream: What wastes arise?, How will these be treated, characterised, stored and disposed?, Where should the necessary facilities be located?, When is the optimal time to bring each type of facility on line? The strategic analyses consider all nuclear options kept open by the country in question. These can include direct disposal of spent fuel, reprocessing (foreign or domestic), early or late disposal of wastes, separated or co-located storage, conditioning and disposal facilities, and purely national facilities or the use of shared regional stores or repositories. Practical outputs from such assessments are a compilation of key decision points and a roadmap, extending out for many decades. The paper outlines high-level recommendations for newcomer waste management programmes.

1. DEVELOPMENT OF AN INTEGRATED NATIONAL SPENT FUEL AND RWM STRATEGY

Developing an integrated strategy for introducing nuclear power (NP) that incorporates considerations of spent fuel management options alongside options and schedules for radioactive waste management (RWM) can avoid many future technical problems, reduce economic problems and enhance societal acceptance of the nuclear power programme.

The majority of newcomer countries would opt to store and then dispose of spent fuel as part of the national RWM programme, if they decide not to, or are unable to enter fuel-leasing or reprocessing agreements. Although timescales can appear long and, perhaps, relaxed, many aspects of a RWM programme require long lead times and the credibility of the total NP programme can depend on factoring this into planning.

An early requirement is to establish an appropriate legislative and regulatory environment, supported by the relevant knowledge base. When the first nuclear plant comes on line, a newcomer country should have a small core body of waste management experts to ensure that critical issues are not neglected. This can be supported by external expertise and networks, but core competence is essential. Financial structures need to be set up to ensure that funding is available for safe management of spent fuel and wastes throughout and beyond the expected lifetime of the planned NP programme. Calculation of lifetime costs requires a feasible RWM strategy to be devised that incorporates all necessary facilities, options and contingencies. The RWM strategy must include all waste types and must cover the long period of time that may occur between waste production and ultimate disposal.

Immediate technical issues are generally the design and commissioning of facilities for handling, processing and storage of the low-level wastes (LLW) and intermediate-level

wastes (ILW) that will start to be produced as soon as the first reactor commences operation. Planning for disposal of this waste needs to be started early too. There is considerable experience in these areas in the developed nuclear power programmes. Simultaneously, planning for the long-term storage of spent fuel needs to be incorporated into the overall RWM programme.

An additional factor in these early considerations is whether some aspects of the RWM programme and some facilities might be shared, especially if there are more newcomer nations in the same region that could optimise their NP programmes by working together.

The establishment of a programme and timescale leading to the implementation of RW disposal facilities has historically caused the most difficulty for long-established NP programmes. Numerous LLW disposal facilities exist, in many countries, but the first repositories for disposal of spent fuel and HLW will not be operational until the next decade. This has left the global nuclear power industry open to heavy criticism from opponents of nuclear power – a factor that newcomer countries would presumably wish to avoid. However, for newcomer NP programmes it is impractical and unnecessary to implement geological disposal facilities on a short timescale — it will be many decades before the volumes of spent fuel or HLW from the first reactors are sufficiently large, and the heat emission sufficiently low, to allow disposal. Nevertheless, stakeholders and commentators in newcomer NP countries will expect to see a credible disposal strategy and pathway to eventual disposal.

An approximate timeframe for a RWM programme is set by the operational and decommissioning lifetime of the initial nuclear power plants (NPPs) and any further NPPs that may be developed at a later stage and the longest pre-disposal times required for the latest arising wastes. For a new NP programme, this timeframe is about 150–200 years: e.g., a nuclear power programme lasting 90–100 years and 60 to 100 years of pre-disposal storage for the last fuel to be discharged. During this timeframe there are likely to be significant developments in both national energy policy and global nuclear power and fuel cycle technologies. Such changes mean that the eventual amounts of different types of waste that could be generated must be expected to differ from the initial estimates. It is thus important to ensure that a RWM strategy acknowledges these variables and uncertainties and is robust from the outset, so that it can adapt to developments. Not only does it need to be flexible and adaptable, it must also provide credible solutions at any and all times throughout the NP programme.

A useful starting point is to develop a reference scenario (or a range of conceivable scenarios) for the future NP programme, taking into account relevant political, legal, societal and economic issues. This allows a first estimate to be made of the waste types and volumes arising over time. Then, the technical waste management strategies to be considered are identified and their implications examined (including infrastructural, societal and financial requirements). Choices are determined by the scope of nuclear activities foreseen for the future, the scale of the NP programme, the relative merits of technology imports versus build-up of autonomous know-how, etc. A prudent strategy balances keeping options open and establishing a sufficiently concrete planning base. Other decisions that affect strategy include choice of a financing system, allocation of responsibilities, etc.

The next step is to generate specific waste management scenario sets, combining options related to:

- Fuel cycle choices (leasing, direct disposal or reprocessing of spent fuel);

- What technologies are foreseen (for waste treatment, storage and disposal);
- Where might the waste management facilities be located (independently or co-located with others);
- When activities must be initiated (as early as possible, technically optimised, delayed for economic optimisation) and for how long.

The pros and cons of scenarios are analyzed and the scenarios ranked (also accounting for national preferences) to arrive at a manageable number that can be kept open, together with decision dates and criteria for narrowing down. Throughout, it is important to identify uncertainties in the programme and to maintain flexibility.

2. NATIONAL INVENTORY OF SPENT FUEL AND RADIOACTIVE WASTES

The RWM strategic plan requires estimates of all the types, quantities and times of arising for all radioactive wastes, for many decades into the future, and it is advisable to include those from outside the nuclear power sector (e.g., NORM-TENORM and MIR wastes). Challenges are that:

- The numbers and types of nuclear facilities that will be implemented are not certain;
- The trend in all nuclear technologies will be to continue minimization of waste quantities;
- Developments in waste conditioning will lead to changes in form and quantity of waste.

The inventory estimates must thus be based on the NP scenario assumptions discussed above, with care being taken to ensure that these are sufficiently conservative. Inventory description can be organised according to the source of the wastes — e.g. NPP and other facility operations; decommissioning; reprocessing (if utilised). The standard IAEA classification scheme can be used as the basis for developing an initial national inventory. More specific inventories will be required later in the RWM programme.

3. ALLOCATION OF ROLES AND RESPONSIBILITIES

All countries with RWM responsibilities must have a legal, regulatory and organizational framework within which to work — this is explicit within the IAEA Joint Convention [1]. The framework consists of a national policy, a strategy for implementing the policy and a waste management system, including all necessary components. Responsibilities for waste management are divided between the state, the regulator and the waste producer. The prime responsibility for safe RWM rests with the waste producers, who might undertake RWM directly or establish a dedicated implementing body — a Waste Management Organisation (WMO). The most recent IAEA publication [2] to include advice on responsibilities for waste management identifies the first three phases in implementing a nuclear power programme and the WM activities that should be completed by the milestones at the end of each phase.

4. ESTABLISHING A WASTE MANAGEMENT ORGANISATION

Structural questions to be considered in establishing a waste management organisation are affected by:

- The spent fuel and waste inventory;

- The activities assigned to the WMO (e.g., conditioning, transport, interim storage, siting, over packing, disposal, remediation, commercial aspects);
- Links to other stakeholders (waste producers, government, public, national laboratories, contracting companies etc.);
- The level of activity undertaken directly within the WMO (e.g. Actual WM activities, or mainly contracted-out project management);
- Resources available and the time at which these become available.

The WMO will have a key role in interacting with political, governmental and other bodies at the policy level and direct strategic planning responsibilities, so it must be led by high-quality executive staff. A competent and motivated project staff is also required to accomplish the technical work successfully and to interact with external groups. Project staff should be both good managers and competent technical experts. Their diverse backgrounds (from industry, universities etc.) will make further and continued training of special importance. Professional financial management is also essential. Sound science and high quality engineering must be utilized in developing, constructing and operating world-class facilities. Close interactions with public figures and scientific and technical bodies at national and local levels must be maintained. The public must be fully and openly informed and consulted directly in key areas (e.g. in siting of facilities). All the skills necessary need to be represented at a senior level within the WMO, with an important decision being how much work to carry out in-house and how much to do externally, by specialized contractors. In all key areas, in-house personnel need to manage to a level that allows conceptual and technical input to be given to contractors, while retaining the responsibility for key decisions within the WMO.

The spent fuel and RWM programme will be phased and will last many years, with costs varying considerably between phases. Although there are high costs and significant uncertainties for options such as fuel reprocessing or long-term storage, among the largest and least known costs are those for constructing and operating a geological repository. However, many of these costs arise far into the future and will depend to some extent on the size of the inventory, whereas, at the outset, the funding needed for the initial phases of the WMO is of immediate concern. This is independent of the inventory, depends on the extent of the technical and social science programmes and, most importantly, on the facility site selection and site characterisation strategy. A small WMO might run its baseline programme on an annual budget of some millions of dollars, with siting costs depending upon the facilities to be sited and the strategy adopted. For example, a geological repository site characterization programme might last for 5 years or more and can typically cost 10-50 million dollars per year. Prior to site-related work beginning, there would be a period during which the WMO would establish itself, staff-up and tackle the initial political, legal and financial issues. The technical challenges facing the WMO in its initial years are more concerned with the treatment, and storage of operational wastes, which must be tackled in close cooperation with the NPP operator.

5. ACTIVITIES AND OPTIONS IN THE NATIONAL RWM STRATEGY

The development of the NP scenarios described above will lead to alternative options for most of the essential component activities and facilities in the national strategy. Some of the generic elements that might be found within strategic and technical options are discussed below.

5.1. Spent fuel reprocessing

An important RWM strategic decision concerns the option of reprocessing abroad, rather than 'direct disposal' of spent fuel. Some national programmes have deployed overseas reprocessing to get SF 'off site' for prolonged periods until the return of HLW becomes inevitable and has to be managed. Timely, integrated planning for SF storage and associated facilities at the outset of a NP programme means that decisions on reprocessing can become a matter of simple economics, weighing storage costs and disposal cost against reprocessing costs and HLW disposal costs. In reality, decisions on reprocessing (and/or MOX use) are only weakly linked to preferences within an overall RWM strategy: policy and nuclear power generation drivers are much more important.

5.2. Management of spent fuel

The different strategic options for managing spent fuel require different technologies. An attractive option for newcomer NP programmes could be to lease the fuel from a supplier and return it after use. At present, most countries regard spent fuel as waste to be disposed of directly in a geological disposal facility, after a certain cooling period, which can extend up to some decades (depending upon the layout and the host rock of the disposal facility). Thus, this strategic option requires storage facilities from the outset, an encapsulation plant before disposal, and a disposal facility.

5.3. Management of other operational wastes

The wastes that need most immediate attention when a country introduces nuclear power are those that are continually produced during reactor operation, treated and conditioned to convert them into a form suitable for subsequent transportation, storage and disposal. The principal aims are to minimize the volume and to reduce the potential hazard of the waste. Facilities for processing operational wastes should be commissioned by the start-up of the NPP. The conditioning and packaging for storage of LILW should meet the WMO's waste acceptance criteria at the time of initial conditioning. Raw wastes can be stored at each NPP for many years, or can be conditioned on-site or shipped to a centralized conditioning plant.
Drivers for developing a LLW repository (LLWR) are mostly practical — the volumes of LLW that accumulate become large and there is no need to wait for a decay period before disposal. Operating a LLWR illustrates to the public that there is a definitive end-point for at least some waste streams from NPPs and a disposal route is often needed anyway for LLW that results from medicine, industry and research.

5.4. Interim storage of radioactive wastes

A strategic RWM policy option might be to store all wastes for as long as reasonable/possible — a default position that almost all nuclear programmes worldwide have had to accept in the absence of disposal facilities for higher activity wastes. The longer that any waste can be stored, the lower the discounted investment cost of locating and constructing a disposal facility. This must be offset against the growing volume requirement for storage and the eventual costs of refurbishment of stores with multi-decade lifetimes.

Prolonged initial storage of spent fuel in ponds at the NPPs makes subsequent handling easier and less hazardous. One driver for prolonged storage of SF subsequently is the possibility that its resource potential may eventually outweigh its liability. Arguments in favour of early disposal are related to the relative security (against both natural and human hazards) of

underground compared to surface-based facilities. Consequently, an alternative strategic policy might be based upon disposing of all wastes in a timely fashion, as soon as it is reasonable to do so.
The minimum cooling time before spent fuel can readily be disposed of in a geological disposal facility is around 50 to 60 years — perhaps longer, for high burn-up fuels. Even if this delay were not necessary for technical reasons, there are other arguments for delaying disposal. For newcomer nuclear programmes, many years of NPP operation would be required to accumulate an inventory of spent fuel that justifies embarking on an expensive deep disposal facility project. By extending surface storage times for decades, the large expenditures needed for implementing such a solution can be postponed.

If reprocessing abroad is adopted in the RWM strategy, the returned vitrified HLW and compacted hulls and end-caps will eventually have to be stored prior to disposal. Storage of vitrified HLW uses similar technology to storage of spent fuel, with considerably reduced volumes needing to be managed. The separated plutonium poses greater storage problems, as it is a highly sensitive material from a proliferation and security standpoint. It deteriorates with time and becomes less suitable for processing into MOX fuel, so it is usually made into MOX fuel for early reuse. Reprocessing agencies currently store plutonium for their customers at a cost, or use it to make MOX fuel.

5.5. Geological disposal of spent fuel and radioactive wastes

The only recognized safe disposal route for spent fuel, vitrified HLW and ILW is in a geological disposal facility (GDF). There are many arguments for delaying the construction of a GDF, with cost being a primary concern. The bulk of GDF costs are associated with spent fuel and/or HLW disposal, rather than ILW. Since at least 50 years pre-disposal cooling of SF or HLW is needed, the only technical arguments for having an early GDF are a requirement to dispose quickly of ILW (for which there are few drivers) or a perceived advantage in identifying a GDF site early, so that other facilities, which could be needed sooner (e.g. a centralised interim store for all waste materials), can be developed along with some of the surface or access works for a GDF.

However, other reasons for an early GDF might outweigh the arguments for delaying it for 50 or more years — chief among these being nuclear security and safeguards for spent fuel. Surface storage of spent fuel is less secure than deep underground storage or disposal, for obvious reasons of accessibility and dispersion hazard potential. Also, disposed radioactive waste is considerably less susceptible to events and processes affecting the surface environment and consequently safer than when stored in an interim storage facility of any kind. The potential impacts of both natural and societal disruptive events need to be considered, on a timescale of many decades. There is also a socio-political argument for early disposal: a newcomer NP programme is easier to present and justify if the waste management aspects are demonstrably all in place, rather than having one of the most important and controversial activities postponed for several generations.

6. FACILITY SITING OPTIONS IN THE RWM STRATEGY

A newcomer programme will find several siting options available for the storage and disposal concepts outlined above.

6.1. Centralised interim storage (CIS)

All categories of waste can, in principle, be stored at interim facilities at the NPP sites until disposal facilities are available. The advantages of a CIS include the potential to reduce costs, as fewer facilities would be needed, which could in turn enhance security (because spent fuel is located at only one site for many decades — possibly up to 100 years). A single CIS allows the benefits of a single 'site culture' and can help improve uniformity of products for disposal. There is also the opportunity to co-locate 'attractive' facilities. If a CIS is located at a GDF site, then its construction and operation provide an early presence on and activities at the site, which might help improve local infrastructure and community relations. Balanced against these advantages is the burden of locating and developing an additional nuclear site, unless the CIS is at the GDF site.

6.2. Co-location of several facilities

It is commonly assumed that an encapsulation plant for producing final disposal canisters for spent fuel or HLW would be located together with the GDF, for technical reasons to do with the transport of massive, overpacked containers (although this is not always the case: e.g., Sweden). Other facilities might usefully be co-located too. Co-location of all waste management facilities (LLWR, GDF, CIS) at a single site in a 'nuclear park' has the attraction of a single work-force and culture, as well as providing the opportunity to locate other high-tech facilities (e.g. research laboratories) at the same place to take advantage of the concentration of scientific and technical expertise. This could also have societal benefits.

6.3. Sharing facilities with other countries

Newcomer countries might identify possibilities for development of some facilities in collaboration with other (especially, neighbouring) countries, taking advantage of the drivers of shared costs, expertise, concepts; cementing national NP programmes within the region; wider ranges of technical siting options etc. Sharing facilities has timing implications, with optimum implementation dates needing to be negotiated between all users, which may not be ideal for any individual RWM programme.

7. SOME RECOMMENDATIONS FOR A NATIONAL SPENT FUEL AND RWM STRATEGY

For all newcomer NP programmes, a task that should not be postponed is the establishment of a National Radioactive Waste Management Strategy (NRWMS), including the management of spent fuel. Below we make some recommendations about how the NRWMS should be developed.

(1) The NRWMS should be based around a set of Principles derived from internationally accepted waste management principles and national considerations and requirements;
(2) The suggested Principles underpinning the NRWMS are as follows:
    a) Safety and security must be the prime concerns of the NRWMS. All radioactive wastes arising from the nuclear power programme must be managed so as to provide protection for people and the environment to internationally accepted standards, both now and in the future;

b) Organisational responsibilities should be clearly defined, and the relevant management structures will be put in place before wastes are generated from the nuclear power programme;
c) The NRWMS must ensure that the country meets and will continue to meet its international obligations with respect to the safe and secure management of radioactive and fissile materials;
d) Funding provisions for the management of radioactive wastes should be put in place at the start of nuclear power generation so that implementation of all components of the NRWMS is assured;
e) The financial liability for managing radioactive wastes lies with the waste producers up to the point at which disposal facilities are removed from institutional control, after which responsibility lies with the government;
f) Safe and secure facilities for managing all radioactive wastes should be made available in a timely manner with no unnecessary delays in implementation of final disposal of materials;
g) The NWRMS should be flexible to accommodate possible future changes in the nuclear power programme and fuel cycle policy;
h) The NRWMS should be implemented in an open and transparent fashion and subject to thorough regulation and national and international review and assessment.

(3) The ultimate responsibility for ensuring that the NRWMS is carried through will lie with the government. A national WMO should be established and delegated the task of implementation of the NRWMS. The WMO should be answerable to the relevant government Ministry for progress with the NRWMS and its activities will be regulated by the national regulatory authority;
(4) It is prudent for newcomer nuclear programmes to pursue a dual-track approach to radioactive waste management, whereby timely national solutions are pursued and developed in line with the Principles in parallel with seeking possible regional solutions for sharing of activities and facilities;
(5) Regular reports on progress of the NRWMS should be prepared by the WMO and submitted to the regulator for scrutiny. The report, along with the regulators' commentary, should be submitted to the government for approval;
(6) The NRWMS can best be developed and presented in the form of a **road-map** that includes scenarios, options, decision-points and time schedules for all the spent fuel and RWM facilities being considered.

8. CONCLUSIONS

For any newcomer NP country, establishing a spent fuel and RWM programme and infrastructure is critical from the viewpoint of both practicality and credibility. It also helps a new programme to develop strong international relationships with like-minded nations who wish to see spent fuel and radioactive waste managed safely and securely. Multiple options need to be considered in devising a strategy, many having potentially far-reaching national and international policy implications. Some options need to balance sensible policy decisions against strong external commercial drivers that will arise from nuclear technology and material suppliers.

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SESSION 2

THE STATUS OF SPENT FUEL MANAGEMENT AND THE PROCESS OF BUILDING PUBLIC OPINION IN KOREA

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Abstract

This study examined the current status of spent fuel management in Korea and the process of building public opinion that is currently underway to determine the final national policy for spent fuel. Spent fuels generated in the PWR plants are stored in a spent fuel pool for each unit. To date, almost all PWR plants continue to implement measures to address the lack of storage capacity such as installing storage racks additionally, installing high-density storage racks or transferring spent fuels to the spent fuel pool of other neighboring units. Spent fuels generated in PHWRs which are Wolsong Units are stored in spent fuel pools for longer than six years and then transferred to the dry storage facility on site. The Nuclear Energy Promotion Commission decided on the "Execution Plan for Spent Fuel Management Plan" in the 2nd meeting held on 20 November 2012 and setting a direction for developing a spent fuel management plan as follows: (1) put safety first; (2) develop short-, mid- and long-term management plans, and (3) prepare supporting measures which can be accepted by the general public to ease the burden borne by future generations and local residents. At the same time, it was decided to form and run a public engagement commission so as to develop a management plan while securing the highest possible level of acceptance. On 30 October 2013, the Public Engagement Commission on Spent Nuclear Fuel Management was lunched and embarked on public engagement activities with an aim of presenting recommendations on the method for managing spent fuel to the government by late 2014. Therefore, a national policy, strategy and management plan for spent fuel will be presented in detail in the Basic Plan for Radioactive Waste Management where the results of above-mentioned public engagement activities will be incorporated.

1. INTRODUCTION

The 23 nuclear reactors in Korea produce an annual 750 tons of spent nuclear fuel, and 13 806 tons are being stored in the temporary storage facilities of each reactor. This averages to 72% of the storage capacity as of late 2014. These temporary storage facilities, beginning with Kori Nuclear Power Plant, are expected to reach saturation beginning in 2016. Even if high-density storage racks are replaced and storage capacity is expanded by transferring spent nuclear fuel to new reactors, storage will saturate beginning with the Hanbit Nuclear Power Plant in 2024.

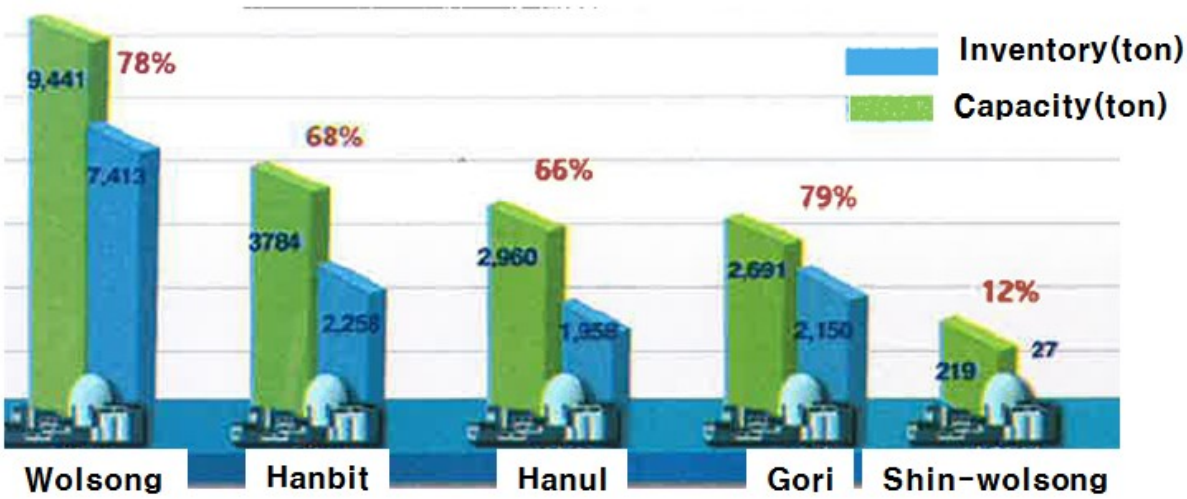

FIG. 1.Status of SNF generation in Korea NPP.

2. STATUS OF SPENT FUEL MANAGEMENT IN KOREA

2.1. Effort to solve the shortage of storage capacity

Spent fuels generated in the PWR plants are stored in a spent fuel pool for each unit. To date, almost all PWR plants continue to implement measures to address the lack of storage capacity such as installing storage racks additionally, installing high-density storage racks or transferring spent fuels to the spent fuel pool of other neighboring units. Spent fuels generated in PHWRs which are Wolsong Units are stored in spent fuel pools for longer than six years and then transferred to the dry storage facility on site.

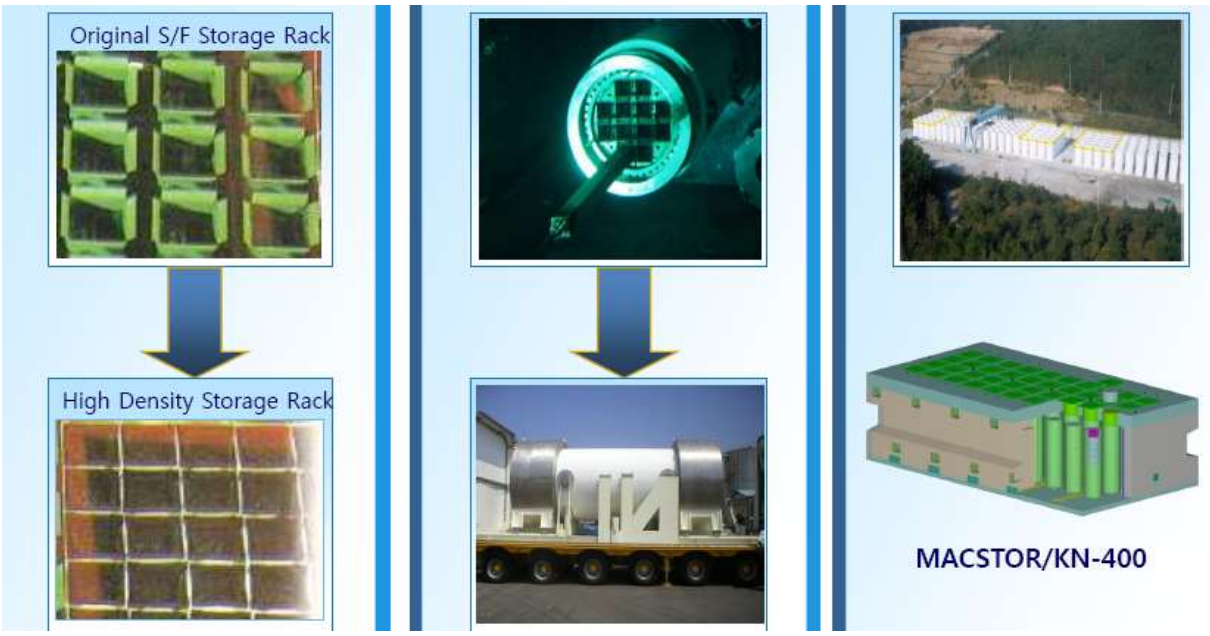

FIG. 2. Effort to solve the shortage of storage capacity.

2.2. Process of building public opinion

The launch of the Public Engagement Commission on Spent Nuclear Fuel Management on 30 October 2013 initiated national-level discussions on the management of spent nuclear fuel. Established by the Ministry of Trade, Industry and Energy, the Public Engagement Commission exhibits the characteristics of a civilian advisory council. Working independently of the government, it gathers public opinions on the management of spent nuclear fuel and makes recommendations based on the results. With reference to recommendations by the Public Engagement Commission, the government plans to establish management policies near the end of 2015. The Public Engagement Commission was originally scheduled to remain active until December 2014, but this period was extended for six months to June 2015.

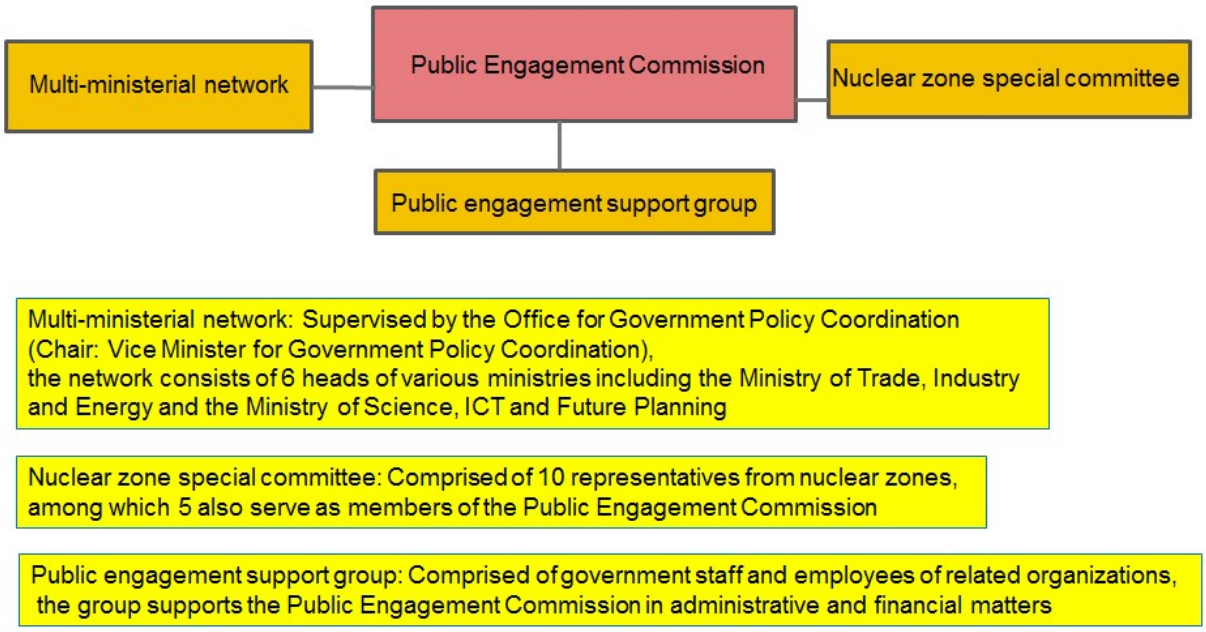

FIG. 3. Organization of PEC.

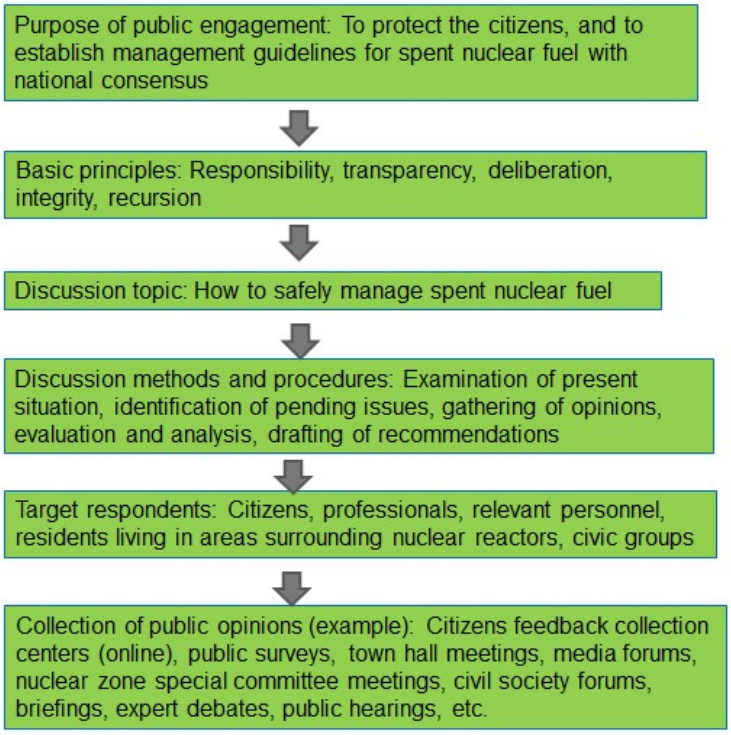

FIG.4. Strategy of public engagement.

2.3. Intermediate results

Based on intermediate results, the following agendas have been derived. The most important principle behind management policies for spent nuclear fuel is the safety of citizens. Spent nuclear fuel must be managed in a safe, responsible and effective manner. Safety must be technically verified, so as not to place an excessive burden on future generations. The purpose of management policies must be specified, along with a timeline for each goal. This process requires permanent disposal as well as storage before permanent disposal. The management policies must consider the amount of spent nuclear fuel being generated. If the expected year of saturation is postponed due to the transfer of spent fuel to spent fuel pools of other neighboring units and installation of high-density storage racks, the safety of storage facilities and soundness of the spent nuclear fuel should be verified. Policies on spent nuclear fuel must include plans and technical solutions from the point of generation to permanent disposal. Storage facilities may be positioned within or outside nuclear reactors, and storage can be either wet or dry. In consideration of Korea's circumstances, permanent disposal facilities should be constructed and operated with the goal of 2055. However, areas affected by storage and temporary disposal facilities must receive a certain amount of support. The management policies for spent nuclear fuel must also provide information on the necessary research, technology development, verification, and supervising personnel. The details to be included for each stage are supervising personnel, the scope of supervision, cost, financing, education, and training plans. If required, relevant laws and policies should be modified or newly established for policy goals to be met.

3. CONCLUSION

Spent fuels generated in the PWR plants are stored in a spent fuel pool for each unit. To date, almost all PWR plants continue to implement measures to address the lack of storage capacity

such as installing storage racks additionally, installing high-density storage racks or transferring spent fuels to the spent fuel pool of other neighboring units. Spent fuels generated in PHWRs which are Wolsong Units are stored in spent fuel pools for longer than six years and then transferred to the dry storage facility on site. Through the Public Engagement Commission, Korea plans to derive recommendations for the management of spent nuclear fuel based on public feedback, and to subsequently establish management policies. Based on intermediate results, several agendas have been derived. In consideration of Korea's circumstances, permanent disposal facilities should be constructed and operated with the goal of 2055. This study is expected to transfer experience of and lessons from determining national policy to countries that have not established their final management policy for spent fuel, based on Korea's solutions to shortage of spent fuel pools and the formation of national consensus.

ACKNOWLEDGEMENTS

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SESSION 3

LINKING NUCLEAR SECURITY TO SPENT FUEL MANAGEMENT

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MCNP MODEL FOR RESEARCH REACTOR (TRR-1/M1) FUEL STORAGE RACK CRITICALITY ANALYSIS

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Abstract

The spent fuel storage of TRIGA reactor (TRR-1/M1) was designed. Detailed criticality safety analysis was conducted according to the basic safety standard and within the framework of the regulatory aspect. The Monte Carlo computer code (MCNP) was used for the detailed geometry modelling. Two types of fuel elements were considered for the calculation including 8.5 wt% and 20 wt% fuel elements. The critical number of the fuel elements was determined. The dependence of multiplication factor (k_{eff}) on the distance between the fuel elements in the rack (pitch) and the water density which varied from 0.01 g/cm^3 to 1 g/cm^3 were calculated. The MCNP results shown that the k_{eff} strongly depend on the lattice pitch, moderation effect and the water density. At present, there is limited experience in the safety management of spent fuel for research reactor. The strategy development and implementation of the integrated approach to management of the spent fuel from research reactor are important for the long term strategy of the ageing management program.

1. INTRODUCTION

The criticality safety analysis has been conducted for power reactor fuel due to the problem from the fuel pool storage capacities. Two methods were considered and implemented including the fuel rack and pool design and fuel burnup reduction. For the research reactor spent fuel pool nearly full storage capacities is not a problem. However, considering the fuel burnup and criticality analysis will be beneficial for determining the risk and reducing the cost for the fuel rack while storage spent fuel according to the safety margin and subcriticality guarantee for the spent fuel storage pool of research reactor TRR-1/M1. Different study cases of pitch decrease, and the burnup influenced was studied systematically. The criticality study analyses the spent fuel pool for the effect of burned fuel elements and pitch reduction. The standard TRIGA fuel elements were used including 8.5 wt% and 20 wt% enriched UZrH fuel. The number of the cadmium absorber rods was investigated.

1.1. Description of the Fuel Rack and the Reactor Pool

The geometry model for the fuel rack was based on the rack in the reactor pool. The reactor pool is rectangular in shape and is divided into two sections by means of a removable watertight aluminium gate. The main experimental section is 3.2 m long, 3.54 m wide and 8.5 m deep referring as ‘small pool’. The second section which located the spent fuel rack is 5.52 m long, 3.54 m wide and 8.5 m deep referring as ‘large pool’. The pool wall consists of 2 concrete layers of different densities; the inner layer made of conventional concrete and the outer layer made of high density concrete for biological shielding. The thickness of conventional concrete layer is approximately 0.46 m and the high density concrete layer begins at about 2 m from the top of the pool wall and increases stepwise until its thickness is 0.915 m making 1.375 m .as the overall thickness of both layers at the bottom. The inside pool walls and the pool floor are painted with epoxy coating which is capable of withstanding high radiation in order to protect the concrete surface from the action of demineralized water.

1.2. Description of the Fuel and Fuel Rack Models

The 90 elements fuel storage rack geometry in vertical and top view using MCNP code [1] is illustrated in Fig. 1. Dimension of the fuel rack are 67.5 cm wide by 91.5 cm long and 61 cm high (approximately 26.6 inches wide by 36 inches long by 24 inches high). Lifting points are

located at each corner of the top. The aluminium fuel storage rack is attached to the bottom of the large reactor pool consisting of top and bottom support plates. Holes are arranged in a square array with 9 cm pitch. Top support plate has 9 × 10 holes for inserting fuel elements. In the model, the pitch was varied from 9 cm to 3.75 cm, simulating the compaction of fuel elements from the rack design pitch down to contact pitch. The geometry specification for each fuel element includes the central zirconium rod, the fuel meat and the cladding as well as the top and bottom graphite reflectors and end pieces (Fig. 2).

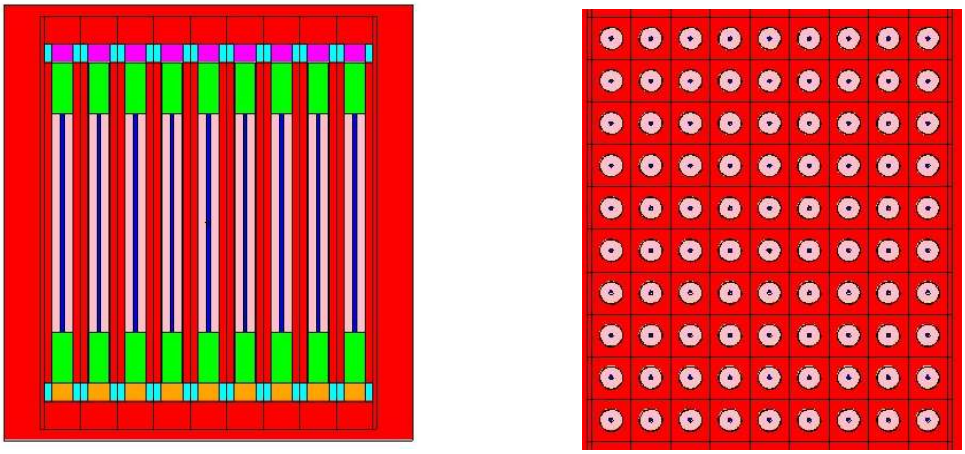

FIG. 1. Fuel storage rack vertical and top view using MCNP model.

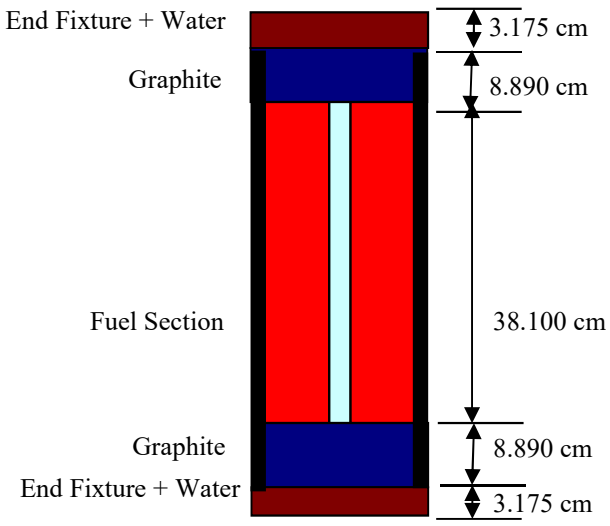

FIG. 2. TRR-1/M1 fuel element lattice model (vertical cross section).

2. CALCULATION METHODS

The Monte Carlo computer code MCNP5 [1] was used. Pointwise cross section data evaluated from ENDF/B-VI data [2] was applied. The scattering function for graphite, hydrogen in water molecule and hydrogen in zirconium were taken from the ENDF/B-IV library. The fuel elements, the absorber rods and the storage rack were modelled and taken into account for the axial and radial leakage.

2.1. Pitch calculation

Two calculations were studies for 8.5 wt% and 20 wt% fuel elements. The spent fuel rack was filled with 10% burnup of 8.5 wt% and 20 wt% fuel elements for square lattice pitch. The calculation determined the effective multiplication factor k_{eff} for the storage rack with different pitches from 9 cm down to 3.75 cm contact pitch.

2.2. Fuel burnup calculation

The composition of fission products was calculated from COREBN which is an auxiliary code of the SRAC system [3] for multi-dimensional core burn-up calculation based on the diffusion theory and interpolation of macroscopic cross-sections tabulated to local parameters such as burn-up degree, moderator temperature. The macroscopic cross-sections are prepared by cell burn-up calculations with the collision probability method of SRAC. The burnup values were calculated from specific power of 1.2 MW and time of burning. Stable and long lived nuclides are accumulated in burned fuel, so that the concentration with burnup and time increased. The criticality analysis was determined by calculating the k_{eff} for the spent fuels with the fuel burnups from 10%, 15%, 20%, 25%, and 30%. The pitch was decreased from the rack design pitch (9 cm) down to contact (3.75 cm) in a square arrangement of the fuel.

2.3. Water density calculation

The spent fuel rack design is to have an over-moderated array. However, any situation which reduces the amount of moderator could lead to more critical configuration. Two situations were possible including (1) air or steam voids in the rack from pneumatic equipment or boiling (2) reducing the amount of water covering a rack. The full 90 elements rack with 15% burnup were study by varied the density of water from 0.01 to 1 g/cm^3. Without any active cooling, boiling is not a credible mechanism for production voids within the spent fuel rack for TRR-1/M1.

2.4. Absorber rods calculation

For this study, the spent fuel rack was filled with different burnup of 20 wt% fuel elements. Several rack positions were filled with absorber rods. The efficiency of the absorber rods made of cadmium was studied. Considered the thickness of absorbing layer of absorber rods, the results show that k_{eff} depended on the thickness of the cadmium layer. The calculation was performed for cadmium thickness between 0.0001 mm to 20 mm [4]. Results show that the influences of the cadmium thickness on k_{eff} were insignificant for cadmium layers greater than 1 mm as shown in Fig. 3. Therefore, cadmium layer which is considered for the absorber rods calculation used in the MCNP model was 7 mm. thick. The final model used was with top and bottom plugs made of stainless steel and 7 mm thick cadmium plate, rolled in the hollow cylinder.

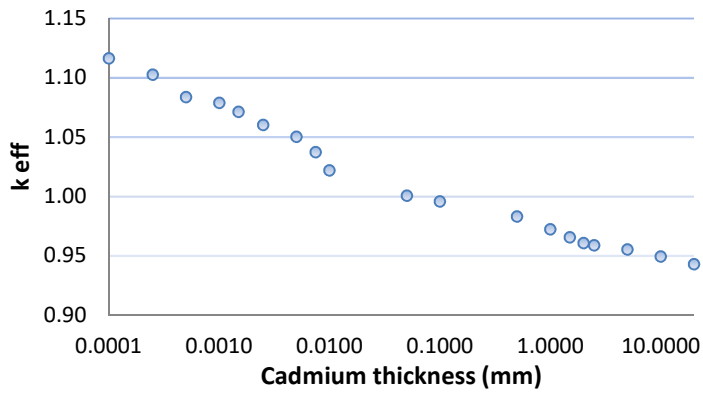

FIG. 3. K_{eff} for TRR-1/M1 with cadmium rods versus the thickness of cadmium layer.

3. ANALYSIS AND RESULTS

3.1. Critical analysis of different burned fuel (10%, 15%, 20%, 25% and 30%) with decreased pitches

The spent fuel rack was filled with several burnups of 8.5 wt% and 20 wt%. The k_{eff} was determined with different pitches from 9 cm down to square contact pitch (3.75 cm). Results for 90 fuel elements occupying all positions are shown in Fig. 4 for 8.5 wt% and Fig. 6 for 20 wt%. It can be observed that k_{eff} strongly depends on the lattice pitches in both 8.5 wt% and 20 wt% fuels. For 10% burnup of 8.5 wt%, the k_{eff} at the contact pitch (3.75 cm) is 1.00953 ± 0.00022 and for 20 wt% k_{eff} is 1.07526 ± 0.00033. Without the additional absorbers, it may become supercritical if the distance is at the contact pitch. The plot from Fig. 4 and Fig. 5 are relevant in the sensitivity to pitch variation for TRR-1/M1 storage rack model. The spent fuel for 20 wt. % may become critical for all burnup if the lattice pitch of the rack is below 4.23 cm. The results for spent fuel rack without absorber rods at pitch 4.23 cm, the k_{eff} for 10% burnup of 8.5 wt. % is 0.99091 ± 0.00032 and k_{eff} for 30% burnup of 20 wt% is 0.99847 ± 0.00042.

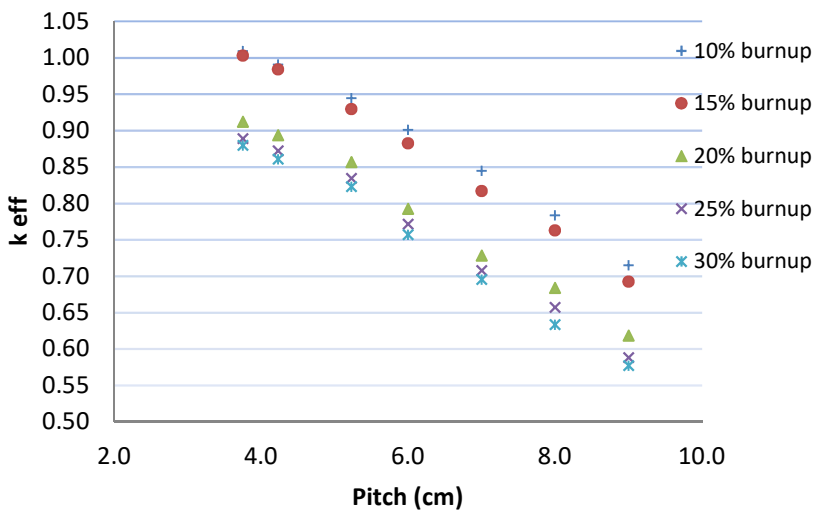

FIG. 4. K_{eff} for TRR-1/M1 as a function of lattice pitch for different burnup of 8.5 wt% fuel.

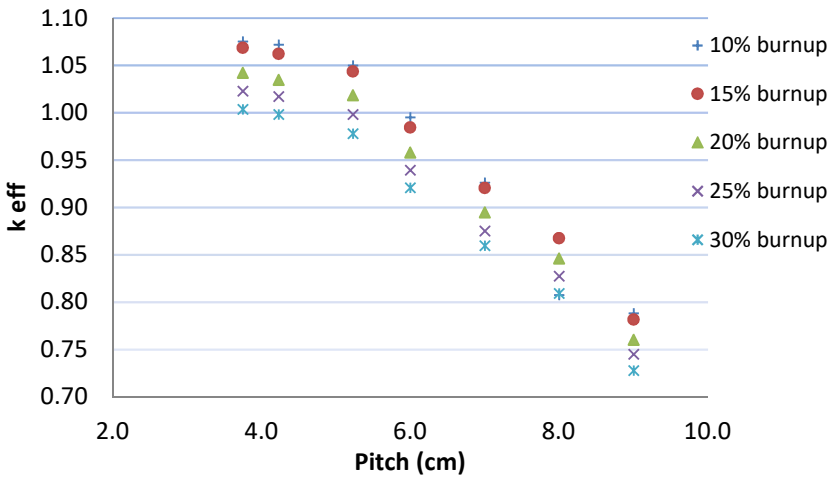

FIG. 5. K_{eff} for TRR-1/M1 as a function of lattice pitch for different burnup of 20 wt% fuel.

3.2. Critical analysis for water density variation of 15% burnup of 20 wt% with 7 cm pitches

Fig. 6 shows the result of k_{eff} calculation for uniformly varying the density of water moderation within the array of storage racks from 0.01 to 1 gram per cubic centimetre (g/cm^3). The results shown that the k_{eff} increases as the water density is decreased as is expected from the over moderated design. In case of 7 cm pitch, the lattice is over-moderated and the effect of k_{eff} is positive if water density is decreased. At low water density the effect of leakage prevails over the effect of better moderator to fuel ratio. The optimal moderation effect is observed at approximately 0.4 g/cm3 water density. The k_{eff} is 0.99728 ± 0.00032 for 15% burnup of 20 wt. % and k_{eff} is 0.81728 ± 0.00026 for 15% burnup of 8.5 wt. % for the water density 0.40 g/cm^3 for 15% burnup of 20 wt% with 7 cm pitches. The normal pool temperature is 30 - 35°C, the water density 0.3 g/cm^3 and the boiling is not possible at normal pool temperature.

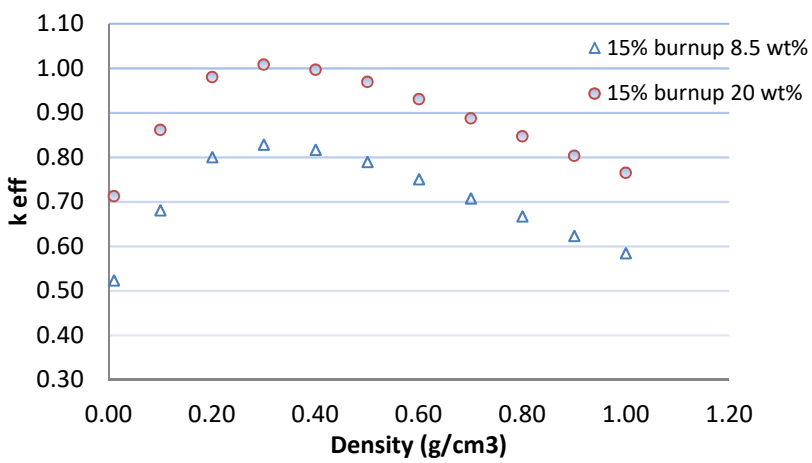

FIG. 6. K_{eff} as a function of water density for 15% burnup of 8.5 wt% and 20 wt% fuel.

3.3. Critical analysis for absorber rods calculation of different burnup of 20 wt% with different pitches

The pool being loaded with fresh fuel was critical at 3.75 cm pitches for 8.5 wt% and for 20 wt%. The results show that the storage rack with 12 cadmium absorber rods remain subcritical for all burnup at 5.23 cm pitches. The value of k_{eff} for 10%, 15%, 20%, 25%, and 30% burnup at 5.23 cm pitch are 0.99552 ± 0.00021, 0.94802 ± 0.00019, 0.84302 ± 0.00029, 0.79622 ± 0.00025, 0.74212 ± 0.00024, respectively. Fig. 7 presents the k_{eff} calculation for 20 wt% fuel with 12 cadmium absorber rods for different pitches and burnup level.

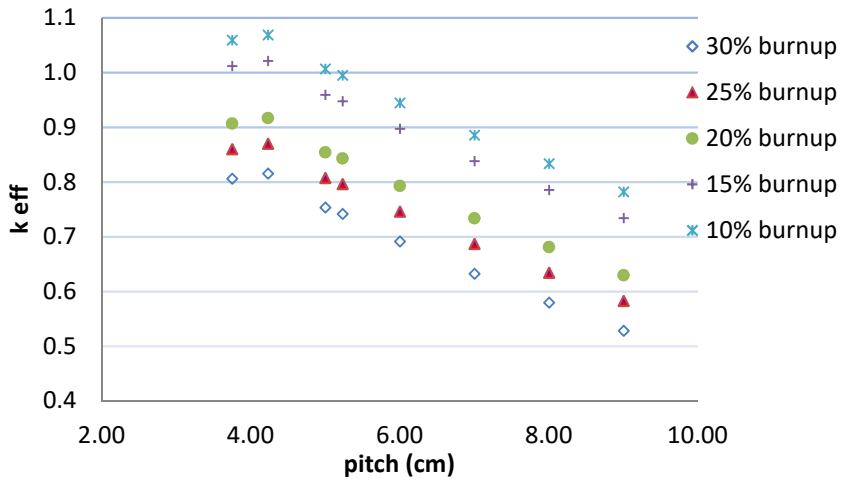

FIG. 7. K_{eff} as a function of pitch for different burnup of 20 wt% fuel with 12 Cd absorber rods.

4. CONCLUSION

Criticality safety for the storage rack of TRR-1/M1 depends on several parameters including pitch, burnup, water density and number of absorber rods. Each physical parameter affects the nuclear reactivity of the fuel storage rack design and construction. It can be concluded that burnup affects the multiplication factor (k_{eff}) of spent fuel storage rack less than the pitch, the water density and the number of absorber rods. The sub criticality is not guaranteed for the spent fuel rack during the postulated accident if the rack disintegration with lattice compaction and the fuel elements pile together. Replacing some fuel elements in the storage rack by cadmium absorber rods can decrease the probability for super criticality to an acceptable level. In order to reduce the risk of criticality, the rack should design for minimum lattice pitch more than 6 cm for 20 wt% fresh fuel or cadmium absorber rods should be used. Administrative controls are necessary for loading, moving and handling the spent fuel storage rack as well as the operational control. The administrative controls or limitations must be incorporated into any operating or handling procedures for the storage rack of TRR-1/M1. The criticality safety analysis according to standard for TRIGA spent fuel storage is necessary and feasible. Existing power reactor standard are not applicable because of the specific TRIGA fuel physical properties. Specific research reactor standards, recommendation, guideline and procedures for TRIGA reactor spent fuel are required and necessary.

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MAJOR CONSIDERATION FOR SAFETY ASSESSMENT FOR INTERIM STORAGE FACILITY ON BEYOND DBA IN KOREA

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1. INTRODUCTION

At present, Korea is operating nineteen PWR reactors and four PHWR reactors in total. The current storage capacities will become insufficient for the storage of the spent fuels discharged from the year 2024 and thereafter. According to nuclear policy decision by the Korean government in 2012, the spent fuel shall be stored within the premises of respective nuclear power station until 2024 by expanding its on-site storage facility.

KHNP (Korea Hydro & Nuclear Power Co., Ltd), that owns and operates all nuclear power plants in Korea, is performing the expansion project of spent fuel storage capacity in a reactor site by application of high density storage rack, concrete silo-type and consolidated dry storage facility (MACSTOR/KN400), and its trans-shipment to the adjacent reactor facility.

To save time, the Korean government recently draws up a plan to complete the construction of interim storage facility by 2024 for spent fuel management. In addition, the interim storage facility should satisfy conservative design criteria, focusing on safety for long-term storage integrity and durability against potential attacks from the most probable enemy and/or terrorism.

This paper aims to establish a standard scenario for interim storage of spent fuel in Korea and review several considerations for Safety Assessment for Interim Storage Facility on Beyond DBA in Korea.

2. INFLUENCE ANALISYS OF SFENT FUEL STORAGE SYSTEM BASED ON FUKUSHIMA NPP

The Fukushima accident has resulted in more stringent requirements for nuclear facilities in case of beyond design basis accidents. Therefore, there has been growing demand for developing scenario on interim storage facility to prepare for beyond design basis accidents and conducting dose assessment based on the scenario to verify the safety of each type of storage.

2.1. The influence outline on each type of Fukushima NPP storage facility

The cause of Fukushima nuclear power plant damage can be divided into two groups: the earthquake of 9.0 magnitude and massive tsunami. Especially, the second 15-meter-high tsunami out of a series of seven tsunamis caused the gravest damage to NPPs.

The floating debris from tsunami destroyed all intake structures, leading to a loss of ultimate heat sink. Also, the seawater flow into the building destroyed the electricity supply system and cooling system. Consequently, station blackout (SBO) occurred in Fukushima NPP units 1 to 4 while only one emergency diesel generator (EDG) could be used to cool the reactor and spent fuel pool in the NPP units 5 and 6.

The earthquake and tsunami which triggered the Fukushima accident were beyond design basis accidents. Therefore, the structural integrity and cooling function of spent fuel pool, all of which were aimed at preparing for beyond design basis accident, emerged as a big issue in global nuclear power industry in the wake of Fukushima accident. Fukushima NPP has 3 types of spent fuel storage: spent fuel pool inside reactor building, spent fuel pool for common use, and on-site dry type spent fuel storage facility. Namely, we can compare and analyze the safety of each type of spent fuel storage based on Fukushima accident.

Spent fuel pool inside containment of Fukushima NPP lost its cooling function since it was greatly affected by the tsunami and earthquake. As the spent fuel pool inside reactor building was located on the upper side of building, people raised the possibility of the massive release of cooling water through the cracks of spent fuel pool structure. Some also concerned about another possibility of lowering level of cooling water since loss of cooling function might lead to evaporation of cooling water containing radioactive materials. As a number of fuel rods might be exposed in both cases, safety of spent fuel pool inside reactor drew urgent attention.

The spent fuel pool for common use contained 60% (6375 assemblies) of total spent fuel assemblies and about 80% of storage capacity was filled with spent fuel. Nonetheless, there has not been much concern for the spent fuel pool for common use because the decay heat of spent fuel in the pool was very low and the pool was barely affected by earthquakes. In addition, although 9 on-site dry type spent fuel storage facilities contained 408 spent fuel assemblies, they reportedly maintained their structural integrity better than two other types of storages even in the middle of rubble.

This is mainly because the cask for dry type storage facility was designed to assure its safety by carrying out simulation tests such as drop test, bursting test, flooding test, and fire test to an extreme level. The global trend in designing dry type storage facility after the 9/11 terrorist attack in the U.S. is to design a storage facility which can even withstand an aircraft engine crash. Likewise, the designing process of storage facility underscores the importance of safety as well.

Studying the influence of Fukushima accident on spent fuel facilities, we realized that the radionuclides leakage of on-site spent fuel pool has drawn grave concern. In this regard, this research developed accident scenario for performing the radionuclides leakage assessment in follow-up study in case of the wet-type spent fuel storage facility of spent nuclear fuel

2.2. Analysis on each type of Fukushima NPP storage system

This study assumes that the type of interim storage facility is wet storage and its cooling system is out of order because of the beyond design base accident (Beyond DBA). The research also assumes that the vapor containing radionuclides is spread out during the period when the cooling system is out of order. In reality, the wet storage cooling and cooling water supplementation systems in Fukushima reactors did not work properly because of the loss of alternating current power source. It took 20 to even 150 days to restore the cooling system depending on the damage of each storage facility. Particularly, in unit 4, all the fuel rods in the core had been transferred to on-site wet storage for the overhaul of the reactor. Therefore, there were many spent fuel assembly in on-site wet storage in unit 4 compared to other on-site spent fuel wet storage, the spent fuels leaded to a relatively heavy heat load.

Therefore, the scenario assumes that the radioactive cooling water evaporated and diffused for 140 days during which the Fukushima NPP unit 4 was in recovery.

TABLE 1. RECOVERY PERIOD FOR COOLING SYSTEMS IN ON-SITE STORAGE FACILITIES IN FUKUSHIMA NPP SITE

Unit	Recovery completion date	Recovery period (day)
1	2011.08.10	150
2	2011.03.11	20
3	2011.06.30	110
4	2011.07.31	140

According to the report released by Tokyo Electric Power, the radionuclide analysis of a water sample taken from the unit 4 spent fuel pool on 28 April 2011 detected levels of ^{137}Cs of 55 Bq/cm^3, indicating that spent fuel rods were damaged because of Fukushima accident. This figure could be a base data in assuming the spent fuel damage rate for developing accident scenario after beyond design basis accident took place. In this study, it is assumed that the spent fuel damage rate determined is based on the inventory of spent fuel documented in Yucca Mountain Safety Evaluation Report [2] and the amount of ^{137}Cs detected in the water sample. The main described above are presented in Table 2 as follows:

TABLE 2. MAIN ASSUMPTION OF ACCIDENT SCENARIO

	Assumption	Basis data
Method and type	Wet-type	-
Accident consequence	Loss of cooling system	-
Inventory	PWR spent fuel inventory	Yucca Mountain repository SAR
Diffusion period	140 days	Recovery period of Fukushima Unit 4
Damage rate	5.6E-6 %	Leakage rate of Fukushima Unit 4

3. PRECONDITION FOR SAFETY ASSEMENT

3.1. Precondition for safety assessment

There are two types of interim storage method: namely, wet-type interim storage facility and dry-type interim storage facility. Globally, the ratio of dry-type interim facility is consistently increasing. US-EPRI predicted that wet-type interim storage facility would disappear in a few decades. Dry-type interim storage facility is an effective management method in preparing the future need for the excessive temporary storage facility and adopted by many nuclear developed countries.

This paper selected the centralized interim storage facility for interim SNF (Spent Nuclear Fuel) storage method which is efficient in terms of SNF management. PWR adopts dual purpose canisters (transport/storage) or concrete canisters, all of which have been proven safe. PHWR adopts MACSTOR/KN-400 which is currently being used. In Korea, Concrete Silo is

also adopted for temporary storage of SNF. However, MACSTOR/KN-400 is more likely to be adopted due to its advantage in storage density.

The Korean government established and declared the 5th Basic Plan for Electricity Supply and Demand for stable power supply. In its plan, 47 117 MTU (PWR: 35 247 MTU, PHWR: 11 870 MTU) of SNF is assumed to be discharged from 34 nuclear power reactors by 2123. Expected amount of SNF discharged from the nuclear power plants was calculated by extrapolation from the current amount of SNF in Korea as below.

TABLE 3. EXPECTED AMOUNT OF SPENT NUCLEAR FUEL IN KOREA

Type	2004[a] (MTU)	2014[b] (MTU)
PWR	12 000	27 677
PHWR	8 000	11 870
Total	20 000	39 547

[a] 1st Basic Plan for Electricity Supply and Demand, 2004
[b] 6th Basic Plan for Electricity Supply and Demand, 2014

Preconditions and contents for drawing up SNF management scenario aiming at producing a standard scenario therefrom are stipulated as follows:

TABLE 4. PRE-CONDITIONS AND CONTENTS

Classification	Contents	Remarks
Number of nuclear power units	Total: 34 (PWR:30, PHWR:4)	The 6th Basic Plan for Electricity Supply and Demand, 2014
Operation period (years)	PHWR: 30 OPR: 40 APR, APR+: 60	-
Interim storage facility	Start year of operation: 2024 (Interim storage of PHWR SNF first)	The centralized interim storage facility adopted
Disposal	PHWR disposal period: 2050~2074 PWR disposal period: 2075~2124	Total operation period: 75 years

The standard scenario was assumed that the operation of interim storage facility will start from 2024, while that of disposal facility from 2050. As to the adopting centralized interim storage facility, assumption was such that the interim storage of PHWR SNF comes first for its effective management. 300 concrete silos and 12 MACSTOR/KN-400 have been operational at Wolsung site (PHWR) outside fuel building, after having been stored at a wet storage facility. Therefore, we set the disposal year of PHWR SNF to be 2050, given the license terms and expiration date of respective PHWR.

3.2. Major considerations for safety assessment

Korea has to design the spent fuel storage facility considering aircraft crash accident as Beyond DBA. In addition, long term storage issues are very important to design the cask and facility for storage of spent fuel. Korea government have wait & see policy for spent fuel management so that have to plan to store the spent fuel for 100 years or the more because difficulty on decision of final deposal policy. In order to earn more time to get public acceptance, it is necessary to store spent fuel for a long term about 100 years or the more.

The results of sensibility assessments show that the diffusion radius of nuclide material and the period of diffusion are dominant factors. In addition, we need objective data to accurately measure the radionuclide distribution in beyond design base accident (Beyond DBA) scenario. Especially, exposure dose rate varies according to PWR spent fuel inventory and damage rate of spent fuel.

Korea is obliged to have a safety design of SNF storage facility to cope with the plausible military attacks from a hostile and radical enemy in consideration of our unique situation. To be specific, the site to be chosen must be naturally guarded by the mountainous shield so as to be free from the potential enemy's direct attack.

The construction cost of interim storage facility would vary greatly depending on which dry-type method we choose. Cost evaluation would be relatively poor if we adopt structural interim storage facility like Japan and Sweden. These countries put other aspects including safety, security, and residents' acceptance over economic feasibility. The Korean government also has to choose interim storage facility method in considerations of more importance factors such as safety, security, and public acceptance.

4. CONCLUSION

This paper could develop accident scenario for wet-type interim spent fuel storage facility by studying the influence of Fukushima accident on spent fuel facilities. Focusing on Fukushima accident was an appropriate case study to develop the accident scenario because the accident was beyond design-basis accident and three different types of spent nuclear fuel storage facilities existed at Fukushima site.

Since the method of spent fuel interim storage has not been decided yet, we should develop additional scenario to assess another kind of radionuclide leakage. For example, we need to develop scenarios on radionuclide leakage resulting from block of air inflow tract aimed at cooling the canister and damage of the integrity of dry storage facility caused by the long period of the storage.

Operation period of interim storage facility was usually 40~50 years in the past. From the global perspective, however, the period of interim storage has gradually extended to long term storage of at least more than 100 years. It turns out that social and political determinations are very important in every step. It also reflects the fact that a lot of time is required to decide on spent fuel management policy such as site selection and stakeholders' acceptance.

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CRITICALITY SAFETY AND BURNUP CREDIT ANALYSIS FOR MOX FUEL

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Abstract

MCNPX Monte Carlo Computer Code was used to design and simulate the burn up of MOX assembly in a typical operating conditions of PWR reactor. MOX assembly is loaded with three different types of Pu isotopes, low, medium and high Pu content. The fuel burnt up to 85 GWd/T. Typical results show the multiplication factor of the assembly, power fraction produced in each zone, evolution of Pu isotopes with burnup. Total actinides and fission product in the whole assembly are also calculated.

1. INTRODUCTION

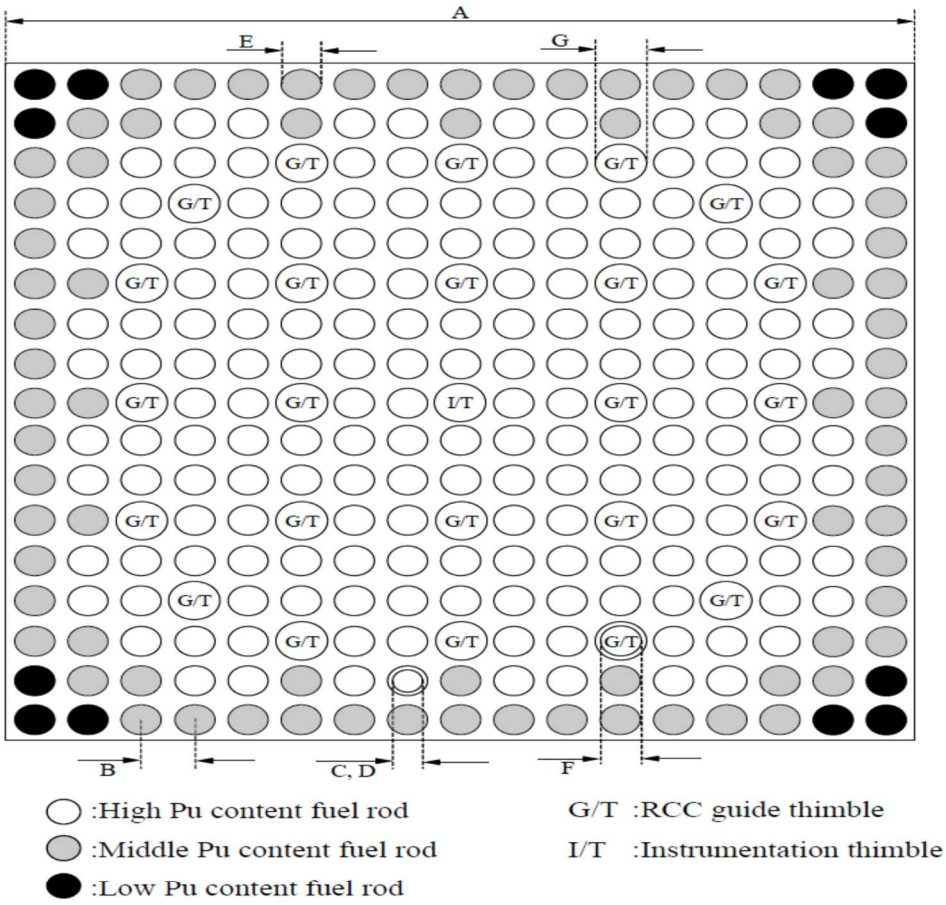
Burnup credit is defined as the consideration of the reduction in reactivityassociated with the use of the fuel in power reactors. Changes in the isotopic composition during fuel burnup which result in a reduced reactivity can be conveniently characterized by the reduction of the net fissile content, the build-up of actinides, the increase of the concentration of fission products, and the reduction of burnable absorber concentration where applicable. In practice, the conservative use of burnup credit requires consideration of all fissile isotopes, and allows consideration of any neutron absorbing isotopes for which properties and quantities are known with sufficient certainty [1−9].

The present research analyse the burn up and neutronic parameters of an assembly of MOX fuels. The discharge burnup is extended up to 85 GWd/T [1]. In the assembly benchmark problems, important parameters for in core fuel management such as local power peaking factors and reactivity coefficients were included in the analysis. A PWR MOX fuel assembly is the same geometrical configuration as 17×17 type PWR fuel design. The assembly pitch 21.505 cm, fuel rod pitch 1.265 cm, pellet outer diameter 0.824 cm, cladding outer diameter 0.952 cm. The average Pu fissile content is 11 % wt assuming 21 full effective power months operations using three batch loading strategy. The assembly is composed of low, middle and high Pu content fuel rods.

A typical compositions of low, medium and high Pu and the structure content can be found at Ref. [1]. The power density is 36.6 MW(th). The density of water coolant at hot and cold conditions are given at reference [1]. MCNPX computer code package (which is based on Monte Carlo method Ref. [10]) is used to model the assembly and a three dimensional computer model has been designed to simulate the burn up of the assembly in a typical operation conditions of PWR reactor. Horizontal Cross section of assembly Model is illustrated in Fig. 1.

The results indicate burnup dependency of multiplication factor and local peaking factor, burnup dependency of fission rate distributions, thermal flux, power distributions, and the effect of water temperature on the multiplication factor.

2. GEOMETRY AND MATERIAL SPECIFICATIONS OF THE ASSEMBLY



Abbreviations	Description	cm
A	Assembly pitch	21.505
B	Fuel rod pitch	1.265
C	Pellet outer diameter	0.824
D	Cladding inner diameter	0.824
E	Cladding outer diameter	0.952
F	I/T, G/T inner diameter	1.140
G	I/T, G/T outer diameter	1.220

Remarks:
Thickness of cladding:0.064cm
Gap between pellet and cladding:0.0cm(No gap assumed).
Dimensions of MOX fuel rod is identical to that of pin cell problem.
I/T: Instrumentation thimble, G/T: RCC guide thimble
I/T, G/T thickness:0.040cm
Gap between assemblies:0.0cm(No gap assumed.)

FIG. 1. Geometrical compositions of PWR MOX fuel assembly.

A PWR MOX fuel assembly is 17×17 type PWR fuel design. The assembly pitch 21.505 cm, fuel rod pitch 1.265 cm, pellet outer diameter 0.824 cm, cladding outer diameter 0.952 cm. The average Pu fissile content is 11 % wt assuming 21 full effective power months operations using three batch loading strategy. The assembly is composed of low, middle and high Pu content fuel rods.

A typical compositions of low, medium and high Pu and the density of water coolant can be found at Ref. [1]. The power density is 36.6 MW/TH.

3. COMPUTER AND MATHEMATICAL SIMULATION

MCNPX computer code package (which is based on Monte Carlo method reference [10]) is used to model the assembly and a three dimensional computer model has been designed to simulate the burn up of the assembly in a typical operation conditions of PWR reactor. Horizontal cross section of MCNPX assembly model is illustrated in Fig. 2.

Neutron interactions are simulated with 60 cycles (50 active cycles) each cycles contains 20000 neutrons which means that one million neutrons are used to accumulate the reactor tallies and calculate the output. During reactor operations, three fuel materials are allowed (each material represent one fuel element zone) to burnup and the new composition for each material are calculated by the computer code at every time step. The core fuel cycle (time between two refueling) are divided into 9 time steps.

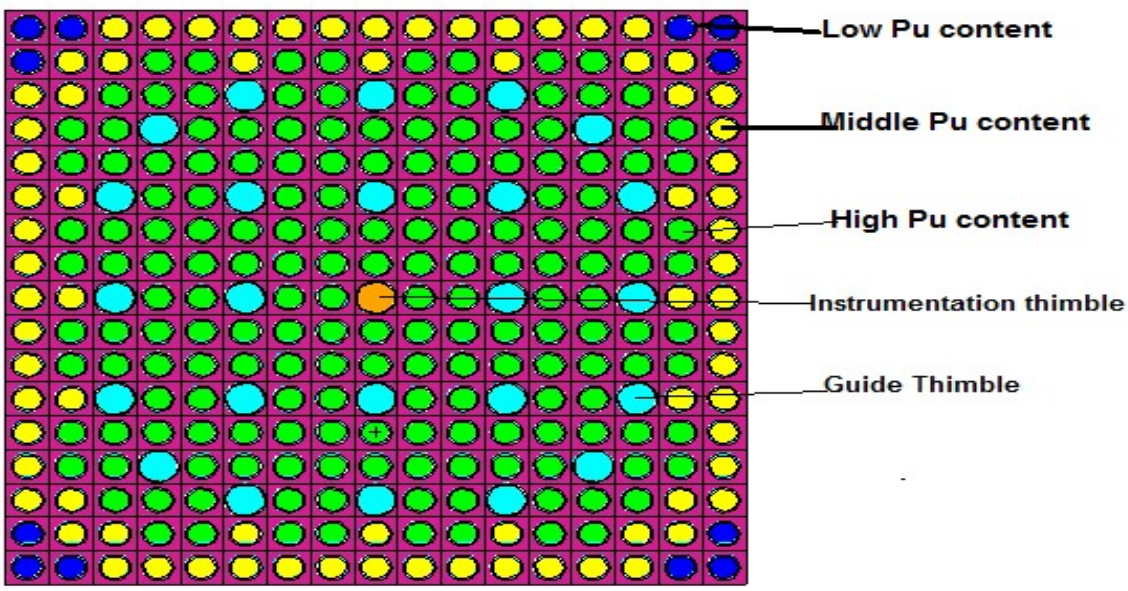


FIG. 2. MCNPX horizontal cross section of MOX assembly model.

4. RESULTS AND DISCUSSIONS

Fig. 3 illustrates the assembly multiplication factor K_{eff} versus fuel burnup (GWd/T). The results are produced at operating conditions of the fuel i.e. operating fuel, clad, coolant temperature and xenon in equilibrium. The maximum value of Keff at fresh fuel is 1.2175 and 0.997 at 85 GWd/T. The loss of reactivity is given by Eq. 1 and represents reactivity loss from fresh fuel assembly up to burnup of 85 GWd/T.

$$\Delta\rho = \frac{\Delta K}{K_1 K_2} = -0.18165 \quad (1)$$

Reflective boundary conditions are assumed around the assembly walls and free boundary on the top and bottom.

Fig. 4 illustrates the concentrations of some fissile isotopes with burnup (GWd/T) in low Pu zone as indicated at Fig. 2. The results indicates that ^{235}U reduces from fresh fuel concentrations 4.352E-5 (atom/barn.cm) to 1.585E-5 (atom/barn.cm) at 85 GWd/T (63.5 % of ^{235}U is consumed. ^{239}Pu reduces from 9.484E-4 (fresh fuel) to 6.053E-4 at 85 GWd/T i.e 36 % of ^{239}Pu is consumed . The fissil isotope ^{241}Pu increase with burn up. The fertile isotopes ^{240}Pu decreases and ^{242}Pu increases with burn up.

Fig. 5 illustrates the concentrations of some fissile isotopes with burnup (GWd/T) in medium Pu zone as indicated at Fig. 2. The results indicates that ^{235}U reduces from fresh fuel concentrations 4.021E-5 (atom/barn.cm) to 1.768E-5 (atom/barn.cm) at 85 Gwd/T. i.e 56 % of ^{235}U are consumed. ^{239}Pu reduces from 1.815E-3 (fresh fuel) to 9.806E-4 at 85 Gwd/T. (45% of ^{239} Pu is consumed). The fissil isotope ^{241}Pu increase with burn up. The fertile isotopes ^{240}Pu decreases and ^{242}Pu increases with burn up.

Fig. 6 illustrates the concentrations of some fissile isotopes with burnup (GWd/T) in high Pu zone as indicated at Fig. 2. The results indicates that ^{235}U reduces from fresh fuel concentrations 3.623E-5 (atom/barn.cm) to 1.755E-5 (atom/barn.cm) at 85 GWd/T. i.e 51.5 % of ^{235}U are consumed. ^{239}Pu reduces from 2.296E-3 (fresh fuel) to 1.212E-3 at 85 GWd/T (47.2 % of ^{239} Pu is consumed). The fissil isotope ^{241}Pu increase with burn up. The fertile isotopes ^{240}Pu decreases and ^{242}Pu increases with burn up.

Fig. 7 illustrates the concentrations of total actinides (including fissile and fertile) in the assembly and the fission product. The results indicate that the global actinides decreases with time due to neutron absorption and fission. While fission products increase with burnup and time. Some fission product have large absorption cross section with neutron such as ^{135}Xe which have absorption cross section 3.7×10^6 barns.

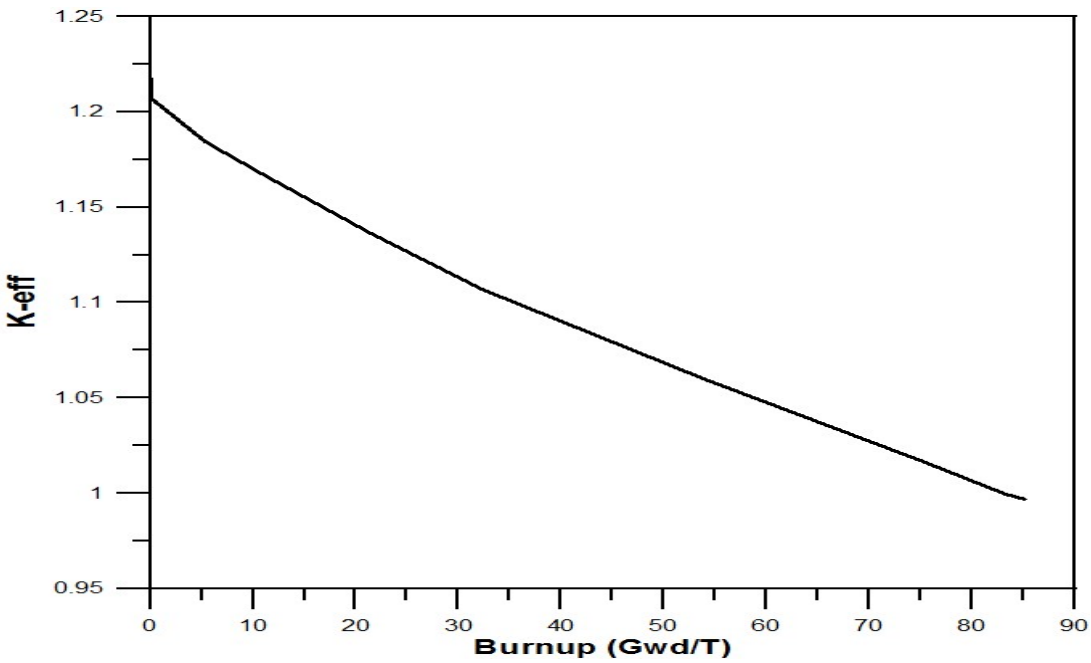


FIG. 3. K_{eff} Versus burnup of the assembly (GWd/T).

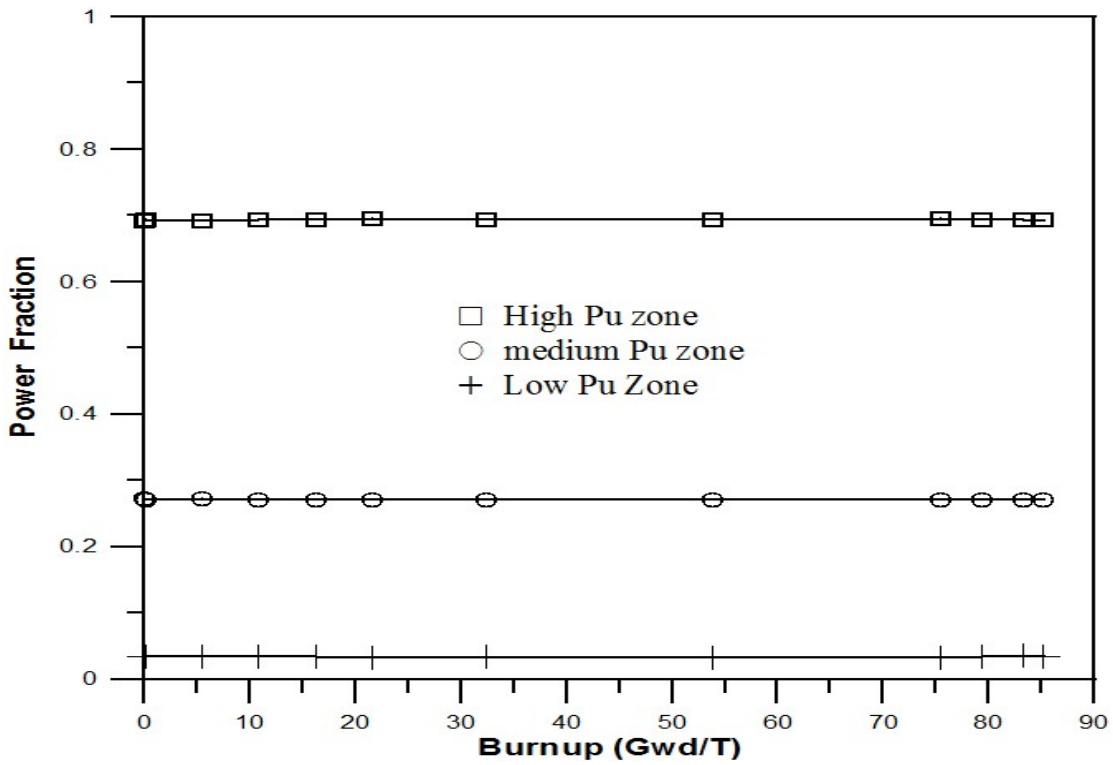


FIG. 4. Pu isotopes concentrations for low pu zone.

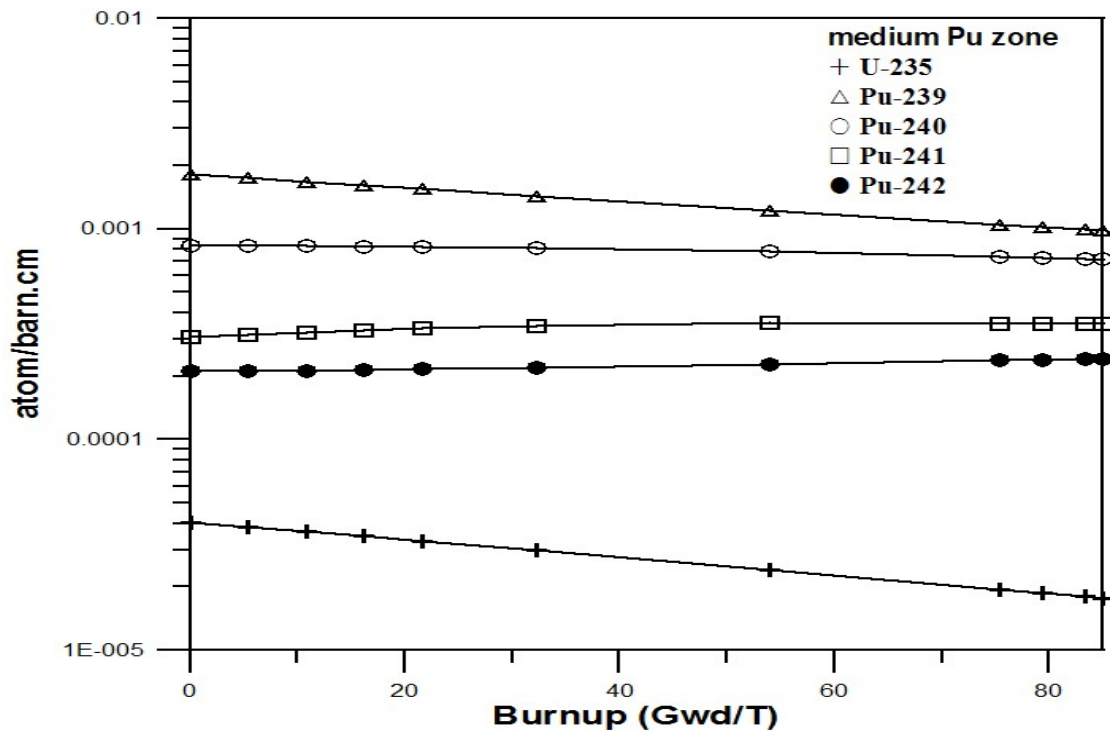


FIG. 5. Pu isotope concentrations for medium pu zone.

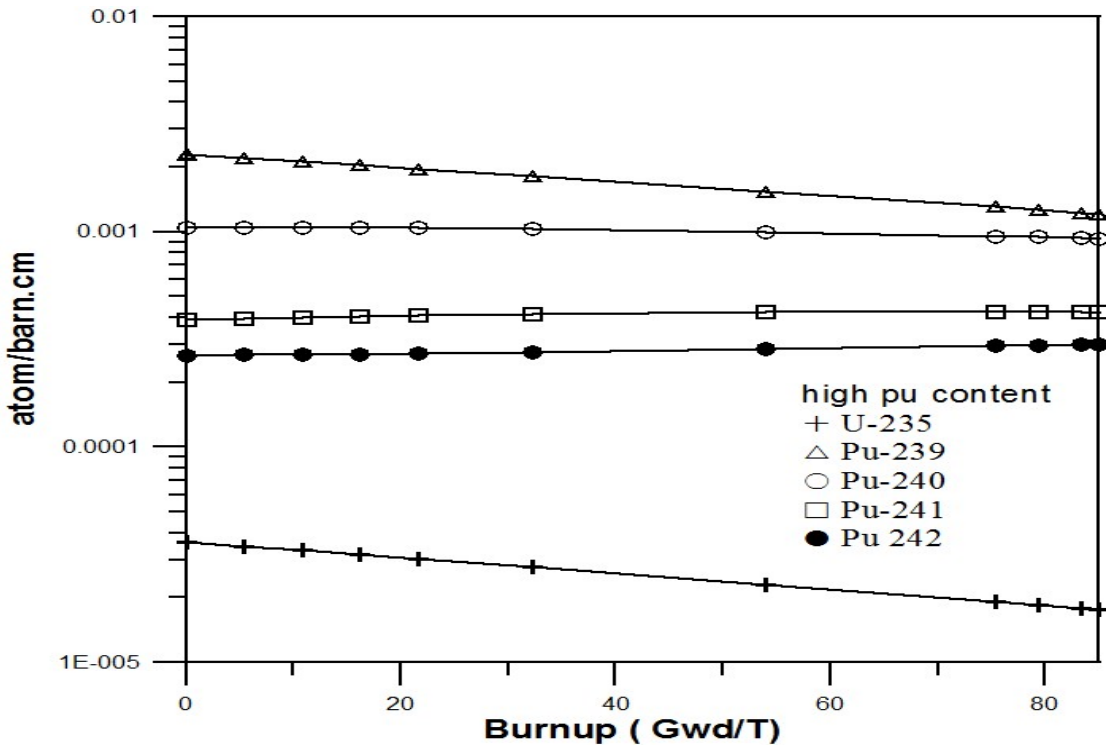


FIG. 6. Pu isotope concentrations for high pu zone.

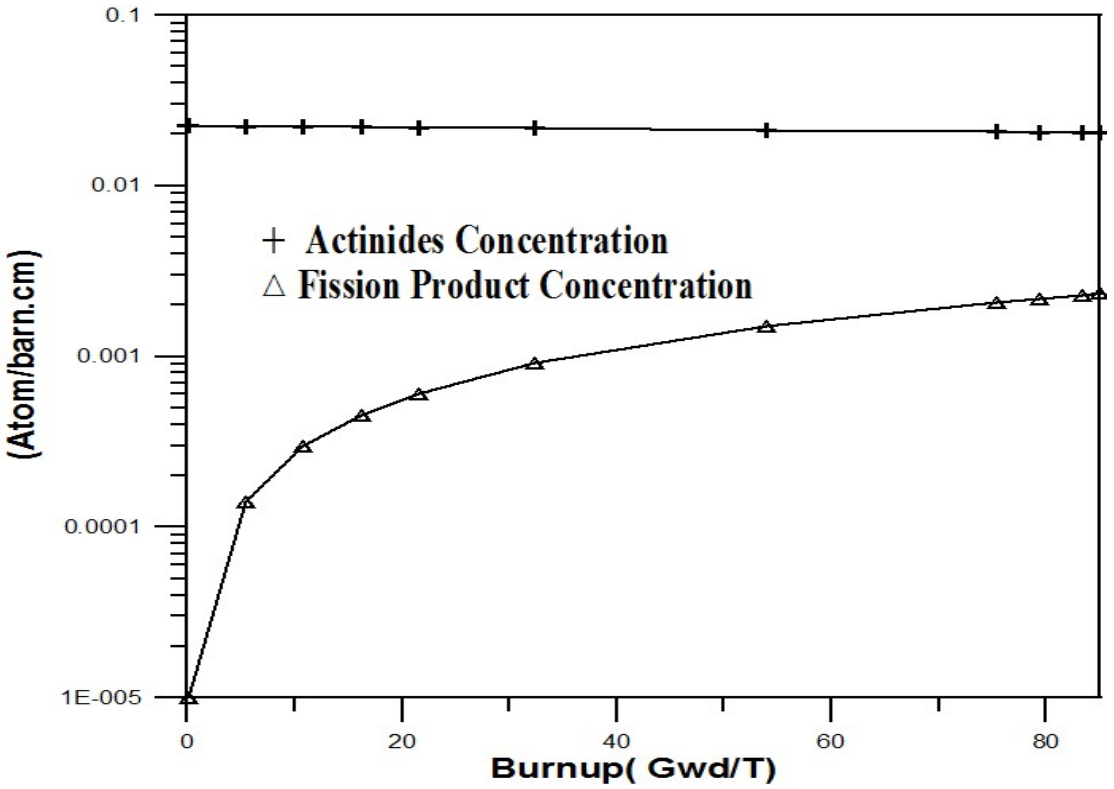


FIG. 7. Total actinides and fission product for all assembly.

5. CONCLUSION

A computer program was designed to simulate the burn up of MOX assembly in a typical operating conditions of PWR reactor. The results indicated that the fuel can be burned up to 85 GWd/T with long operating cycle.

^{239}Pu is consumed during reactor operation by a factor range from 36–45 % according to the initial ^{239}Pu concentration and ^{235}U consumed with 50 % during fuel burnup.

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POTENTIAL OF ACTIVE NEUTRON INTERROGATION TO CONTROL FISSILE MATERIALS IN 'CLOSED' FUEL CYCLE

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Abstract

Non-proliferation of nuclear materials and their losses within 0.1% level define the most important characteristics of 'closed' fuel cycle. The work is devoted to feasibility study of control the fissile nuclear materials via the differential die-away technique in case of high γ-background. Preliminary results obtained by the active neutron generator assembly showed that the minimal detectable mass of unshielded ^{235}U in empty 70 l container is ~1 mg in case of 5×10^8 n/s neutron yield and 8 min duration of a measurement.

Key Words: active neutron interrogation, neutron generator, fissile materials, spent nuclear fuel.

1. FISSILE MATERIAL LIMITS IN RADIOACTIVE WASTES

Detecting of fissile materials (FM) is a very important task in such aspects of nuclear security as material control and accountability purposes, radiation safety and criticality issues of nuclear installations, radioactive waste management sites, and for the containment and security of FM. Nowadays controlling the amount of FM in radioactive waste (RW) has become rather important due to recently accepted Russian federal law #190 "On handling radioactive waste and ..." [1] that defines fares of nuclear-waste disposal. The fares turn out to differ by almost 4 orders of magnitude for entire range of RW categories. In the meantime, the overall "radioactive legacy" in Russia reaches 540 million m^3 [2], and therefore the issue of proper categorization of RW gets a serious financial context.

TABLE 1. RADIOACTIVE WASTE (RW) CATEGORIES [3]

Waste class	Content, kBq/kg RW		
	β – emiters	α – emiters	Transurans
Particularly low level waste (PLLW)	$< 10^3$	< 100	< 10
Low level waste (LLW)	10^3 - 10^4	100 - 10^3	10 - 100
Medium level waste (MLW)	10^4 - 10^7	10^3 - 10^6	100 - 10^5
High level waste (HLW)	$> 10^7$	$> 10^6$	$> 10^5$

Fissile materials (^{235}U and ^{239}Pu) can be inside containers with RW. Table 1 presents quantitative characteristics of the RW categories and Table 2 shows quantities of pure ^{235}U, ^{239}Pu that correspond to the specified categories.

SNF categorization should be performed by both β- and α-activity. This causes no problems in case when only β- or α-active products are presented in RW, however it can lead to discrepancy while categorizing RW that contains, for example, SNF with both β- (fission products), and α-active products (actinides) [4]. To illustrate this, Figure 1 shows SNF α- and

β-activity as a function of cooling time and indicates that α-activity of SNF decreases much slower than β-activity. This means that while 2 years old SNF has the same category if measured by both activity types, the same SNF after 30 years cooling can be categorized by α-activity 5 times greater than by β-activity.

TABLE 2. LIMITS OF ^{235}U AND ^{239}PU CONTENT IN DIFFERENT RADIOACTIVE WASTE (RW) CATEGORIES [4]

Material	PLLW, < 10^4 Bq/kg	LLW, 10^4 - 10^5 Bq/kg	MLW, 10^5 - to 10^8 Bq/kg	HLW, > 10^8 Bq/kg
Pure ^{235}U, g/kg RW	< 0.13	0.13-1.3	> 1.3	-
^{235}U (4% enrichment), g/kg RW	< 0.026	0.026 – 0.26	0.26 - 260	> 260
Pure ^{239}Pu, μg/kg RW	< 4.3	4.3 - 43	43 – 43 000	> 43 000

TABLE 3. LIMITS OF ^{235}U AND ^{239}PU CONTENT IN DIFFERENT RW CATEGORIES WITH SPENT NUCLEAR FUEL (SNF)

Cooling time and SNF α-activity	Maximal content in RW	RW categories by the specific α-activity (Bq/kg)			
		PLLW (< 10^4)	LLW (10^4 - 10^5)	MLW (10^5 - 10^8)	HLW (> 10^8)
2 years,	SNF (mg/kg)	< 0.03	0.03 – 0.3	0.3 – 300	> 300
285 MBq/g	^{235}U and ^{239}Pu[a] (μg/kg)	< 0.61	0.61 – 6.1	6.1 - 6100	> 6100
30 years,	SNF (mg/kg)	< 0.05	0.05 – 0.5	0.5 – 500	> 500
202 MBq/g	^{235}U and ^{239}Pu [a] (μg/kg)	< 0.86	0.86 – 8.6	8.6 - 8600	> 8600

[a] ^{235}U and ^{239}Pu ratio is that is in SNF

Tables 3 and 4 summarize the numerical attributes of waste categories, as well as examples of fissile material content both in the form of pure ^{235}U and/or ^{239}Pu and in the form of spent nuclear fuel (SNF).

TABLE 4. LIMITS OF SNF CONTENT (MG/KG) IN DIFFERENT CATEGORIES OF WASTE

Cooling time and SNF β-activity	RW categories by the specific β-activity (Bk/kg)			
	PLLW (< 10^6)	LLW (10^6 – 10^7)	MLW (10^7 – 10^{10})	HLW (> 10^{10})
2 years, 26.41 GBq/g	< 0.04	0.04 – 0.4	0.4 – 400	> 400
30 years, 4.08 GBq/g	< 0.25	0.25 – 2.5	2.5 – 2500	> 2500

The problem is that conventional tools of FM categorization are based on low-resolution γ-spectrometry systems [5], which are "blind" to α-active actinides. Specific α-activities in such tools are usually assessed on *a priori* relationships between α- and β-activities of various types of SNF. Fig. 1 indicates, however, that such assessments can be true only if reliable and complete information about SNF is available, including burn-up and cooling age, making in turn the procedure of categorization senseless. This circumstance makes it necessary to study the possibility of determining specific α-activity of waste using direct techniques in the presence of background γ-radiation from fission products. One of the solutions of this problem is measurement of the amount of fissile materials (FM) which are presented mainly by ^{235}U and ^{239}Pu. Use of passive neutron techniques of FM control and active well coincidence counting (AWCC) [6] for such kind of measurements is limited via high FM detection thresholds (tens of grams).

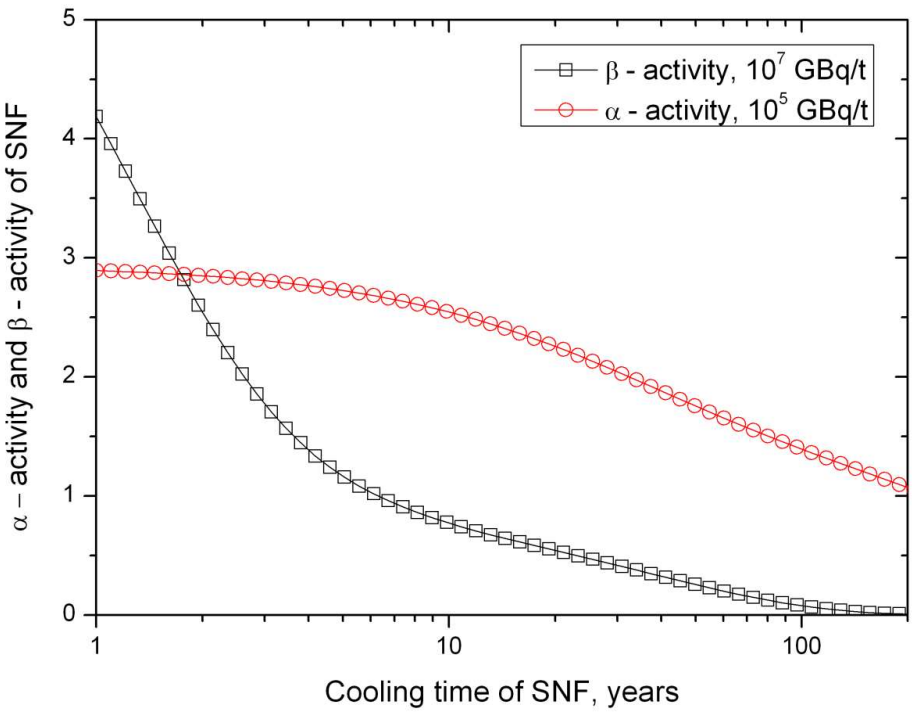

FIG. 1. SNF α- and β-activities as a function of cooling time.

Table 3 indicates that if a categorization item weights ~ 10^2 kg, mass limit that distinguishes PLLW from LLW is about ~ 10^{-4} g of FM, LLW from MLW - ~10^{-3} g, MLW from HLW - ~1 g. Therefore, the objective of this work is to study the possibility of measuring small quantities (~ 10^{-4} g to ~ 1g) of FM with methods that are more sensitive than AWCC, for instance, with differential die-away technique using pulsed neutron generators.

Unlike the open fuel cycle, the closed fuel cycled is to provide wastes with actinide amount within 0.1% of initial value. This makes supplementary requirement to check a categorized waste by absence of even low amount of FM. For instance, the SNF burned during 3 years in fast neutron reactor contains about 10% of FM, mainly ^{239}Pu. The 0.1% loss requirement provides limitation of the FM content down to 0.1g of ^{239}Pu per 1 kg SNF or about ~0.1 mg ^{239}Pu per 1 kg HLW. Meanwhile, γ-dose from SNF is about ~103 $Sv \cdot h^{-1} \cdot kg^{-1}$ or ~1 $Sv \cdot h^{-1} \cdot kg^{-1}$ HLW. So, the general technical task is to measure FM (in form of ^{239}Pu or ^{235}U as analogue) within ~10 mg limit inside a ~102 kg HLW bulk under ~102 Sv/h background.

2. FISSILE MATERIALS DETECTION AT VNIIA

The first step of solving the above task is to determine the measurement capabilities of differential die-away techniques in case of normal background. This step was done at VNIIA by the experimental set-up fabricated in 2011. The set-up (Fig. 2) is currently equipped with ING-07T neutron generator with ~109 n/s maximal flux and inspects 70 l containers.

FIG. 2. The VNIIA experimental set-up for active neutron measurement of FM amount.

The set-up itself and measuring procedures are well described in our previous paper [7] which reports that minimal detectable mass of ^{235}U ranges from 2 to 100 mg depending on the material (from graphite to boron) being filled inside of the container. The current work is to provide our new results measured with the same set-up but under new background conditions improved by optimizing high voltage supply of ^{3}He-counters to find the best ratio of their neutron-to-gamma sensitivities.

Fig. 3 shows some example of the new ^{3}He-response time distributions Φ(t) and Fig. 4 summarizes the results in the form of a function of $R(m_{235})$ measured for different filling materials and defined in Eq. 1 as:

$$R = (S_B - S_{B0}) / S_A \quad (1)$$

where

$$S_A = \int_{300}^{ta} \Phi(t)dt, \quad (2)$$

$$S_B = \int_{ta}^{tb} \Phi(t)dt, \quad (3)$$

t_a and t_b are integration limits taken here as 780 μs and 2348 μs, respectively, after neutron pulse,

S_{B0} is S_B when measured with no FM put inside the set-up.

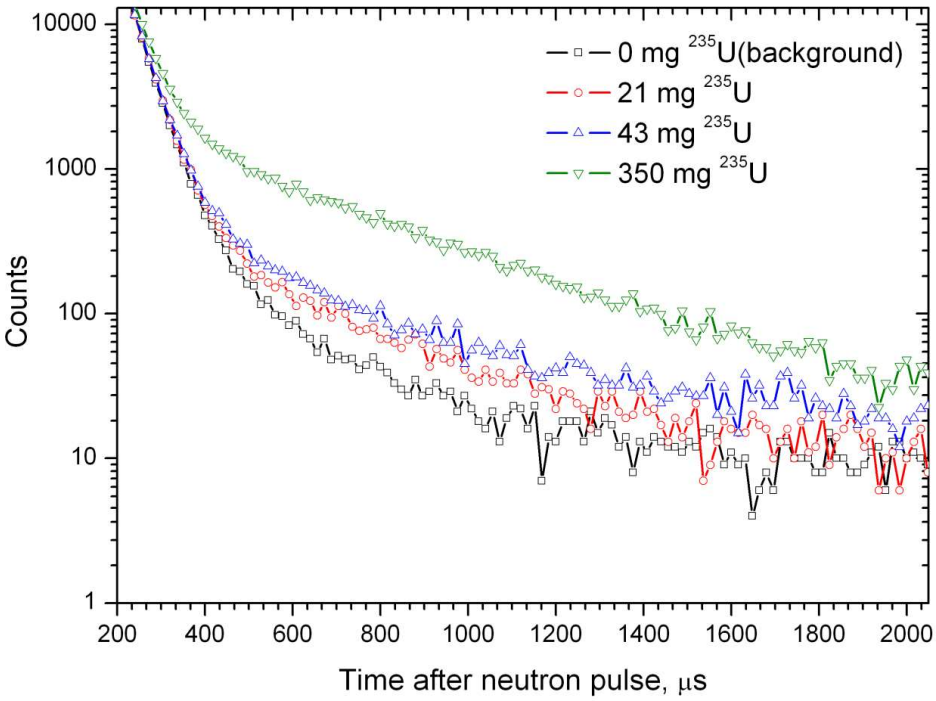

FIG. 3. Time distributions of fission neutrons responses for ^{235}U samples in an empty container (no matrix) for an 8 min measurement time and 5×10^8 n/s neutron flux.

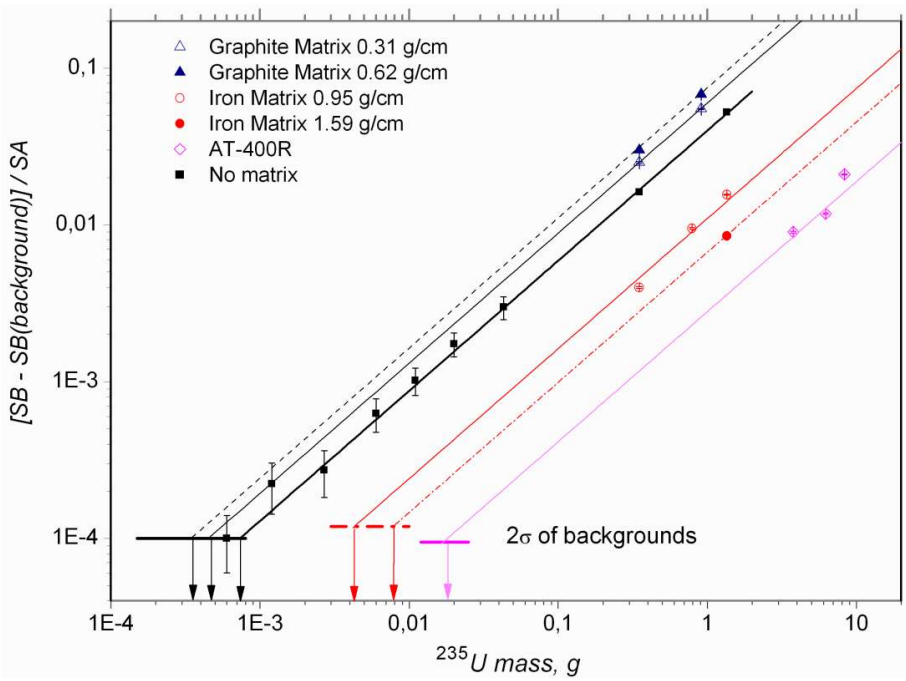

FIG. 4. Extrapolation of experimental data to assess the minimum detectable FM mass for various matrix materials in container.

The intersection of *R(m)* with double values of background uncertainty provides minimum detectable masses of ^{235}U for different materials that fill the container. The new masses presented in Table 5 turn out be 4÷6 times lower the ones reported in [7] with previous ^{3}He-counter settings.

TABLE 5. MINIMUM DETECTABLE MASS OF FM

Matrix type	^{235}U minimum detectable mass (mg)
No matrix	0.8 ± 0.3
Polyethylene 0.16 g/cm^3	0.4 ± 0.2
Graphite 0.31 g/cm^3	0.5 ± 0.2
Graphite 0.62 g/cm^3	0.4 ± 0.2
Iron 0.95 g/cm^3	4 ± 2
Iron 1.59 g/cm^3	7 ± 3
AT-400R container	20 ± 10

3. CONCLUSION

The experiments with milligrams of ^{235}U have proven the detection of ~1 mg FM (^{235}U and/or ^{239}Pu) during 8 min measurements with 5×10^{8} n/s neutron flux under normal γ-background. Detection of smaller quantities (~0.1 mg) of FM is possible using more powerful VNIIA neutron generators with higher neutron flux.

The presented results demonstrate the possibility of active neutron control of FM content in any category of waste in any nuclear fuel cycle, including the closed one. The ^{3}He is assumed as the best candidate as the neutron detector material due to its low sensitivity to γ-radiation. Our future activity is directed to verify this assumption via experiments with intensive γ-sources and to measure a dependence of minimum detectable ^{235}U mass on γ-background.

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SAFETY OF AT-REACTOR HIGH-DENSITY STORAGE OF FUEL IN POOLS
LESSONS LEARNED FROM THE FUKUSHIMA DAI-ICHI EVENT OF 2011

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Abstract

Following the challenges to wet storage of spent fuel during the events at Fukushima, the U.S. Nuclear Regulatory Commission (NRC) evaluated whether the reduced density of spent fuel in at-reactor wet storage would provide a substantial enhancement in safety. The reduced storage density would be achieved through expedited transfer of fuel to dry storage such that only fuel with less than 5 years decay would remain in wet storage. The effect of changes in storage density on the frequency and consequences of an event that drains the coolant from a spent fuel pool were drawn from a detailed consequence study evaluating the effects of a large earthquake on a spent fuel pool in a Boiling-Water Reactor with a Mark I containment design. The study compared high-density and low-density loading conditions in existing fuel storage racks and assessed the benefits of previously implemented mitigation measures. The study found that the density of spent fuel storage had little effect on the frequency of a radiological release, but the consequences of a radiological release from unmitigated damage to the spent fuel pool structure could significantly change. The NRC combined information from past studies of spent fuel storage safety and spent fuel storage licensing activities with the consequence study results to develop a conservative estimate of the safety benefit that could result from reduction in the density of spent fuel storage for the entire fleet of at-reactor storage pools. This evaluation was consistent with previous studies that demonstrated high-density storage of spent fuel in pools protects public health and safety. Furthermore, the evaluation found that a transition to low-density storage would provide only a minor or limited safety benefit, and that its expected implementation costs would not be warranted when compared with the expected benefits.

1. INTRODUCTION

Spent fuel pools (SFPs) at U.S. reactor sites were originally designed to support short-term storage and transfer to a reprocessing facility after a relatively short decay time. However, policy decisions related to nuclear non-proliferation and delays in the establishment of a repository for spent fuel have resulted in the need for extended on-site storage of fuel. The U.S. operating reactor licensees have overwhelmingly favored high-density pool storage for this extended on-site storage. In this context, high-density pool storage refers to the incorporation of solid neutron absorber panels in the storage racks to maintain the required margin to criticality while reducing the center-to-center pitch of stored fuel assemblies.

The U.S. Nuclear Regulatory Commission has found the high density storage of irradiated fuel to provide adequate protection of public health and safety. This finding is based in part on the robust construction of the pool structure, the provision of a leak-tight stainless steel liner, and design measures that preclude a substantial loss of coolant inventory from the pool under both expected operating and accident conditions. However, concerns about security of spent fuel storage in pools and uncertainty about the status of the SFPs at the Fukushima Dai-ichi site following the earthquake and tsunami of 11 March 2011, prompted public interest in a transition to low-density storage.

2. HISTORY OF SPENT FUEL POOL SAFETY EVALUATIONS

The NRC staff has long recognized that very unlikely events that drained the coolant covering irradiated fuel stored in a SFP could have significant offsite consequences. Although containing very little energy relative to a power reactor, the release could be on the order of a

large off-site radiological release from an operating reactor due to the large cesium inventory present in the SFP and the lack of a strong containment structure. In the late 1980's, the NRC staff evaluated the potential for large releases caused by beyond-design-basis events affecting SFP storage and presented the results of this evaluation in NUREG-1353 [1]. The staff determined that storage of fuel in high-density configurations was safe and protective of the overall public health, largely as a result of the low frequency of events that could challenge the integrity of the SFP structure.

After several U.S. power reactors permanently ceased operation in the 1990's, the NRC staff completed a detailed study of decommissioning risk to assess whether emergency planning, security, and insurance requirements applicable to operating reactors could be appropriately reduced for decommissioning reactors and the timing of such a reduction [2]. For spent fuel aged one year, the NRC staff estimated the frequency of fuel uncovery to range from 5.8×10^{-7} to 2.4×10^{-6} per year for the plants studied. The fuel uncovery state was used as a simplifying and conservative surrogate for the severe fuel damage resulting from inadequate cooling following a loss of coolant inventory. Consistent with the results of the previous study, beyond design basis seismic initiating events dominate the fuel uncovery frequency estimate. Fuel cask handling accidents were found to be the next largest contributor to the frequency estimate. Although the frequency was low, the NRC staff concluded that fuel overheating, and thus a radiological release, remained possible indefinitely unless bounds were placed on the damage state to assure a path for air cooling.

3. CHANGES AFFECTING SPENT FUEL STORAGE SAFETY AND ANALYSES

Following the events in the U.S. on 11 September 2001, the NRC staff initiated studies to evaluate the security of spent fuel storage. Pursuit of these studies resulted in improved analytical tools to assess the progression of the event if stored fuel becomes uncovered in a SFP. The improved analytical tools were developed through a combination of separate effects testing and validation of simplified models to experimental studies and precise analytical solutions.

While these security assessments were underway, outside organizations proposed measures to reduce the density of fuel storage in pools, including a report published by R. Alvarez et al. entitled "Reducing the Hazards from Spent Fuel Power-Reactor Fuel in the United States" [3]. In this report, the authors proposed moving all fuel with more than 5 years decay be placed in dry storage casks in an expedited program. The NRC found the recommendation to be based on a simplified model that produced conservative results. Therefore, the expedited transfer was not justified.

At the direction of the U.S. Congress, the NRC funded a National Academies of Science (NAS) evaluation of ongoing security assessments and the recommendations of outside organizations proposing expedited transfer of fuel to dry cask storage. In 2006, the NAS issued a public report entitled "Safety and Security of Commercial Spent Nuclear Fuel Storage" [4] which documented several findings and recommendations. The NAS recommendations generally mirrored NRC staff conclusions, except the NAS recommended early movement of fuel to dry cask storage.

The ongoing security assessments and the NAS recommendations contributed to new regulatory requirements to enhance the safety and security of spent fuel, including strategies to maintain or restore the capability to cool fuel in high-density storage configurations following damage to large areas of the facility. The strategies include actions to optimize the

ability to cool fuel by distributing more recently discharged fuel among older fuel with lower heat generation and maintaining portable equipment to add water to the SFP by various paths.

The earthquake and subsequent tsnami along the northeastern coastline of Japan on 11 March 2011, resulted in reactor damage that presented several challenges to the operators at the Fukushima Dai-ichi site. Among these challenges was monitoring of the status of the spent fuel pools (SFPs), particularly the Unit 4 SFP that contained the recently discharged core from the last Unit 4 operating cycle. Although it was later confirmed that the fuel in the Unit 4 SFP remained covered and the pool structure remained water-tight, the hydrogen explosion that damaged the Unit 4 reactor building heightened concerns among many observers that the fuel had been uncovered. This event and the substantial stakeholder interest prompted NRC staff evaluation of expedited transfer of fuel to dry cask storage. This evaluation was completed through a consequence evaluation that compared the change in frequency and consequences associated with a reduction in storage density and a regulatory analysis.

4. UPDATED CONSEQUENCE STUDY OF A BEYOND-DESIGN BASIS SPENT FUEL POOL EVENT

4.1. Approach and major assumptions

To help address the continued interest in expedited transfer of fuel, the NRC staff set out to evaluate the effects of a beyond-design-basis event that damaged the SFP and the difference in consequences that result between the following two fuel storage configurations:

- A relatively full SFP with the fuel in a high-density configuration with the most recently discharged assemblies distributed in a 1×4 pattern among older, colder fuel;
- A partially filled pool where all fuel with more than 5 years decay has been transferred to dry cask storage and the most recently discharged assemblies distributed in a low-density 1×4 configuration among empty fuel assembly storage locations.

The analysis considered the new regulatory requirements associated with the security assessments. The primary considerations were the implementation of the improved fuel storage configuration and the availability of mitigation capabilities. The improved fuel storage configuration consists of a repeating 1×4 pattern where only one face of each low decay power assembly or empty storage location is adjacent to a high decay power assembly, as depicted in Fig. 1 for the high-density case.

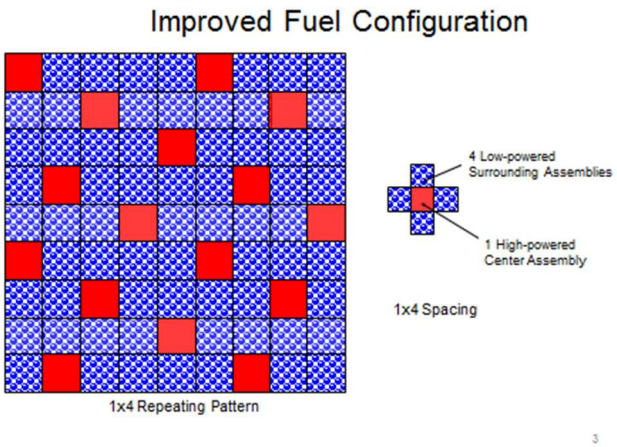

FIG. 1. Improved fuel configuration

The analysis considered a single initiating event, which was a large earthquake with a low probability of occurrence, and evaluated the effect of that event on a representative U.S. boiling-water reactor with a General Electric Mark I containment design. The selected earthquake spectral accelerations were based on the 2008 U.S. Geological Service model scaled to a 0.7g peak ground acceleration. The NRC staff compared the response spectra used for this analysis with accelerations experienced at Japanese nuclear power plants during the 11 March 2011 Tohoku Earthquake with moment magnitude of 9.0 and the 16 July 2007 Niigataken Chuestu-Oki earthquake with moment magnitude of 6.6, which was near the 7-unit Kashiwazaki-Kariwa site. Although these two earthquakes produced severe ground motions at the affected sites, the staff found the response spectra used in the study was more challenging to the SFP structure than the actual event accelerations [5].

4.2. Structural analysis

The NRC staff developed detailed finite element models and analyzed the SFP and reactor building seismic structural response using non-linear analysis methods. The analysis included dead loads from the structural elements, the hydrostatic pressure of the water, and the weight of the fuel assemblies and storage racks; and seismic loads associated with acceleration of structural elements, hydrodynamic forces, and dynamic forces from the fuel and storage racks. The analysis results indicated the potential for concrete cracking along the interface between the SFP walls and floor slab, but the relative movement indicated low strains in the liner embedments. Analysis of the structure surrounding the pool operating floor indicated that it would remain intact as a result of its relatively light weight. Therefore, the most likely outcome was a damage state with concrete cracking but no leakage. However, uncertainty in the liner strains create the potential for liner tears that, when combined with concrete cracking, could lead to slow to moderate leakage from the SFP.

4.3. Accident progression analysis

The accident progression analyses covered numerous cases that considered the point of time in the refueling cycle with respect to decay heat rates and pool configurations, the density of fuel storage, the leak rate (slow or moderate), and the deployment of mitigation capabilities. As a result of the low decay heat later in the operating cycle and the leak location at the bottom of the pool, analyses demonstrated that the fuel would be air-coolable in either storage

configuration after 60 days decay. Early in the operating cycle, the storage configuration did not affect the likelihood of the highest decay heat assemblies overheating within the 72 hour analysis limit. However, the storage configuration did affect the magnitude of the radiological release, with the high-density storage configuration presenting both a greater initial inventory of radioactive cesium and a greater fraction of the total cesium inventory released from the building.

The rate of water loss also affected the magnitude of the release. The moderate leak case resulted in complete draining of the pool before the upper portions of the fuel assemblies could overheat. This condition limited the event to air oxidation of the assemblies and relatively low release fractions. However, the small leak could allow high cladding temperatures to develop before the water level dropped below the storage rack base-plate, which could support steam oxidation and production of hydrogen. In the few cases where hydrogen accumulation exceeded the detonation limit, the cesium release fractions were calculated to be substantially higher, primarily due to loss of the reactor building integrity.

The effective deployment of mitigation capabilities significantly reduced the likelihood of a release. For fuel with greater than one week of decay, mitigation capabilities were effective in preventing a release regardless of the fuel storage density or leak size. However, for fuel with less than one week decay, the mitigation capability may not provide sufficient cooling to prevent a release when the leak rate exceeds the rate of water makeup (i.e., moderate leak scenarios).

4.4. Consequence analysis

The consequence analysis results reflected a slowly developing event producing a large release of volatile long-lived fission products, principally cesium isotopes. Based on the characteristics of this release, no immediate health effects were considered credible. Furthermore, the slow evolution of the event and the relatively low dose rate supported relocation of affected populations to prevent long term health effects. These results were generally consistent with the earlier studies of SFP accidents.

The magnitude of the release could result in significant land contamination. However, the very low likelihood of an event with the potential to result in a large release and the availability of equipment to prevent or mitigate the release support continued safe storage of fuel in high density configurations.

5. REGULATORY ANALYSIS OF EXPEDITED TRANSFER OF SPENT FUEL TO DRY STORAGE

To evaluate whether expedited transfer to dry storage should receive additional analysis, the NRC staff completed a regulatory analysis. The regulatory analysis process involves two principle components: an assessment of the potential increase in the overall protection of public health and safety and a comparison of the implementation costs with the expected benefits. The level of protection of overall public health and safety was measured based on the quantitative health objectives derived from the U.S. NRC Policy Statement, “Safety Goals for the Operation of Nuclear Power Plants” [6]. The expected benefits are calculated by placing a monetary value on averted accident consequences, which include the total occupational and public exposure resulting from the accident and the on-site and off-site property damage. The implementation costs were drawn primarily from an industry evaluation of costs associated with expedited transfer of fuel with greater than 5 years decay to dry storage over a 5 year period [7]. The comparison of implementation costs uses present

values of the implementation costs and the present value of a series of annual benefits, which are calculated by multiplying the accident frequency by the monetized value of the averted accident consequences.

5.1. Regulatory analysis approach

The NRC staff presented the results of its regulatory analysis in COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel" [8]. The NRC staff extended the regulatory analysis to encompass the fleet of operating reactors by evaluating the following groups:

(1) Elevated SFPs (BWRs with Mark I or Mark II Containment designs);
(2) At-grade dedicated SFPs (most Pressurized-Water Reactors (PWRs) and BWRs with Mark III Containment designs);
(3) New nuclear power plants (AP1000 PWRs currently under construction);
(4) At-grade SFPs shared between units (select dual-unit PWRs).

For each group, the NRC staff selected a representative plant and determined representative characteristics of the stored fuel inventory for the high-density (current practice) and low-density (fuel with greater than 5 years decay transferred to dry storage) configurations.

The NRC staff also expanded the scope of initiating events that may lead to fuel no longer being covered by cooling water. Due to the limited scope of the consequence study [5], certain insights gained from this study were limited to the specific reactor group and, for some information, the specific initiating event. Therefore, the NRC staff used simplified and conservative information from past studies in combination with revised seismic hazard information to develop complete models that included a spectrum of initiating events. Additional initiating events included a more severe earthquake, a spent fuel storage cask drop that damages the SFP structure, and several events that result in loss of SFP coolant through boiling or evaporation that uncovered the fuel. As indicated by the consequence study, the frequencies used for fuel overheating were the same for both the high-density and low-density storage configurations, but the cesium release fractions were significantly higher for the high-density storage configuration.

Because the purpose of the regulatory analysis included determining whether additional study was necessary to fully evaluate the differences between high-density and low-density storage, the NRC staff used certain assumptions to maximize the difference between the two configurations. These assumptions included successful deployment of mitigation equipment for the low-density configuration and no mitigation of the high-density configuration. This assumption had the effect of reducing the radiological release frequency for the low-density configuration by over one order of magnitude.

5.2. Regulatory analysis results

The safety goal evaluation determined that the benefit to overall public health and safety provided by a transition to low-density storage configurations would be limited. Despite the use of simplified or conservative values from past studies, the potential benefit was found to be a small fraction of the safety goal. As mentioned in the discussion of the consequence study, this finding results from both a low frequency of radiological release due to the design of the SFP and limited sensitivity to the magnitude of the release due to the slow accident progression and implementation of effective protective actions.

The NRC staff performed the economic modeling using information selected to be representative for each plant group. For the evaluation cases considered most representative for each plant group, the projected cost of implementation of low-density storage significantly exceeded the projected averted costs associated with the consequences of an unlikely severe accident in a SFP with high-density fuel storage.

6. ADDITIONAL SAFETY ENHANCEMENTS FOR SPENT FUEL STORAGE

In response to the Fukushima Dai-ichi accident, the NRC staff implemented regulatory actions to further enhance SFP safety. On 12 March 2012, the staff issued an order that required all licensees install a reliable means of remotely monitoring wide-range SFP levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event. Although the primary purpose of the order was to ensure that operators were not distracted by uncertainties related to SFP conditions during accident response, the improved monitoring capabilities will help in the diagnosis and response to potential losses of SFP leak-tight integrity. In addition, the staff issued an order that requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore SFP cooling capabilities, independent of alternating current power, following a beyond-design-basis external event. These requirements ensure a more reliable and robust mitigation capability is in place to address degrading conditions in SFPs than was assumed in the SFP consequence study.

7. CONCLUSION

The NRC staff combined information from past studies of spent fuel storage safety and spent fuel storage licensing activities with the results of a new consequence study to develop a conservative estimate of the safety benefit that could result from reduction in the density of spent fuel storage for the entire fleet of at-reactor storage pools. This evaluation was consistent with previous studies that demonstrated high-density storage of spent fuel in pools protects public health and safety. Furthermore, the evaluation found that a transition to low-density storage would provide only a minor or limited safety benefit, and that its expected implementation costs would not be warranted when compared with the expected benefits.

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SESSION 4

AL-CLAD METALLIC URANIUM LEGACY FUEL AFTER MORE THAN 50 YEARS OF UNDERGROUND DRY STORAGE

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Abstract

Spent legacy fuel elements from the early 50s and 60s originating from the first Norwegian RR, JEEP I, are stored in an underground dry storage since the mid 60s. It is known that some of the Al-clad naturally enriched metallic uranium fuel elements had defects by the time they were discharged from the JEEP I RR and put into the cooling pond. In the mid 60s an underground dry storage site for JEEP I fuel was opened and the spent metallic fuel elements were transferred to the dry storage site: There, each element was stored individually in an open Al-container in a storage pit. Then, in 1982 the fuel elements were repacked into closed and tight stainless steel packages and stored dry, each element in one tight package and each package inside one underground storage pit. For many years the dry storage pits were sealed and opened for IAEA inspections, only. Now, and due to the age of the storage and the knowledge of the defects in the fuel, the spent fuel aging management program has to include inspection of the storage pits with regard to corrosion of the pit lining, the state of stainless steel package, the state of the inside of the package with respect to hydrogen, humidity, corrosion and signs of fuel degradation.The paper illuminates not only the NDT examinations performed and the findings, but also the challenges, met after 50 years of storage. Next to the NDT examination the long time dry storage induced development of the defects in the fuel elements was studied in detail by neutronradiography and the fuel and defect microstructure by metallography and SEM.

1. INTRODUCTION

IFE manages an underground dry storage of spent, legacy fuel elements originating from the first Norwegian research reactor JEEP I (1951 to the mid-60s). Most of the pits contain aluminium clad fuel elements with naturally enriched metallic uranium. The legacy fuel originates from a time before today's requirements on documentation. In addition, regular inspections as of today were not conducted in earlier days [1−3].

The paper presents background history of the Al-clad metallic uranium legacy fuel, results from PIE performed on one of these elements after more than 50 years in an underground dry storage and finally reveals the challenges that evolved from the PIE findings which have led to increased risk assessments and tasks in the spent fuel ageing management program.

1.1. Background

1.1.1. Fuel element description

Driver fuel from JEEP I first charge was manufactured from metallic, naturally enriched uranium in aluminium cladding. The metallic uranium fuel slugs were originally cast and machined for use in British high power reactors. Microscopic surface flaws on the metallic uranium fuel slugs were however limiting the use in high power reactors although these flaws were accepted in the Norwegian low power reactor JEEP I that was originally built for 100 kW thermal, then upgraded to 450 kW thermal [4].

One metallic uranium slug is approximately 290 mm long and has a diameter of approximately 25 mm and a mass of approximately 2700 g. One JEEP I element is 2.7 m long and consists of two fuel rods with stacked metallic uranium fuel slugs that are press fitted in

aluminium tubings that are separated by an aluminium plate. The active fuel stack length is ≈1.9 m e.g. 6 ½ stacked uranium slugs per rod giving 35.6 kg metallic uranium per fuel element (44 kg metallic uranium per element in total). The lowermost part of the rod contains a graphite slug approximately 60 mm long and with the same diameter as the metallic uranium slugs.

The cladding was made from extruded pure aluminium tubes. The aluminium cladding from JEEP I 1st charge (1951) is non-anodized. All later aluminium claddings were anodized to enhance corrosion resistance.

1.1.2. Inspections during operation

During reactor operation half yearly inspections were carried out of the JEEP I driver fuel elements. During these inspections it was examined whether the aluminium cladding was intact, and the element could be put back into the core or if the cladding was damaged the element should be permanently removed. Signs of swellings, cracks, aluminium surface degradation e.g. corrosion/pitting or blisters in cladding are typical reasons to discharge an element. Such features may result in penetration of the cladding material and cause contamination of the heavy water.

1.1.3. Management, maintanance and inventory of the dry storage site

Discharged JEEP I fuel elements were placed in a cooling pond and later transferred to an underground dry storage site where most were kept from the mid-60s until today.

Dry storage site — The JEEP I dry storage facility consists of top and bottom concrete slabs with the top on ground level and bottom approximately three meters underground. The storage pits were made of carbon steel tubes with and sand/stones as insulation between the tubes. The top of each tube is covered with a metal cap and an expansion gasket (rubber sealing) [3]. For twenty years most fuel elements were kept in open aluminium canisters in the pits to let the elements dry while they were cooled by the surrounding gravel in the underground dry storage.

In the 80s some pits and few open aluminium canisters were subjected to water ingress and subsequent drying of the wells. All fuel elements were repacked into new, dry, originally gas tight 1.5 mm wall thickness stainless steel canisters without measuring the remaining humidity in the elements. One canister was placed in each pit. The pits with originally tight metal lid with a rubber seal ring were sealed by IAEA. In addition, a ground water monitoring system was installed in the dry storage site. No alarm occurred during this time.

Before 2010, defects were observed in the concrete slab around the tubes, such as cracks, exfoliation of the paint, and humidity/condensation on the inner tube surface. Subsequently, an action plan was put up and the upper concrete slab was repaired in 2010 by replacing 15-20 cm of concrete of the top slab. The new concrete was painted and during inspection in 2011 it was noticed that the paint had exfoliated again. As an explanation for this exfoliation it was suggested that the concrete still had a high water content by the time it was painted.

The dry storage site has 76 pits with aluminium clad legacy fuel elements with naturally enriched uranium, thereof 19 pits are difficult to retrieve the fuel canisters stored. Presently maintenance work is ongoing to resolve this issue.

1.1.4. Documentation

For each fuel element there is one log card that specifies identification number, production masses, enrichment, irradiation history, burnup data, observations from interim visual inspections during reactor operation including detected defects, dates when the element was discharged and moved to wet storage and later to dry storage, dates when the elements were moved or repacked and whether maintenance or reparation was performed. No information about changes/degradation in aluminium cladding during storage is provided from the log cards. An ongoing spent fuel managing program will provide information about changes/degradation in both aluminium cladding and metallic uranium fuel during dry storage.

2. RISKS RELATED TO STORAGE OF AL-CLAD METALLIC URANIUM

Three main risks are related to storing and handling aluminium clad metallic uranium:

- Presence of uranium hydride according to Eq. (1);
- Ongoing hydrogen formation according to Eqs. (2, 3);
- Possible Wigner effect in the graphite end plugs.

A number of fuel elements were discharged because of cladding failures detected during interim inspections during operation. The remaining water content in these defected elements after 60 years is not known. Possible water content in defected elements was estimated and it is low due to the limited free volume in the press fitted elements. Remaining water in defected fuel elements stored in a closed canister allows ongoing corrosion, uranium hydriding and formation of hydrogen gas according to [5]:

$$7U + 6H_2O \leftrightarrow 3UO_2 + 4UH_3 \qquad (1)$$

Uranium hydride is an intermediate compound that eventually will oxidize when it is exposed to oxygen or water according to [5, 6]:

$$2UH_3 + 2O_2 \rightarrow 2UO_2 + 3H_2 \qquad (2)$$

$$2UH_3 + 4H_2O \rightarrow 2UO_2 + 7H_2 \qquad (3)$$

3. EXPERIMENTAL PIE

Aluminium clad metallic uranium is not stable for long term storage because of its chemical reactivity as described in Eqs. (1-3). Therefore, an international committee, 'Teknisk Utvalg', was selected in 2009. In cooperation with 'Strandenutvalget' the objectives were to analyze the risks involved with storing and handling this type of spent fuel and especially with partly failed aluminium cladding and to provide recommendations for future spent fuel management. As an outcome the committee made clear the importance of knowing the present condition of the elements and knowing the materials degradation and corrosion products. After recommendations from 'Teknisk Utvalg' and 'Strandenutvalget' the state of the fuel was examined in NMATs hot cell laboratories at Kjeller in 2012 [2, 7, and 8].

PIE in terms of visual inspection, neutron radiography, metallography and SEM/EDS were done on one fuel element from JEEP I 1st charge. The state of the fuel was examined with respect to 'swellings' or 'blisters' (as noted in the log cards) and to check if these were connected with uranium hydriding and cladding hydriding. Further, an issue was to analyze the hydrides to check if they had started to oxidize during storage according to Eq. (3).

3.1. Element selection

There were some criteria that had to be met when selecting which element to examine:

- JEEP I 1st charge from 1951 with non-anodized aluminium cladding;
- Presence of swellings, blisters, cracks and pitting notifications on the log card
- Low burnup;
- The stainless steel canister had to be possible to elevate with the 500 kg crane in the dry storage facility.

The low burnup fuel element with ID 25 was chosen for the examination. Element 25 was in the core from July 1951 to March 1953 and reached a burnup of 0.130 MWd/tU. It was removed early from the core because of swellings in the cladding that were detected during one regular inspection. On 31 January 2000, element 25 was placed in a stainless steel canister and put in hole 94 in JEEP I Stavbrønn.

3.2. Practical procedure

For visual inspection the element was photographed along its entire axis in four orientations to document defect locations, dimensions of cracks, pitting, swellings, blisters, appearance, colour and morphology of eventual corrosion products. After visual inspection one fuel rod was dismantled from the element.

Neutron radiography (NR) was performed on the dismantled rod to document hydrogen rich locations (hydriding of the uranium metal, swellings). Images from NR served as a guide for determining locations of interest and for creating a cutting plan (Fig. 1) for further destructive examinations.

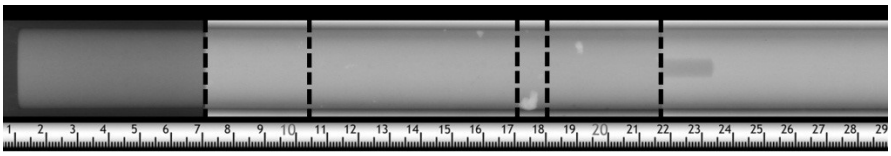

FIG. 1. NR image – Cutting scheme for JEEP I 1st charge fuel element 25 – Hydrogen rich areas appear as bright spots.

4. PIE RESULTS

Following observations were made during visual inspection (see Figs. 2−3):

- Slightly deformed aluminium plate between the two rods of the fuel element indicate creep induced deformation;
- Several local clad swellings (both rods) indicate possible fuel corrosion/uranium hydriding;
- Several locations on the surface looks scratched/chiselled - Chiselling was performed intentionally on the aluminium surface after one of the half yearly inspection when several areas were found to have started pitting corrosion. The purpose was to remove heavy metal particles that were attached to the aluminium surface and were assumed to initiate pitting attacks;
- Brown stains/stripes along the rods due to local mechanical and/or chemical reaction on the surface (deformation and deposition or corrosion);

- Dark stains (pitting, corrosion) and brighter areas (oxidized areas);
- Several small scratches from handling and one large scratch at distance around 50-60 cm from bottom of one rod.

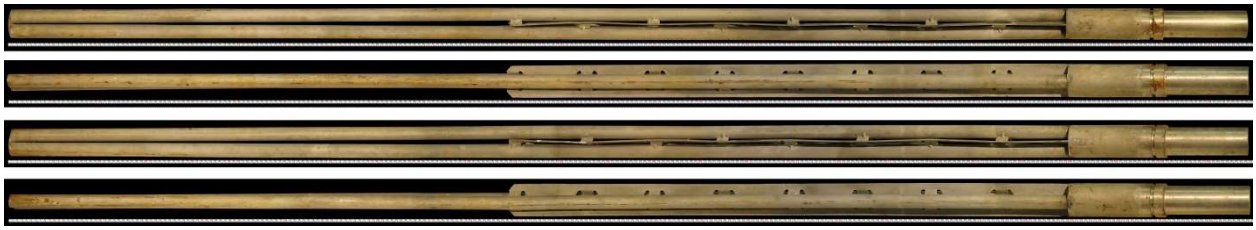

FIG. 2. Overview photos of JEEP I 1st charge fuel Element 25 from four orientations.

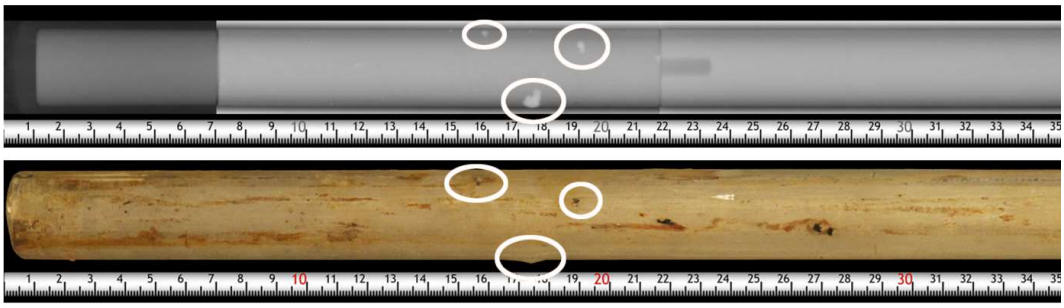

FIG. 3. Lower part of investigated rod from JEEP I 1st charge fuel element 25: a) NR image (Hydrogen rich areas appear bright in NR) and b) overview photo.

Neutron radiograph (NR) is presented in Fig. 3 (upper). The white spots in the NR image show good agreement to surface defects, swellings in the aluminium cladding as is shown in Fig. 3 (lower). Bright spots in the NR indicate presence of an element with high absorption- or scattering cross section for neutrons, in this case hydrogen (uranium hydride). The fuel rod was cut in order to observe a cross section at the position for the largest of these hydrogen rich locations.

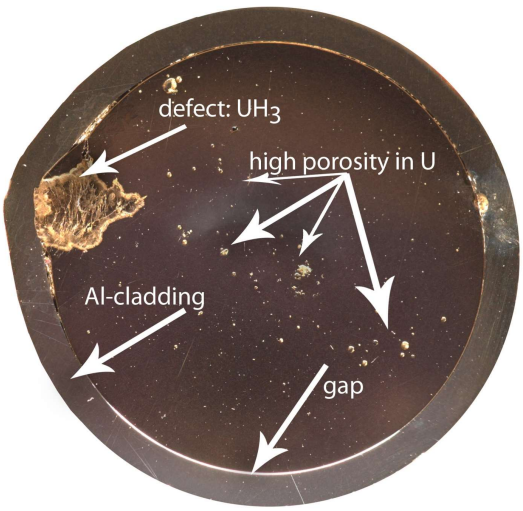

FIG. 4 LOM macrograph — Cross section of JEEP I 1st charge fuel element 25 — metallic uranium fuel in aluminium cladding.

Cross-section through the swollen part showed a defect in the metallic fuel. The size of the defected area is almost 5% of the total metallic uranium area in Fig. 4. A reactor operation induced gap between cladding and metallic uranium of <0.1 mm was observed (see Fig. 4).

Two LOM-images of the defected area are presented in Fig. 5 that shows the defect at the boundary to metallic uranium. When looking at the boundary between defect and metallic uranium there seem to be cracks that stretch into the metallic uranium. If looking closely these potential cracks seems to be filled. These brown coloured threads are not visible in images from SEM (Fig. 7). SEM examination suggests that the dark threads are no cracks but hydrides. The uranium hydride in the large defect has plenty of significant cracks with a thin net of small cracks in between (Fig. 6). Cracks, here, are fast transport paths for humid air to react with uranium metal and form uranium dioxide and uranium hydride according to Eq. (1).

Hydrogen embrittlement is a severe type of hydrogen damage that occurs when hydrogen atoms diffuse via grain boundaries and forms a net of brittle hydrides. The density of uranium hydride is 10.95 g/cm^{-3} compared to the density of metallic uranium that is 19.05 g/cm^{-3}. This means that once uranium hydride is formed a volumetric expansion of the metal matrix occurs.

The optical microscope (LOM) detail images of the swelling (Fig. 5) showed highly cracked uranium hydride with a large surface area. The cracks are easy transport paths for water/humid air which act with uranium hydride and forms uranium oxide and hydrogen gas (as described in Eq. 3). Most of the uranium oxide was detected close to the metal and in hydrides along the grain boundaries in the uranium metal (Fig. 8).

This leaves the question open regarding ongoing corrosion and how much humidity is still present inside the fuel rod to promote the oxidation and hydrogen formation reaction as described in Eq. (3).

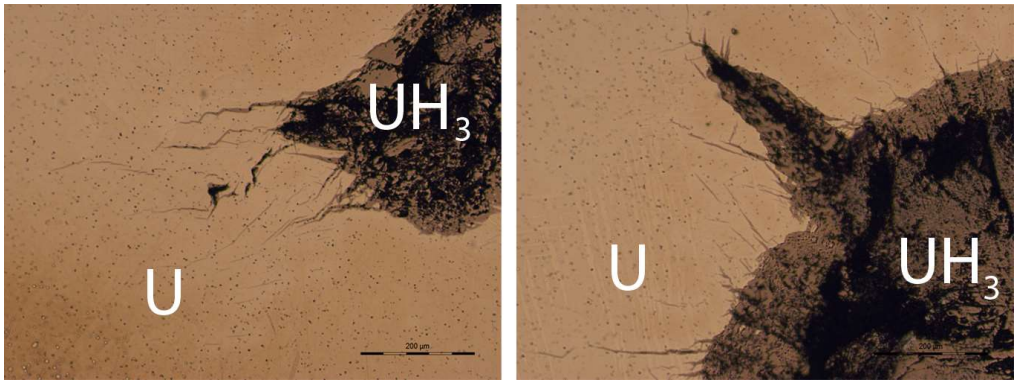

FIG. 5. LOM micrograph – Defected area in metallic uranium slug in specimen from JEEP I 1st charge fuel element 25.

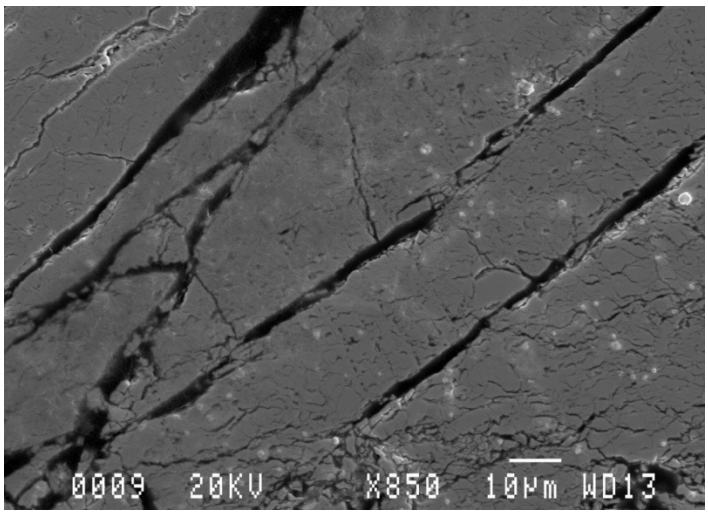

FIG.6. SEM — Uranium hydride in metallic uranium from JEEP I 1st charge fuel element 25.

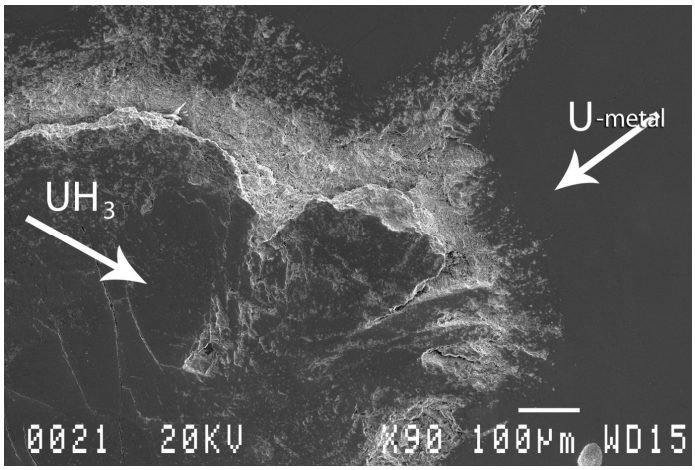

FIG. 7. SEM image — Defected area on specimen from JEEP I 1st charge fuel element 25 — Uranium hydride with oxide rim on the border towards uranium metal.

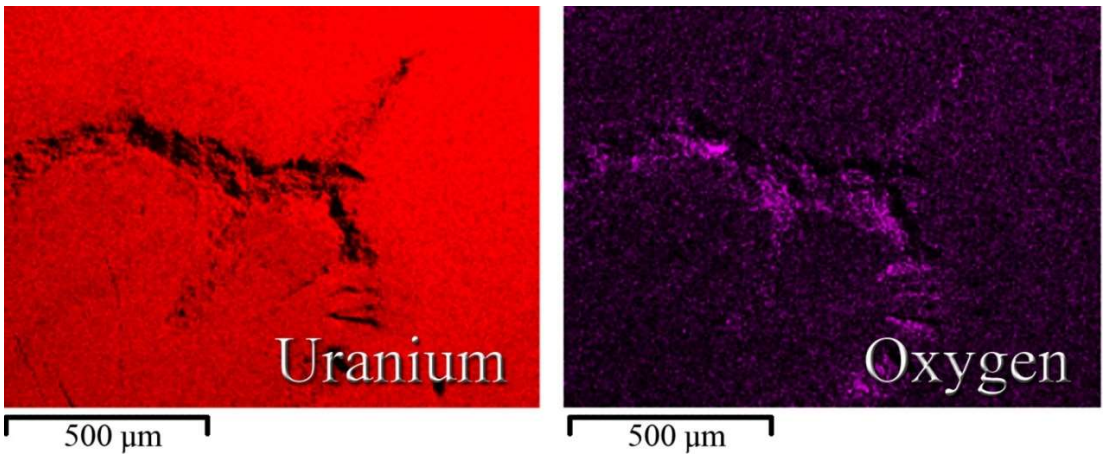

FIG. 8. EDS mapping — Comparison of defected area to intact area of specimen from JEEP I 1st charge fuel element 25 with respect to oxygen presence — Uranium hydride with oxide rim on the border towards uranium metal.

5. FURTHER WORK

Due to the age and the state of the 60 year old storage site and the knowledge of the defects in the fuel, work is ongoing to restore and maintain its functionality.

The spent fuel management program includes three main items:

- Storage site: Maintain and restore the functionality of the 60 year old storage in terms of humidity control, corrosion control of pit liner and assure canister moveability in pits;
- Canisters: Verify the state of the stainless steel canisters, verify the atmosphere (hydrogen gas detection) inside the canister, adapt top lids of canisters to allow for monitoring and assure control of the canister atmosphere (vacuum drying, gas change in canister);
- Fuel elements: Check for ongoing corrosion (indicator is hydrogen evolution).

The work described in this paper, on the Norwegian spent fuel management program for Al-clad metallic uranium after more than 50 years of underground storage, is a necessary safety measure while waiting for a national decision to be taken for final treatment/stabilization of the legacy fuel.

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AGEING MANAGEMENT OF SPENT NUCLEAR FUEL FACILITIES

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Abstract

Argentina has three nuclear Power Plants in operation, Atucha I (CNA I), Embalse (CNE) and Atucha II (CNA II). All of them are based on natural uranium as fuel with heavy water moderation and refrigeration. Atucha I (370 MW(e)) and Atucha II (750 MW(e)) are almost unique in their type, both are Pressure Vessel type. The Embalse Nuclear Power Plant (600 MW(e)) is a CANDU type. In this work we will refered to the Atucha I spent fuel. The fuel of Atucha is 6 meters long and has 36 bars of zircaloy-4. For these fuels there are deposits located in the places where the reactors are situated. Atucha I has two pools houses where about 11 000 spent fuels is accumulated so far. From middle of the year 2000 the whole nucleus are of fuels with 0.85% enrichment and therefore, the consumption of fuel will be of 210 fuels per year. One of the alternatives is when the Atucha Nuclear Power Plants concludes their operation to pass the spent fuel to dry storage and another alternative is to continue with the operation of the pools and pass to dry a group of select spent fuel. The main objective is to elaborate a life management programme of the deposits of spent fuel inside the general life management programme of nuclear facilities. The life management plan of a facility is permanent and it is adjusted to the changes that may occur in the ageing mechanisms and the technological advances that allow them to be mitigated or solved. For the selection of the technology it should be kept in mind, those characteristic of the fuel element Atucha, their longitude and their burn up one that it will be different according to the time that is considered. The construction should also be to modulate, allowing toadd sectors according to the necessities. It should also be considered the conditions in which it can be carried out the transfer from the pools to the deposits in dry.

1. INTRODUCTION.

Nowadays, Argentina has two nuclear Power Plants in operation, Atucha I (CNA I), since 1974 and Embalse (CNE) since 1983, and one under construction, Atucha II. All of them are based on natural uranium as fuel with heavy water moderation.

Atucha I (370 MW(e)) and Atucha II (750 MW(e)) are almost unique in their type, both employ a Pressure Vessel type. The Embalse Nuclear Power Plant (600 MW(e)) is a typical CANDU plant. Since September 1994 the organisation of the Argentine nuclear activities, confirmed in April 1997, is as follows:

Nucleoeléctrica Argentina Sociedad Anónima (NASA), which operates the two NPP (Atucha 1 and Embalse).

Autoridad Regulatoria Nuclear (ARN) — National Regulatory Board of Nuclear Activities.

Comisión Nacional de Energía Atómica (CNEA) — National Atomic Energy Commission.

Within this scheme, one of the main activities to be undertaken by CNEA will be to provide technological assistance to NASA in problems concerning NPP operation. Works on life management of NPP are included in these activities.

Atucha I has two pools houses where the spent fuels is accumulated. From August of 2001 all the fuels of Atucha I nucleus are of Natural uranium dioxide, with slightly (0.85%) enriched U. This takes to that the burnt means of extraction passes from 5900 MWd/TU to 11 300 MWd/TU and the numbers of combustible elements burned per year from 430.3 to 230.

Fuel elements characteristics are:

- Fuel type: Natural uranium dioxide;
- Number of fuel elements: 252;
- Fuel rod array: Bundle;
- Number of fuel rods: 36;
- Number of empty rods: 1;
- Pellet diameter: 10.6 mm;
- Pellet length: 12 mm;
- Total weight: 210 kg;
- Total length: 6180 mm;
- Cross-section diameter: 108 mm;
- Fuel rod length: 5650 mm;
- Active length: 5295 mm;
- Cladding material: Zr-4;
- Fuel weight: 152.5 kg U/ assembly;
- Structural components (head and foot): Austenitic steel;
- Spacer: Inconel;
- Reloading of fuel elements: during full power service.

The handling, transport and storage of spent fuel elements is done under light water. In Pool House 1, the facility has three individual pools. One is used to assemble and disassemble fuel elements columns. The other two are decay or storage pools.

In Pool House 2, there are five pools: one for operation and the rest for storage. The watertight sheathing of the pools is built of steel plates and has concrete embedded supporting structure. It is designed for a temperature difference of about 35°C. The pools are equipped with cooling tubing and emergency spillways.

The sheathing plates are welded to the supporting structure and to the anchoring plates, and watertightness is tested. All parts in contact with pool water are made of stainless steel. Spent fuel elements are submerged, and they hang vertically in steel frame. They are cooled through natural circulation of the pool water.

The circuit of refrigeration of the pools of combustible elements has for purpose to eliminate the residual heat from the burnt combustible elements placed in the storage pools, heat that passes to the circuit of you dilute of the river by means of heat exchanger. The extraction of the water of the pools is carried out by means of two overflow systems for pool, this it dilutes it is taken to the coolers, those that reinject of cold water to the pools. The feeding pipe ends in the vicinities of the bottom of the pools, so that the flow in the pools this always guided from below up. In the coolers the pressure of the secondary side is bigger to avoid that activity transmission exists to the river water. The extracted water of the pools is subjected to a purification process in the sedimentation filter and after had cooled reinjected in the pool.

The one filled of water of the pools as well as the attache to replace lost takes place from the ion exchange system, being attache's system to replace lost regulated by floating regulation valves of automatic working. To avoid in cases of an eventual break of a pipe that the pools can loss water for siphon effect, the connexion has been foreseen to each pipe of having filled with a pipe of usually open ventilation. Also the system of water purity supplies the pipes of risk that they run in lathe of the pools; this pipes are good to dew the walls of the pools with ends of cleaning and reduction of the aerosol activity.

The purification system and cooling of the pools of burnt combustible elements is designed for to maintain the temperature of the storage pool below the 32°C, under normal conditions of operation, when the pool contains fuel accumulated along ten years of operation from the reactor to a factor of load of 80% and the cooling water is 22°C.

1) To maintain the temperature of the water in the pools of storage of failed fuel and reception of discharge of fuel below the 32°C, when the cooling water is 22°C;
2) To maintain the purity and cleaning of the water to allow the observation of the operations of handling of the burnt fuel under the water;
3) Proportional an appropriate capacity of extraction of radioactive material in suspension and dissolved of the water to allow the access to the work areas;
4) To maintain a level of water minimum in the pools to assure appropriate shielding during all the phases of the handling procedures and storage of burnt fuel;
5) To reduce to the minimum acceptable the blended of the water of the storage pools with that of discharge and reception of fuel. Also to reduces to the minimum the blended of the water of the pool of storage of fuel failed with that of the discharge pools and reception of fuel. This way the dispersion of material radioactive is avoided liberated by the fuel failed in all the water of the system.

Two cooling systems and essentially separates purification but with connection; one for the pool of main storage and another for the discharge pools, reception and storage of fuel. The system of the storage pool is designed primarily to eliminate the disintegration heat, as long as the elimination of fixed radioactivity the flows for the other pools. The objective of this design is to provide operative flexibility to satisfy varied purification necessities without making an unnecessary duplication of equipments.

To control the transport of the fuel elements, submarine spotlights have been installed in each pool, which can be adjusted vertically and horizontally. The handling system is made of a traveling crane with pools and devices to assemble and disassemble fuel elements columns. The traveling crane can be displaced along the entire pool zone.

2. METHODOLOGY

Therefore, the fundamental point here is to keep in mind that it is the effects of ageing due to time and use that cause net changes in the characteristics of a system, structure or component (SSC). We employ formal processes to systematically identify and evaluate the Critical Systems, Structures and Components (CSSCs) in the facilities. Afterwards, we apply a plan to ensure the facilities surveillance, operation, and maintenance programs, monitoring and control of the component degradation within the original design specifications, essential for the facilities life attainment.

A Technology Watch Programme is being established to ensure that degradation mechanisms, which could impact on facilities life, are promptly investigated so that mitigating programmes can be designed.

The methodology we will use for these studies is shown on Fig. 1.

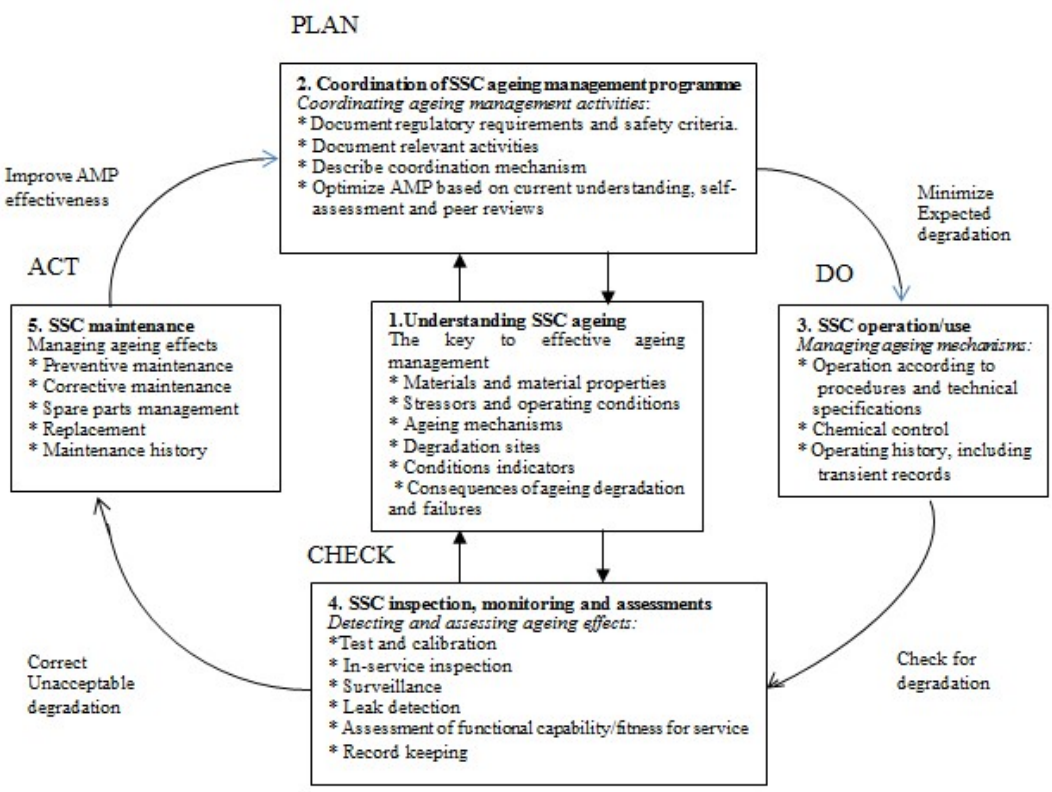

FIG 1. Ageing management methodology.

3. RESULTS

After analysis of both Atucha I pool houses, we obtained the following results:

Concrete wall stability: for all pool structures we carried out only visual inspection, no damage to fissures, ruptures or shelling material.

Integrity of concrete structure: the structure is in perfect state.

Pool lining: pool linings do not present signs of corrosion.

Integrity of metal structure: metal structure does not present signs of corrosion.

Degradation in storage racks: storage racks do not present signs of degradation.

Pipe failures: flaws of the system have not been reported.

4. CONCLUSIONS

The house of pools II can be operated independently of the reactor system, except in the case of the water of the water cooling systems. Therefore it would be necessary to carry out a modification in the facilities of the cooling water to be able to continue with their operation.

The house of pools I can be operates from the house II with few modifications in the systems. Both house are in very good conditions.

If the country decides for dry systems we think that the better solution is canister-based system. The canister is loaded with fuel in the reactor spent fuel pool. The canister is drained,

vacuum dried, decontaminated and pressurised with helium. The canister is transfer to a vertical concrete cask system and the casks may be enclosed in buildings. The system must be modify for the characteristics of the Atucha fuel, low burnup, large fuel (length: 6 meters) and large cooling time.

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SESSION 5

DRY STORAGE OF IRRADIATED NUCLEAR MATERIAL IN DUAL PURPOSE CASK AT THE JRC-ISPRA SITE

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Abstract

The Joint Research Centre (JRC) is currently involved in a Decommissioning and Waste Management (D&WM) Programme to progressively eliminate its historical liabilities related to the old nuclear installations and radiological waste generated in the past at Ispra site, *inter alia*: nuclear research reactors, hot-laboratories, radiochemical facilities and a variety of plants to treat and store the liquid and solid radiological waste. In the frame of the Programme, the Irradiated Nuclear Material (INM) accumulated over 40 years of research activities is foreseen to be retrieved, repacked and temporarily stored on-site by means of dry storage solution in one or more 'all-metal' Dual Purpose Cask (DPC).

1. INTRODUCTION

The inventory of remaining INM on the JRC Ispra site collected during several years of experimental campaigns is quite various, the nuclear material is mainly constituted by 'exotic' fuel items that were irradiated in experimental rigs with a broad range of burn-up during the '70s and '80s in the ESSOR research reactor located at the Centre. The material is indeed characterised by different enrichments, physical forms, claddings and dimensions. In addition, pins (and segments of them) coming from German and Italian BWR and PWR nuclear power plants, used to perform destructive analyses and partitioning of fission products and minor actinides with innovative solvent extraction processes, are included in the inventory of the INM to be managed by means of their transfer into DPC for on-site temporary storage before the definitive transfer to the Italian National Repository.

The INM items are stored in several locations: part of them are underwater in the reactor decay pond and part in dry pits inside hot-cell or shielded containers. The material can be divided on the basis of it the origin and use done during the past scientific campaigns. To simplify, the material can be divided in two main families: i) INM still included into experimental rigs; and ii) pins, segment of pins, pellets and residues of fuel.

1.1. Experimental rigs

The ESSOR research reactor was aimed to test new type of fuel. Therefore the reactor core allowed the insertion of nuclear material inside long instrumented tubes, namely rigs, able to provide data on the neutron irradiation, thermal and mechanical behaviour of non-conventional pins, fuel capsules and elements. Usually, the rigs have been longer than 5 meters with a short active part and low amount of nuclear material characterised by several enrichments, burn-up, cladding and dimensions. Hereafter are reported the three rig families still present in ESSOR decay pond are described.

1.1.1. CART

The rigs have been used to irradiate and perform post-irradiation analysis on CIRENE experimental fuel elements, innovative fuel developed during the eighties and similar to CANDU type fuel bundle, see Ref. [1]. In the decay pond there are 2 CART test rigs inside two storage containers with 5 fuel bundles of CIRENE type fuel each, 3 with nuclear material inside and 2 dummy bundles on the top and on the bottom part of the rig. CART rigs contain approximately 200 pins with natural uranium as fuel in Zircaloy 2 cladding and a burn-up ranging from 0–7 MWd/kg. The active fuel bundles have a total active length of 1500 mm divided in 3 segments of 500 mm and total bundle length of 2200 mm. In Fig.1 is reported the cutaway of the rig inside the storage container. It is envisaged to dismantle the rigs removing the 3+3 active fuel bundles and transfer them into dedicated storage containers.

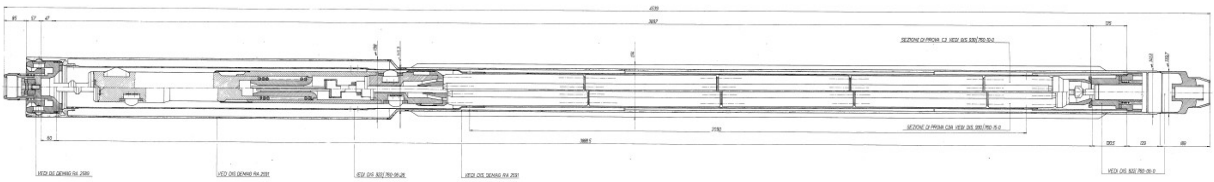

FIG.1. Cutaway of CART rig.

1.1.2. Colibri

The experiment denominated Colibri was intended to study the release of fission gas by HTGR type fuel under irradiation, Ref. [2]. The Colibri 1 rig (height 5500 mm, diameter 180-300 mm) is made of a stainless steel tube containing four Hastelloy C fuel capsules (height 20.8 mm, diameter 19.6 mm) positioned at different heights along the axis, which were irradiated at different temperatures (850–1150 °C). Each fuel capsule contains 2563 U-Th fuel coated particles, accommodated into 70 graphite discs. Each UO_2 and UO_2-ThO_2 microsphere is protected by a first layer of pyrolytic carbon (PyC), a second additional layer of silicon carbide (SiC) and an outer coating of pyrolytic carbon. The fresh uranium used for the microsphere had an enrichment comprised between 10% and 93%. In Fig. 2 shows the cutaway of the rig where the four Hastelloy capsules with nuclear material are identified with the letter A, B, C and D. During the dismantling operation the four sections with nuclear material will be separated from the rigs and divided before their transfer into special storage container.

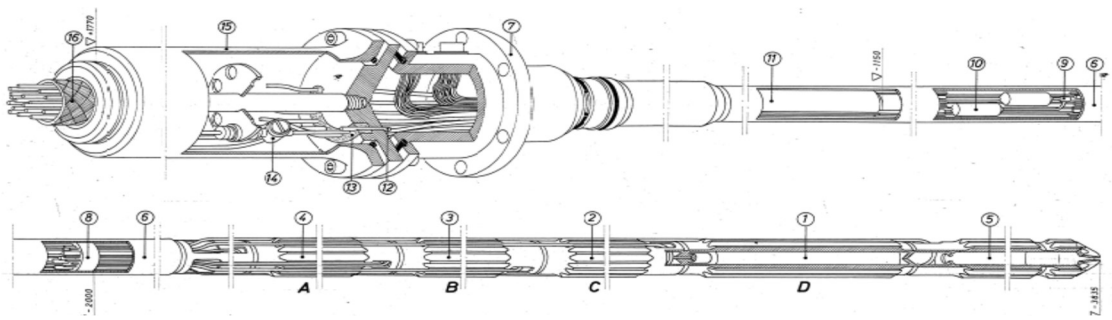

FIG. 2. Cutaway of Colibri rig.

1.1.3. Gioconda

There are 4 Gioconda rigs currently stored in the decay pond. The fuel capsules present inside each item consists of a Zircalloy-2 tube which holds the fuel column with the capsule placed in an outer stainless steel tube containing a metallic alloy of sodium-potassium (NaK, 44-56% by weight of potassium). The fuel capsule itself is approximately 210 mm long. There are two types of fuel used for the four experiments carried out in ESSOR reactor, the rigs marked with the numbers 1 and 2 contain natural uranium oxide pellets, those marked 5 and 6 contain natural uranium oxide microspheres [3]. The cutaway of Gioconda rig is reported in Fig. 3. A crucial aspect of the dismantling operation of the Gioconda is the safe retrieval and handling of the NaK to free the fuel capsules and remove completely residues of the very reactive alkali metal alloy before their storage in special containers.

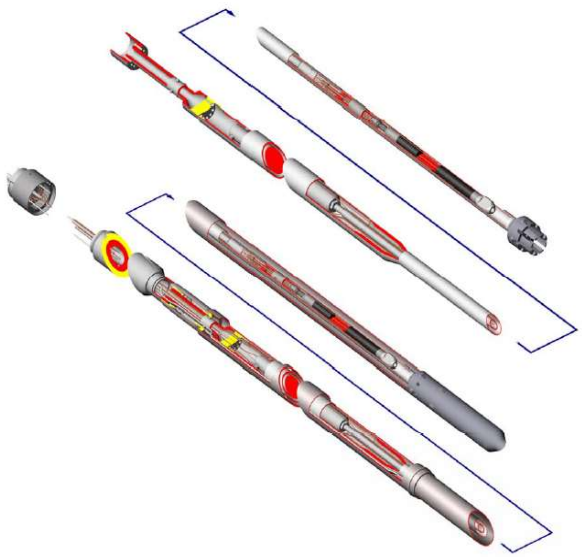

FIG. 3. Cutaway of Gioconda rig.

1.2. Spent fuel

Included in this family are a number of long whole pins coming from Trino (Italy), Obrigheim, Würgassen and Gundremmingen (Germany) nuclear power plants. Moreover, pellets and segment of pins generated by the cutting of the pins to allow sampling of spent

fuel for post-irradiation analysis performed during 1970s and 1980s at the JRC-Ispra or at the JRC-ITU [4, 5] are also present. The whole pins are currently stored in baskets and racks in ESSOR decay pond while fragments and pellets are placed in cans and boxes inside dry pits and shielded containers located in the hot-cell facility. The most challenging aspect regarding the repacking and storage of the spent fuel family in DCP are the physical dimensions of some whole pins, up to 4140 mm, and the presence of damaged spent fuel constituted by segments of pins and pellets. Such limitations will impact on the decision regarding the repacking operations to be performed to make the material suitable for a DPC and on the final decision concerning the type of DPC to be selected.

2. JRC STRATEGY

Currently there are two ongoing projects at the JRC-Ispra site to implement the preliminary repacking and subsequent transfer of INM in DPC. They can be summarised as follows:

- The first step is aimed to the retrieval, dismantling, repacking and transfer to a temporary buffer store called Transit Safety Area, TSA, the nuclear material inside the experimental rigs and the spent fuel in terms of whole pins, segments of pins and pellets. The TSA is a revolving rack located in a dedicated shielded cell able to host up to 24 cylindrical containers, namely pots, as show in Fig. 4. The INM will be retrieved from the current storage locations (i.e. dry pits in hot cell facility and reactor decay pond) and repacked in to pots before the transfer to the TSA. The rig dismantling activities and the repacking of the items will be performed by means of bespoke tools inside and equipped hot-cell close to the TSA (cell 4411 in Fig. 5); the transfer of the loaded pots to the TSA and loading operations will be carried out via a remote shielded transfer package which can be coupled to a cells docking ports; the transfer, one by one, of the pots from the working cell 4411 to the TSA cell via the cell transfer 4304 will proceed as presented in the scheme reported in Fig. 5. The INM will remain in the TSA for an estimated period of 5-10 years, time necessary to design, license, manufacturing the cask and to construct the facility for the on-site storage of the cask/s at the JRC Ispra site;

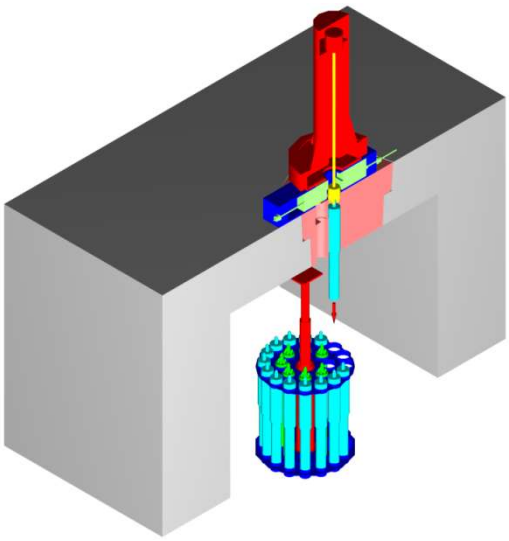

FIG. 4. 3D simulation of TSA loading operation with shielded transfer bell.

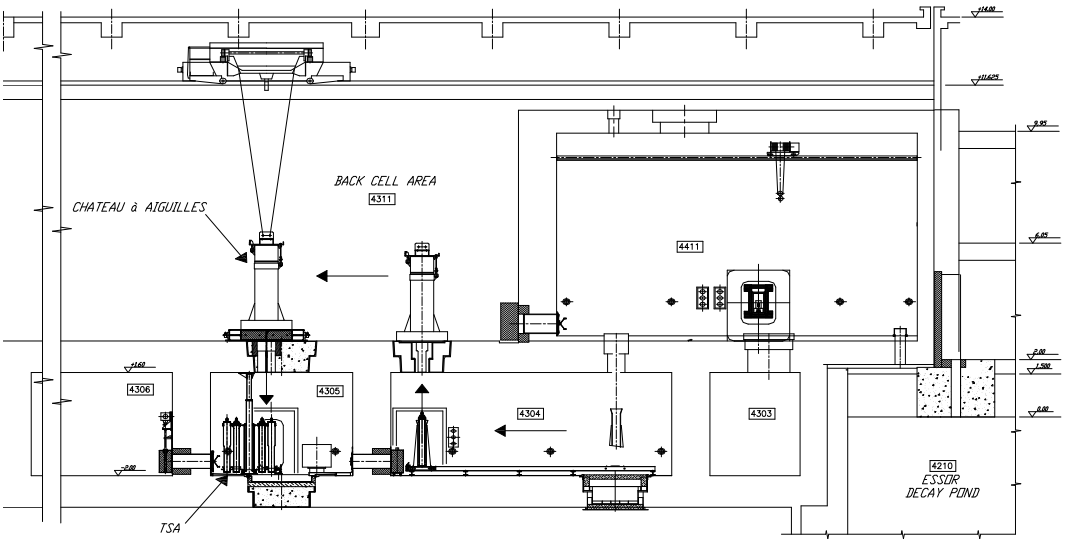

FIG. 5. Scheme of TSA loading with pots coming from working cell 4411.

- The second part of strategy implementation foresees the supply of the needed DPC/s to which the material inside pots temporarily stored in TSA will be definitively transferred. The loaded cask/s will be moved to the on-site storage facility. Currently one or two DPC are envisaged to store all the INM in accordance with one of the following configurations: *i)* all the material repacked in 23-24 pots or *ii)* in 19-20 pots plus a basket for the whole pins in case it will be possible to store the long pin in a cask as they are without cutting. As regards to the temporary storage facility to which the DCP/s will finally be transferred, two main options are still under discussion in order to identify the facility around 400 m far from ESSOR reactor in the radioactive waste management area of the Centre (Fig. 6). The options can be briefly summarised as follows: *a)* the DPC will be stored in a new dedicated facility; *b)* the DPC will be stored in a facility with other ILW or HLW. In both cases the DPC and the facility will be designed and built taking into account safety requirements foreseen by Ref. [6-8] and the security prescriptions for Category II nuclear material or higher in accordance with Ref. [9].

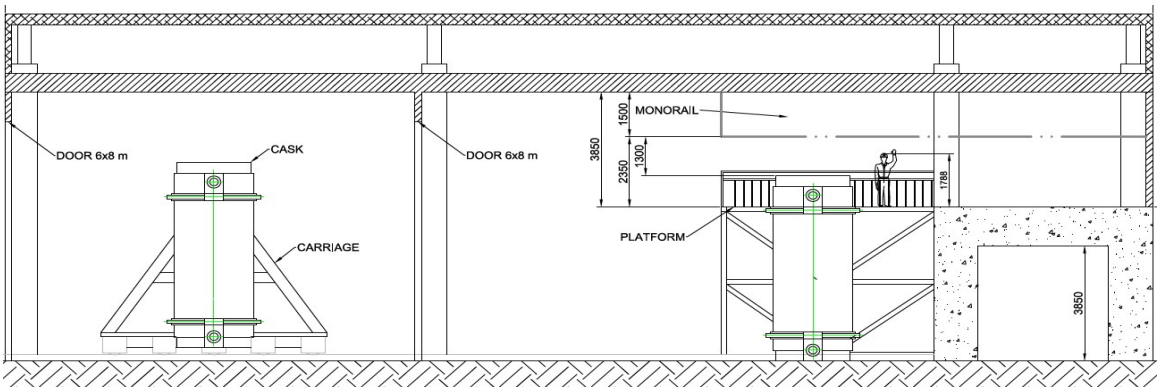

FIG. 6. Section of the on-site DPC storage facility.

3. CONCLUSIONS

The removal of the irradiated nuclear material from the facilities where it is currently stored and the subsequent transfer to a new dedicated temporary store is a critical activity to be fulfilled for the prosecution of the entire decommissioning program at the JRC Ispra site. The removal of the INM from ESSOR facility will allow the beginning of the dismantling operations of the reactor and ancillary installations which represent around 40% of the entire commitment of the programme. The most challenging aspects to be faced by the JRC to implement the strategy can be summarised as follows: i) remote dismantling of the rigs to remove and repack the nuclear material inside of them; ii) taking into account that the JRC intends to use state-of-the-art pre-designed DPC body with a bespoke inner canister that shall be able to fit INM features, to identify appropriate cask/s able to host various types of fuel in terms of physical and chemical features, integrity of the INM items and length of the pins; and iii) due to the infrastructural constrains of ESSOR facility, the dry loading of the DPC outside the decay pond will be mandatory and it will be performed by means of existing equipment like shielded transfer package and bespoke tools to couple and index the pots with the internal basket of the cask. The above mentioned technical aspects will be deeply assessed and carried out in the next 5−7 years to respect the time, licensing and budget constraints of the JRC decommissioning program.

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PROGRESS TOWARDS A SOLUTION FOR INTERMEDIATE STORAGE OF SPENT NUCLEAR FUEL IN NORWAY

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Abstract

Spent Nuclear Fuel (SNF) in Norway has arisen from irradiation of fuel in the NORA, JEEP I and JEEP II reactors at Kjeller, and in the Halden Boiling Water Reactor (HBWR) in Halden. In total there are some 16 tonnes of SNF, all of which is currently stored on-site, in either wet or dry storage facilities. The greater part of the SNF, 12 tonnes, consists of aluminium-clad fuel, of which 10 tonnes is metallic uranium fuel and the remainder oxide (UO_2). Such fuel presents significant challenges with respect to long-term storage and disposal. Current policy is for the fuel to be stored for a period of at least 50 years. In the meantime a national final disposal facility should be constructed and taken into operation. Several committees have advised the Government of Norway on, among others, policy issues, storage methods and localisation of a storage facility. Both experts and stake holders have participated in these committees. As the next stage in the process, a 'Concept evaluation' study was completed in early 2015. This paper presents an overview of the spent fuel in Norway and a description of current storage facilities. The prospects and plans for long-term storage are then described, including a summary of recommendations made to government, the reactions of various stakeholders to these recommendations, the current status, and the proposed next steps. It has been recommended that the aluminium-clad fuel be reprocessed in an overseas commercial facility to produce a stable waste form for storage and disposal. This recommendation is controversial, and a decision has not yet been taken on whether to pursue this option. An analysis of available storage concepts resulted in the recommendation to use dual-purpose casks. A further recommendation was that a public organisation, independent of the producer of the spent fuel, be founded to manage the SNF and that this organization also should have the responsibility for managing radioactive waste in Norway. Funding and operation of this organisation should be based on the principle that the polluter pays.

1. INVENTORY OF SPENT FUEL IN NORWAY

Spent nuclear fuel (SNF) in Norway today has arisen from three heavy water research reactors: JEEP I, which was operated at IFE's Kjeller site from 1951 to 1967; JEEP II, also located at Kjeller, and in operation from 1966 to the present; and the Halden Boiling Water Reactor (HBWR) in Halden, which commenced operation in 1959 and is still in use. (Another reactor, NORA, was operated at Kjeller from 1961 to 1968. The fuel, which was identical to that used in JEEP I, was returned to the USA, and thus is not discussed further in this paper.)

Some spent fuel from JEEP I was used in a pilot reprocessing plant at the Kjeller site, which was in operation from 1961 to 1968, and later decommissioned. The second core loading of the HBWR was reprocessed in Belgium in 1969. The recovered uranium and plutonium was sold for civilian use, and the waste was disposed of in Belgium. With these exceptions, all Norwegian spent fuel is currently stored at Kjeller and in Halden.

The JEEP I and the HBWR 1st charge fuels were metallic uranium clad in aluminium, while the JEEP II fuel is 3.5% enriched UO_2 clad in aluminium. With the exception of the first charge fuel, the majority of the HBWR fuel is UO_2 clad in Zircaloy, and is thus similar to commercial light water reactor fuel.

The JEEP 1 fuel was, as mentioned above, all discharged by 1967, with typical burn-ups in the range from 200 to 400 MWd/tU.

On average, JEEP II fuel elements are irradiated for 10 years, to a burn-up of 15 000 – 16 000 MWd/t U. The JEEP II core contains about 220 kg of fuel, and approximately 45 kg per year is discharged.

The HBWR first charge fuel was irradiated from 1959 to 1962, for 1000 hours at low temperature and power, and the average rod burn-up was approximately 12 MWd/t U. Currently, the majority of the HBWR fuel is discharged after reaching a burn-up of approximately 30-40 000 MWd/t U. The core contains approximately 0.4 tonnes of uranium, and approximately 80 kg per year is discharged.

There is in addition spent experimental fuel from the research program conducted in the HBWR. Although the amount is small, this fuel will present challenges in its future management as there are several types of fuel, with varying dimensions. This experimental fuel is not discussed further in this paper.

In total, Norway has some 16 tonnes of spent fuel, of which six tonnes are stored at Kjeller and 10 tonnes in Halden. There is approximately 12 tonnes of aluminium-clad fuel, of which 10 tonnes is metallic uranium fuel and the remainder oxide (UO_2). Details are given in Table 1.

TABLE 1. INVENTORY OF SNF IN NORWAY

	JEEP I	JEEP II	HBWR 1st charge	HBWR
Fuel material	Metallic U	UO_2	Metallic U	UO_2
Cladding	Al	Al	Al	Zircaloy
Enrichment, wt%	Natural	3.5	Natural	6
Discharge burnup, MWd/t U	1 - 1000	15 000	12	40 000
Irradiation period	1951 - 1967	1966 - present	1959 - 1962	1962 - present
Total mass U, tonnes	3.1	2.0	6.7	3.5
New arisings, kg/year	0	45	0	80
Current storage	Dry, IFE Kjeller	Dry, IFE Kjeller	Dry, IFE Halden	Wet, IFE Halden

2. ORGANISATION AND RESPONSIBILITIES

All four research reactors in Norway together with the associated spent fuel storage facilities have been operated by Institutt for energiteknikk (Institute for Energy Technology, IFE), which has been an independent foundation since 1980. Previously, from its establishment in 1948, the organisation was state-owned and operated as Institutt for atomenergi (Institute for Atomic Energy, IFA).

Current policy is that existing spent fuel will, as far as possible considering its suitability for later direct disposal, be stored until final disposal is possible [1]. However, while there is capacity in the current storage facilities for spent fuel arisings for several years into the future,

there is recognition that the current storage arrangements are not appropriate for long-term intermediate storage and that they do not constitute an endpoint for the fuel, and thus that other facilities and/or routines will be required.

State funding for the Norwegian research reactors is managed through the Ministry of Trade, Industry and Fisheries (Nærings- og fiskeridepartementet), previously the Ministry of Trade and Industry (Nærings- og handelsdepartementet). Thus, it is also this organisation that has taken the lead in the definition of arrangements for the long-term management (i.e. intermediate storage and final disposal) of spent fuel in Norway. The Ministry has appointed committees comprised of various stakeholders and technical experts to provide independent advice and make recommendations. A broad cross section of stakeholders is included in the committees, including local government, community representatives, Non-Government Organisations, experts in geology, safety analyses and SNF management, and the Directorate of Public Construction and Property (Statsbygg). These recommendations, if accepted, will then form the basis of government policy. At present, however, the organisational and financial arrangements for long-term spent fuel management have not been defined.

3. CURRENT SPENT FUEL STORAGE ARRANGEMENTS

The aluminium-clad spent fuel from the three reactors is currently stored in separate dry storage facilities.

After discharge, JEEP I fuel was initially stored in aluminium baskets in a wet storage facility for periods up to 10 years. Transfer of the fuel to the JEEP I dry storage facility started in 1961, and the final transfer took place in 2000. All fuel elements were inspected and repacked into tight stainless steel baskets in 1982. The JEEP I dry storage facility consists of top and bottom concrete slabs, with the top one at ground level and the bottom approximately 3 m underground. There are 97 vertical holes in each block, and in each hole is positioned a 2.7 m long stainless steel pipe (to hold the aluminium baskets), which is fixed into the 20 cm thick slab. The top of each tube is covered with a metal cap and an expansion gasket. The space around the tubes between the two slabs is filled mainly by sand. The storage block is covered by a free-standing building.

Spent fuel from JEEP II is stored in water-filled wells (with a purification system) for six to 12 months. The fuel elements are then packed into tight stainless steel cans before transfer to the dry storage facility. This consists of a concrete block with its top at ground level, located beneath a building specifically designed for loading and unloading of transports of radioactive material. The concrete block drains through a pipe to a delay tank. The block houses 84 vertical steel tubes, of depth 3–3.5 m. After insertion of the stainless steel cans, the tubes are sealed with lead plugs.

The 1st charge HBWR fuel was placed in wet storage for approximately three years. In 1962, the rods were dried and loaded into aluminium storage capsules. Inspection showed that all the fuel rods were intact. These capsules were loaded into the dry storage facility, which consists of a concrete shell with 2 m thick walls, with a 1 m thick front shield of reinforced concrete. There are 202 holes in the front, in which 7 m long steel tubes are fitted horizontally. The aluminium storage capsules are placed in the steel tubes. A metal frame to support the tubes is fitted inside the concrete shell. The steel tubes are cooled by natural circulation of air. This facility, together with wet storage ponds, is positioned inside a building fitted with a ventilation system.

The Zircaloy-clad UO_2 fuel from the HBWR is stored in one of two wet storage facilities at the reactor site, each of which consists of steel-lined pits. A forced-circulation water loop maintains the required temperature, and includes a purification system to maintain water quality.

The capacity of the storage facilities will be reached around 2024 in Halden and in 2032 at Kjeller if the current operations continue. These dates would be brought forward by 3 – 4 years if buffer capacity is required for removal of fuel from the reactors in case of an incident or accident.

4. PROCESS TO 2013

A previous paper [2] described the process to 2013. At that time, the most recent recommendations had been made by the 'Stranden-committee', comprised of a wide range of stakeholders and technical experts, with the mandate to find the most suitable technical solution and localisation for intermediate storage for spent nuclear fuel and long-lived waste [3]. (One of the authors of this paper (EL) was the committee's Technical Secretary.)

Since approximately three-quarters of the spent fuel consists of metallic uranium fuel and/or fuel with Al cladding, and with the recognition that this fuel presents significant challenges because of its chemical reactivity and physical instability, a 'Technical Committee' (comprising solely of technical experts) was appointed to recommend methods to condition such fuels to render them eligible for interim storage and final disposal [4]. One of the authors of this paper (PB) was chairman of this committee, and the other members were from IFE, Studsvik Nuclear AB (Sweden) and the International Atomic Energy Agency (IAEA).

Following the issue of the recommendations of these two committees, the Ministry of Trade and Industry arranged for a wide-ranging public consultation. The public in general were invited to give comments on the recommendations, while many organisations were specifically requested to do so. These organisations included relevant government ministries and directorates; local government in the communities which host the research reactors and current storage facilities, and which were identified as possible hosts for a new interim storage facility; the Norwegian Radiological Protection Authority (Statens Strålevern); IFE and other research establishments; and self-identified environmental groups, including Friends of the Earth Norway (Norges Naturvernforbund) and Greenpeace [5]. In the following, the main recommendations of these two committees are summarised together with the main reactions to them given by various stakeholders.

Although the amount of SNF in Norway (16 tonnes) is very small in comparison with countries with commercial nuclear energy generation, the majority of the fuel presents significant challenges because of its chemical reactivity and physical instability and is therefore unsuitable in its current form for storage and deposition. It was therefore recommended that the fuel should be stabilised before being transferred to an interim storage facility and subsequently to a final repository. This stabilisation should be done immediately: long term storage with postponed decision on final treatment (often called "wait and see") is unethical as it places a burden on future generations. The only proven and available stabilisation process is reprocessing. Further, it is neither cost-effective nor environmentally sound to construct a domestic reprocessing facility for the small amount of SNF, and thus a contract is required with a commercial supplier. In practice, there are two options: AREVA's Cap La Hague facility in France, and the Mayak facility in the Russian Federation (the Technical Committee was not authorised by the terms of its mandate to consider UK facilities at Sellafield as an option).

The majority of those who responded had no comments on the proposal to reprocess. However, some environmental groups, including Friends of the Earth Norway and Greenpeace, oppose reprocessing under all circumstances and have demanded that this recommendation be rejected. Others, including the Norwegian Radiological Protection Authority, are of the opinion that it needs to be shown that there are no other options to reprocessing. However, no other realistic technical solution apart from "wait and see" has been proposed by these stakeholders as an alternative.

Dual purpose storage and transport casks were identified as the most suitable storage concept. The main reason behind this recommendation was that the fuel would be ready for transport to a repository at the end of the intermediate storage period. It was also considered an advantage that the casks are autonym and therefore not dependent on external systems or structures to maintain safety, security and radiation protection. This would give a large degree of flexibility if it was found necessary to move the storage to another site in the future. There is wide experience in the use of such casks, both for in- and out-doors storage. They are suitable for storage both of spent fuel assemblies and the products of reprocessing, including uranium product, PuO_2 and vitrified high level waste.

Building a storage facility inside a mountain tunnel would give several advantages compared with an outside building, although both alternatives were considered acceptable solutions. A tunnel facility would give a high level of protection against external impacts, stable storage conditions, low energy costs and low maintenance costs. Investment costs for a tunnel facility were considered to be comparable to a standardized industrial building and at least a factor three lower than the costs of a building designed to withstand a high level of external impact.

Because the quantity of spent fuel is very limited, a new facility should be made to store both spent fuel and long lived radioactive waste (of which Norway has a minor quantity consisting of non-irradiated uranium, used long lived radioactive sources), as well as the products arising from the proposed reprocessing of metallic uranium / Al-clad fuel.

The primary recommendation was to establish the new intermediate storage facility on the site of the Halden reactor. Alternative locations were a site approximately 1 km outside the HBWR reactor site, a site 10 km from the JEEP II reactor site and a site 60 km from both sites.

All sites were selected based on a geology and topology suitable for a mountain tunnel. Additional criteria were road standard, availability of infrastructure and potential conflict with other interests. All the proposed sites were considered to have a low value for alternative applications, including as recreation areas.

Without exception, local government in the communities identified as possible hosts have rejected the recommendation. Those not currently hosting nuclear facilities are against the establishment of such facilities in their communities. Perhaps more surprisingly, the communities that currently host the research reactors and associated storage facilities also rejected the building of a new storage facility on the current sites even though there is nearly 60 years' experience of spent fuel storage and that the primary advantage of a new facility based on a cask concept would be improvement of barriers compared to the existing facilities and that it would be well protected against extreme external impact.

It was recommended that the organisational and financial framework for an interim storage facility should be defined as soon as possible by the government, including definition of the financial responsibility for legacy and future waste arising. A public sector organisation,

independent of the producer of the spent fuel, should be founded to manage the SNF, and this organization also should have the responsibility for managing radioactive waste in Norway. Funding and operation of this organisation should be based on the principle that the polluter pays.

IFE, as the operator of the research reactors, had no objection to these principles, but is of the opinion that costs for management of legacy fuel should be borne by the state since the then governments made the decisions to build, commission and operate the research reactors. Furthermore, IFE, as a non-profit organisation, cannot meet the costs of legacy fuel management.

5. RECENT DEVELOPMENTS

In 2012, six metallic uranium fuel rods with aluminium cladding were inspected: five rods from the HBWR 1st charge and one rod from JEEP I. The five Halden rods were intact with no signs of significant degradation or corrosion. However, the one JEEP I rod, which was irradiated from 1951 to 1953 to a burnup of 0.13 MWd/t U, contained an internal blister of uranium hydride (UH), which covered almost five per cent of the total uranium cross section. The cladding was intact, and it is considered that the blister was formed as a result of water penetration during irradiation or initial wet storage. Detailed information on these investigations is given in a companion paper at this conference [6].

The identification of this defect reveals the urgency of making a decision on the long term management of the fuel before further deterioration occurs, and which may reduce or make more difficult the available options. As a further result of this finding, a project is underway to investigate all JEEP I and HBWR 1st charge fuel to register its condition and evaluate the sufficiency of the current storage methods and facilities. This project is being conducted independently of the government-led evaluations detailed below, but its findings may influence the overall process depending on the state of the fuel and storage facilities.

All Norwegian government projects with a budget of over 750 million Norwegian Kroner must be evaluated in line with quality assurance requirements for major public investment projects. In March 2013, the Ministry of Trade and Industry issued a call for tender for two "Concept evaluation" studies (Konseptvalgutredning, KVU) on handling of spent fuel and on future decommissioning of the Norwegian nuclear reactors. A concept evaluation study is an iterative process where the aim is to recommend one concept to meet a specified goal. The recommended concept is based on identified needs, goals and requirements, and includes identification and evaluation of possible concepts, together with uncertainty analyses of selected options and an assessment of the benefits to different stakeholders against the cost of the investment.

The specific questions to be addressed in the spent fuel study were: (i) which solutions provide the greatest socio-economic benefit to society; and (ii) what is important to focus on in the further planning to implement the chosen solutions? The study was carried out by DNV GL AS, together with Studsvik Nuclear AB, Westinghouse Electric Sweden AB, Quintessa Ltd and SBF — Centre for Applied Research at NHH. The same group, with the exception of Quintessa Ltd, prepared the study on decommissioning, and the two studies were conducted in parallel with significant information exchange. The report was issued in January 2015 [7], and the main recommendations are summarised below.

Five alternative strategies for management of SNF were identified: (i) reference alternative (i.e. continued storage of SNF at Kjeller and Halden); (ii) storage of all SNF in one location in

Norway; (iii) a repository in Norway; (iv) international cooperation on a repository; and (v) reprocessing of all spent fuel. All of the three latter alternatives include reprocessing. It was also stated that the recommended strategy would depend on the duration of the operation of the reactors and whether reprocessing is acceptable to the Norwegian authorities.

From the results of the analyses and an evaluation of the time horizon, the group concluded that reprocessing of all spent fuel is the recommended solution for managing spent fuel under the assumption that both reactors are shut down over the next few decades, and that reprocessing is still a commercially available service.

A further recommendation was that spent fuel should be stored in as few locations as possible to minimize costs for monitoring, operation and maintenance. Also noted was the need to establish a process for finding the location of a storage facility/repository within municipalities that could potentially host such a facility, built on mutual trust and respect. The government should consider what incentives can stimulate voluntarism and which can provide new value for a host municipality and the developer and owner of the new facility.

6. DISCUSSION

Over the past five years, three recommendations have been made to the Norwegian authorities to reprocess spent fuel (from the Technical Committee, Stranden Committee and the concept evaluation study). It has also been stated repeatedly that the authorities need to make a decision on whether reprocessing is an acceptable solution.

The Technical Committee's mandate was limited to the aluminium-clad fuel, and thus it did not make any recommendations concerning the UO_2 fuel with Zircaloy cladding. Subsequently, the Stranden Committee specifically excluded Zircaloy-clad fuel from its recommendation to reprocess as it considered direct disposal to be an adequate disposition option for this fuel type. However, the group that produced the concept evaluation study found that the greatest socio-economic benefit to society would be to reprocess all the spent fuel as this would simplify subsequent storage and disposal.

The next step in the official process is quality checking of the concept evaluation study. This is done in two stages, so-called KS1 and KS2. KS1 undertakes socio-economic analyses of the different identified options, while KS2 includes quality assurance of a more precise cost estimate for the chosen concept. At the time of writing this paper, the schedule for these steps is unknown.

7. SUMMARY

Spent Nuclear Fuel (SNF) in Norway has arisen from irradiation of fuel in the NORA, JEEP I and JEEP II reactors at Kjeller, and in the Halden Boiling Water Reactor (HBWR) in Halden. In total there are some 16 tonnes of SNF, all of which is currently stored on-site, in either wet or dry storage facilities. The greater part of the SNF, 12 tonnes, consists of aluminium-clad fuel, of which 10 tonnes is metallic uranium fuel and the remainder oxide (UO_2). Such fuel presents significant challenges with respect to long-term storage and disposal.

Current policy is that existing spent fuel will, as far as possible considering its suitability for later direct disposal, be stored until final disposal is possible. Several committees have advised the Government of Norway on, among others, policy issues, storage methods and localisation of a storage facility. Both experts and stake holders have participated in these

committees. Following the work of these committees, the results of a concept evaluation study have been recently published.

A common recommendation is to reprocess the spent fuel. While the Technical and Stranden committees limited this recommendation to the aluminium-clad fuel, the concept evaluation study recommended reprocessing of all spent fuel.

While the Stranden committee recommended that a new interim storage facility should be constructed at the site of the Halden Reactor, the concept evaluation study has left open the question of location, instead recommending the establishment of a process for finding the location of a storage facility / repository within municipalities that could potentially host such a facility, built on mutual trust and respect.

It is also recommended that a public sector organisation be founded to manage the SNF, independent of the producer of the spent fuel, and that this organization also should have the responsibility for managing radioactive waste in Norway. Funding and operation of this organisation should be based on the principle that the polluter pays.

A wide-ranging public consultation exercise has been held. The recommendation to reprocess is controversial, and self-identified environmental organisations have been voracious in opposing it. While the principle of a new interim storage facility is accepted (also by the environmental organisations), local government in each of the communities identified as possible hosts has rejected the recommendation for the location of the facility.

The identification of a uranium hydride blister in a JEEP I fuel rod reveals the urgency of making a decision on the long term management of the fuel before further deterioration occurs, and which may reduce or make more difficult the available options. A project is underway to investigate all metallic uranium fuel with aluminium cladding to register its condition and evaluate the sufficiency of the current storage methods and facilities. This project is being conducted independently of the government-led investigations.

The next steps in the process is quality checking of the results of the concept evaluation study. The schedule for this is unknown at the time of writing.

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FIRST EXPERIENCE IN BASIC DESIGN OF A DRY STORAGE FACILITY FOR SPENT FUEL IN ARGENTINA

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Abstract

This paper is a description of the technical option adopted for a basic and detailed engineering design of a spent fuel dry storage in Argentina, called ASECQ (acronym in Spanish: Almacenamiento en Seco de Elementos Combustibles Quemados). The purpose of ASECQ Project is to develop an innovative and safe long term approach to the management of Spent Fuel Element (SFE), solving the necessity of storage capacity of the Atucha I NPP facilitating the Operator, Nucleoeléctrica Argentina Sociedad Anónima (NA-SA), transfer the SFE from wet to dry storage and the National Regulatory Authority (ARN) control the operation inside the proper Plant, fulfilling the nuclear surveillance and safeguards. A first progress report of the conceptual engineering of ASECQ Project was presented by NA-SA at the 2010 International Conference on Management of Spent Fuel from Nuclear Power Reactors.

Topics: Spent Fuel Element, On-Site Dry Storage Option, Design of Contention Barriers, Basic Engineering for Nuclear Facility.

1. INTRODUCTION

Atucha I NPP located in Lima, Buenos Aires Province, was started up in 1974 initiating the nuclear energy production in Argentina. It is a PHWR, pressure vessel type of 357 MWe. It was fuelled with natural Uranium during its first 21 years of operation; the average burnup of the spent fuel was approx. 6000 MWd/tU. Between 1995 and 2000, fresh fuel was gradually modified from natural to slightly enriched uranium (0,85 % nominal).

With the purpose of solving the current situation that both fuel pools capacity will be soon complete (Pool Buildings I and II) and considering the future life extension of the Plant, CNEA was asked to develop a project, integrating systems in a new building for dry storage of spent nuclear fuels.

ASECQ Project was carried out considering the requirements of the National Program for Radioactive Waste Management (PNGRR in CNEA) and the recommendations of IAEA through a review mission called "Peer Review Mission IAEA IFMAP CNAI Spent Fuel Dry Storage Project" held from 12 to 16 March 2012 in Atucha I NPP site. This revision was asked by Nucleoeléctrica Argentina Sociedad Anónima (NA-SA) in the framework of Irradiated Fuel Management Advisory Programme.

The Operator is planning the life extension of the plant because the end of life by design will be reached when finish 32 full power years.

ASECQ Project provides the transfer of longer decay Spent Fuel Element (SFE) of natural Uranium deposited in the station storage pools to dry storage in a controlled area and it was focuses on Pool Building I enlargement, in order to build a vertical underground SFE storage.

ASECQ Project is currently under construction and its starting up is planned for 2018.

2. TECHNICAL JUSTIFICATION

Different alternatives of storage have been evaluated in order to solve the necessity of increasing capacity. After a cost-benefit analysis, it was decided to carry out a dry storage as a controlled area extension of Atucha I NPP, considering it like a permanent modification of the installation. See Fig. 1.

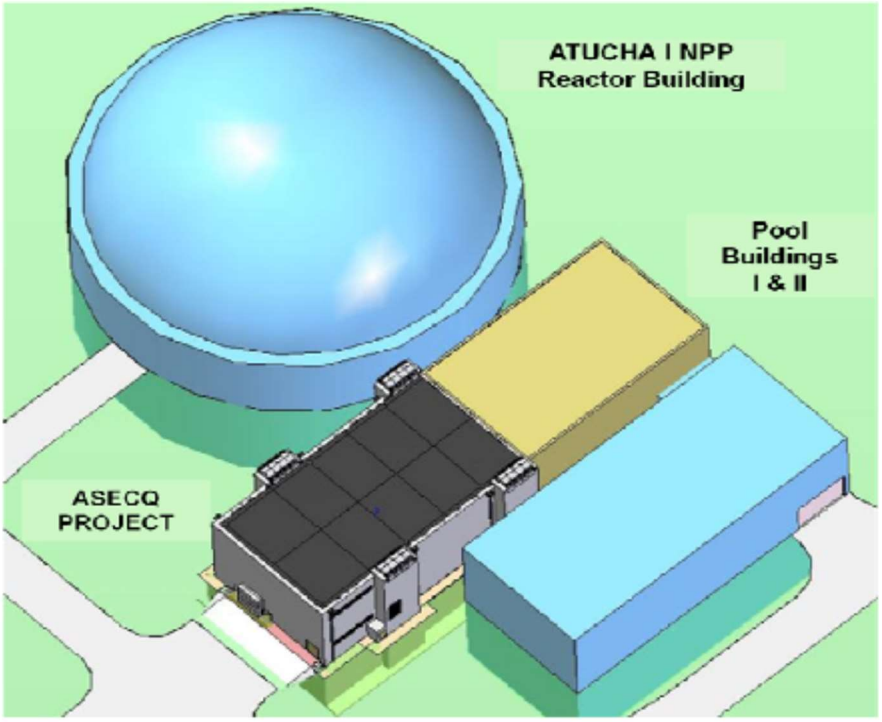

FIG. 1. Location of ASECQ Project in Atucha I NPP.

To avoid interferences with the normal operation of the Atucha I NPP, that at the moment uses the Building Pool II with fuels that come from the Reactor, and considering that the fuels selected for dry storage are those of more antiquity located in the Pool Building I, the new facility for storage is built as an amplification of the existing building pool.

Some relevant aspects of ASECQ Project are:

- It was selected between other valid alternatives, having analyzed the advantages and disadvantages of different options (finally it was selected: on-site storage, dry storage and shielding silo type);
- In the design it was considered: the removal heat process, the shielding structure, the criticality conditions, the integrity and confinement of the SFE;
- It is a great experience in development and implementation, engaging many interested parts in Argentine nuclear sector;
- It is an approach in developing a long-term strategy, and it will allow a better surveillance and monitoring of SFE;
- It requires the latest technology and the best national companies cooperation;
- It will use the existing crane of Pool Building I and other systems, already validated and operative for transfer in vertical position of SFE;

- It will be also possible to return the SFE to the Pool in order to be removed for further Management;
- It will be operated applying controlled area procedures of Atucha I NPP, upgrading only those when necessary, in order to include the operation and control of ASECQ facility;
- It will use as an extension the systems and services that are already in operation such as the normal electric supply, the secured electric supply, the search people, the phone communication, the compressed air supply, the drainage and the ventilation system;
- This configuration will facilitate NA-SA transfer the SFE and the National Regulatory Authority (ARN) control the operation inside the proper plant, fulfilling the nuclear surveillance and safeguards;
- It can be built in simultaneous with the operation of the NPP;
- Once ASECQ will be finished, the installation can be operated and maintained by the same qualified personnel of plant, with special training and experience in spent nuclear fuel management;
- According to Argentine Standard (AR 0.0.1), ASECQ Project is classified as a relevant nuclear facility class I;
- The original design lifetime of ASECQ Project was established in 50 years.

3. MAIN REQUIREMENTS

ASECQ Project, including all their associated Structures, Systems and Components, should guarantee a reliable operation, minimizing the probability of occurrence of human errors or accidents. See Fig. 2.

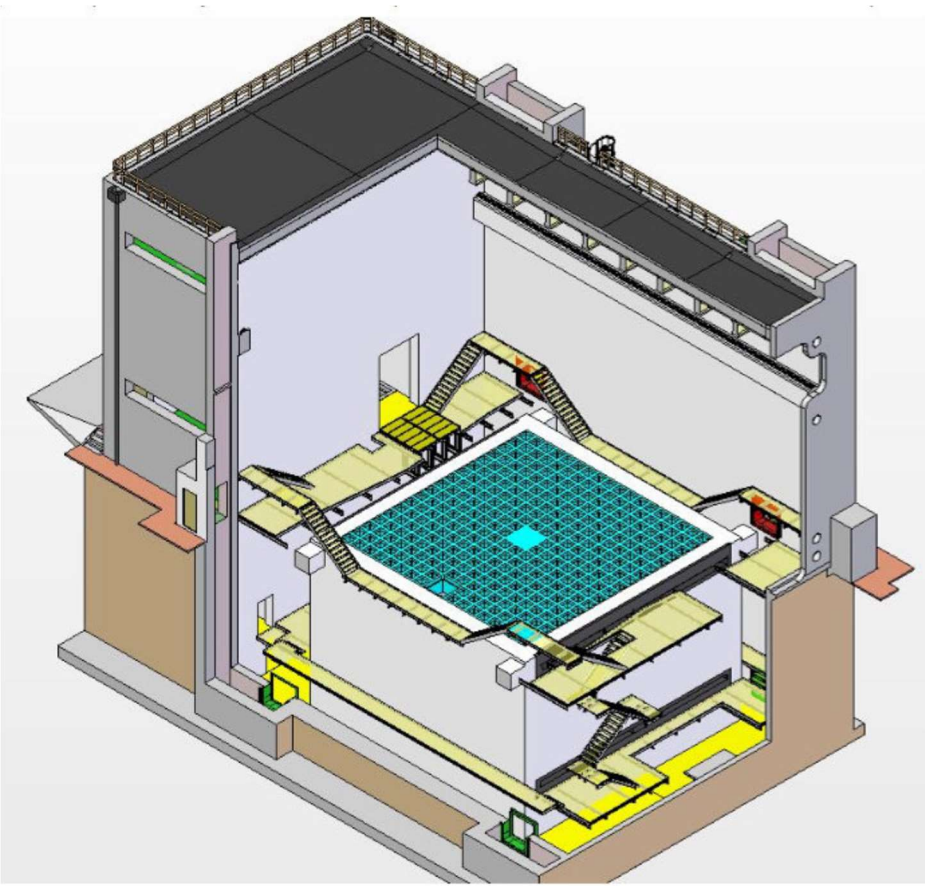

Fig. 2. The ASECQ Project.

For the design and basic engineering of ASECQ, national and international nuclear safety standards have been followed to assure the protection of operators and the environment. See Refs 1 to 14.

ASECQ Project should reach some specific objectives:

- To constitute a physical contention barrier between SFE and the environment, verifying that the structures and systems will respond satisfactory as well in internal as external events postulated;
- The grade of tightness of the building should always allow its normal operation; and be reliable and sure for the personnel and the plant;
- The intrinsic design, or the geometry of structures, system and components, should eliminate the generated residual heat into the silo through a simple mechanism with redundant characteristic, achieving the passive form under the basic principles of thermodynamic / physics, this is:
  - To operate under normal conditions with natural convection, once completed the SFE transference into the Silo Building;
  - To operate under ventilation forced mechanisms during charge and discharge processes of the Storage Units (Unidades de Almacenamiento in Spanish) into the Silo Units or under emergency events.
- To be prepared to resist a possible internal flooding successfully in an acceptable limit, without affecting components for the operation of the installation;
- To be designed to combat successfully a possible internal fire event;

- To be designed to confront successfully the thermal solicitation in case of loss of electric power supply, maintaining the temperature at design levels, without requiring the human intervention.

4. DESIGN OF CONTENTION BARRIERS

The barriers of physical contention regarding the environment must avoid the eventual migration of nuclear material in case some damage takes place in the SFE.

In ASECQ case, it has been foreseen 4 (four) contention physical barriers:

- 1st barrier: Rods of SFE (mechanical design of fuel element, provided by manufacturer);
- 2nd barrier: Silo Units (mechanical element, provided by ASECQ design);
- 3rd barrier: Silo Building (dry storage, provided by ASECQ design);
- 4th barrier: Building structural system (provided by ASECQ design).

It considers that the 1st and 2nd barriers are the two contention barriers regarding the external environment when the heat extraction of SFE is natural convection. The Silo ventilation system will be monitored in order to examine early noble gases detection. In this case, an alarm system will initiate automatically the aerosol system and it can be decided to put by hand the forced removal heat cooling system by closed circuit.

The 3rd and 4th barriers are added if the nuclear material is analyzed regarding the external environment of the building.

In addition to the previous physical barriers mentioned, dynamic barriers were established by ASECQ ventilation system, assuring air renovations and depressions required by nuclear standards for the silo and auxiliary rooms. Partial renovations of new air are carried out using a minimum volume from the ventilation system of Atucha I NPP; the same air quantity will be returned for previous treatment before expulsion to environment, through the Atucha I NPP chimneys.

5. FEATURES OF FUEL ELEMENTS

The main features of Fuel Element (FE) are: an active length of 5.3 m and a circular cross section of 0.10 m diameter, with 36 fuel rods plus one structural rod. Each fuel rod contains 442 UO_2 pellets cladding in a zircaloy-4 alloy tube with an external diameter of 11.92 mm and 0.55 mm thick. Each FE is loaded with approx. 176 kg of UO_2. The total number of FE in the Reactor core is 252. The design and fabrication is provided by CONUAR, national company of fuel elements.

Once used in the Reactor core, all SFEs are stored temporarily under water hanging vertically in stainless steel racks.

The SFEs stored in ASECQ will be those that do not have damages (zero injuries) in their structural components or in the rods, during the use in the Reactor or the transport to Pools I or II, and previously will be verified by the operator with a video camera system.

The SFE to be deposited in ASECQ will be the oldest ones located in Pool Buildings I, coincident with the first cores downloaded from the Reactor. It means that heat generation per each SFE will be very low, approximately 30 Watts.

In conclusion, it is expected that the first barrier will be in good condition, without detected damage and with a external surface temperature of the rod maintained below of 200°C thanks to the natural convection ventilation system.

6. SPENT FUELS TRANSFER AND DRY STORAGE DESIGN

Each SFE will be stored in square section baskets with 9 units of capacity, this group is called Storage Units (Unidades de Almacenamiento, UA in Spanish). These UA loaded with SFEs will be placed in a transference shielding unit, to carry it from Pool Building I to the Silo Unit. See Fig. 3.

The UA will be made of stainless steel, taking into account geometrical location to avoid criticality, similar to wet configuration in pools.

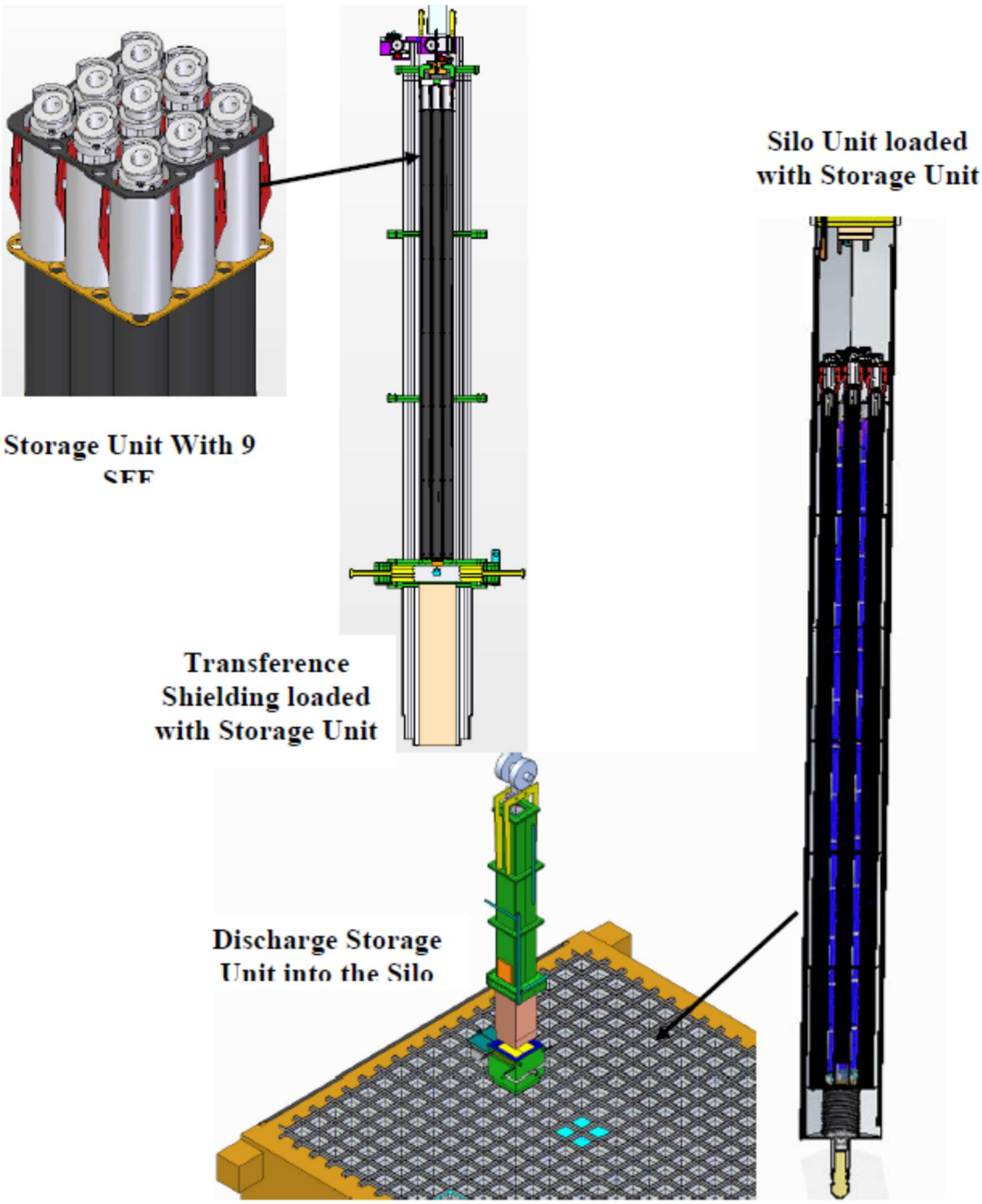

FIG. 3 – Details of transport and storage components for SFE.

Each UA has a support system on top which allows automatic locking and unlocking of the SFE when operators charge or discharge it at 9.5 m of depth in the pool, with a lifting tool.

The movements and the transfer between buildings are carried out exclusively in vertical form, inside the transference shielding unit to minimize structural efforts on the spent fuels avoiding inclining them or putting them to a horizontal position. This condition of transport is important because in the future the SFE will return to the Building Pool I.

The drying of UA with the nine SFE will be done inside the transference shielding unit blowing dry and hot air before introducing them into the Silo Unit.

Silo Unit is a robust box of stainless steel of 5 mm thickness and 8,5 m longitude, that will be subjected to a tightness testing control, in order to reach the second contention barrier.

Some Silo Units will be provided with instrumentation to allow getting information of the equilibrium temperature that the SFE bundle (UAs and Silo Units) will reach. When temperatures up to 200ºC are measured in the surface of the SFE rod, an alarm will be announced in the control room requiring intervention of the Plant Operator.

7. STORAGE SILO

The Silo Building is the central part of ASECQ Project, and will be separated from the external building structure. This provides good accessibility to the Silo Units and the possibility of inspecting the outside surfaces of walls and bottom of the Silo Building.

Silo Building represents the first physical and radiological barrier for operators and the third barrier for the environment. Some reports were made for the Silo: see Refs 15 to 17.

The total ASECQ storage capacity is 2754 SFE, contained in 306 Silo Units and 10 Silo Units empties for free use further decided by the Operator. See Fig. 4.

ASECQ Silo Building is a monolitical reinforced concrete structure with mechanical accessories embedded, made with traditional constructive techniques, in-situ civil work. It has perimetral walls with important thickness to provide a shielding structure for radiation, to resist impact of accident transport, the extreme thermal conditions, impact of missiles and earthquakes. Its geometry is like a huge cube.

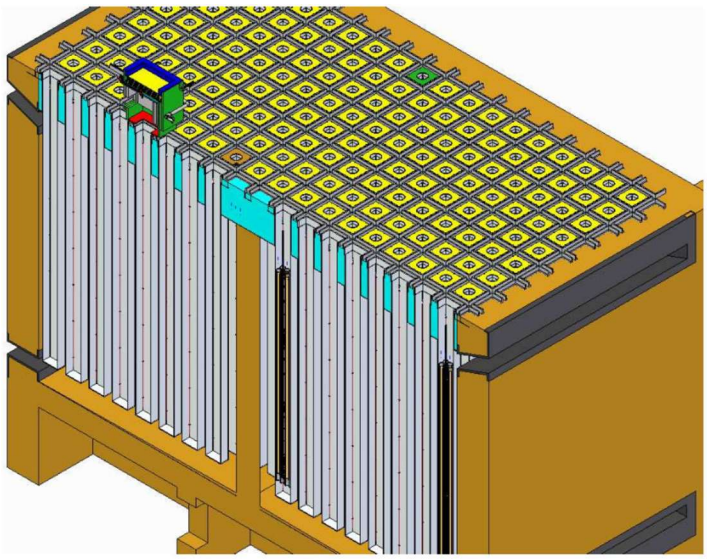

FIG. 4. The Silo Building.

8. BUILDING STRUCTURE SYSTEM

ASECQ Building has been designed as a nuclear installation with special local characteristics because its location is inside Atucha I NPP controlled area.

In 1968, SIEMENS AG was the original designer (architect/engineer) of the Plant. So, ASECQ Project design started analyzing existing drawings 'as built' of Pool Building I.

CNEA first steps in the civil design were the selection of an appropriate plan layout and, in parallel, the studies of geotechnical parameters, and the definition of civil engineering and earth structures. See Refs 18 to 20.

In the iterative design process software of civil engineering and architecture were used. The events and hazards analyzed for basic and detailed civil engineering design (DBA) are earthquakes, internal and external missiles, tornados, wind, explosive pressure waves, and dead and use loads.

The general distribution, in lay-out drawings, shows one nuclear room beside existing Pool Building I, and several auxiliary floors on the opposite side (west side). The building will have the same services of Pool Building I. See Fig. 5.

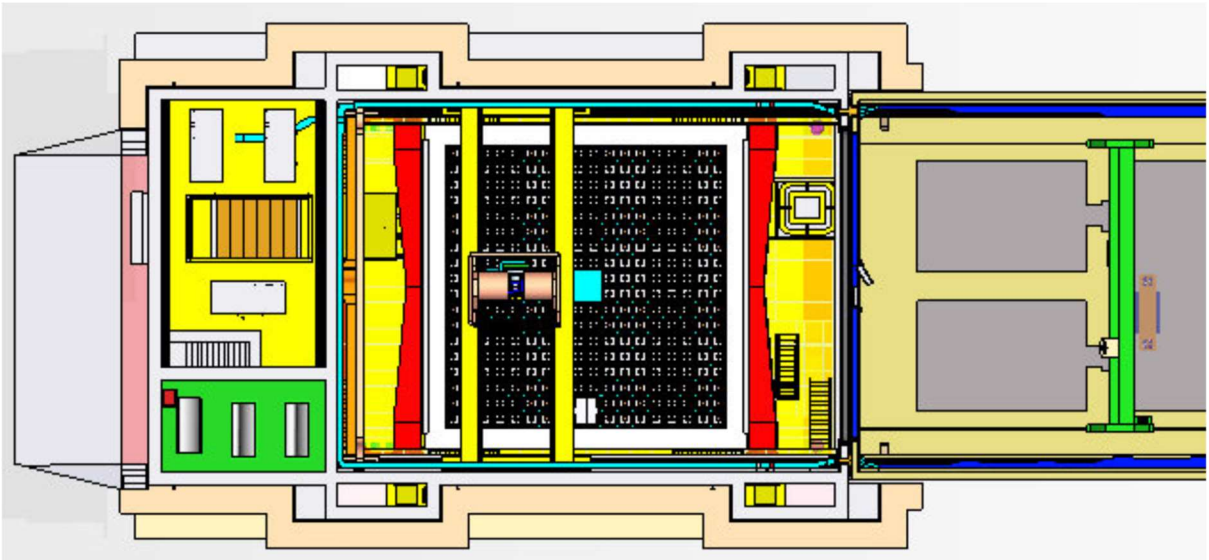

FIG. 5. Lay-out of ASECQ Building.

9. THE MOCK-UP FACILITY FOR TESTING

To demonstrate the thermal design performance, CNEA and NA-SA are building a special mock-up, near ASECQ construction site.

It consists on a representation model in scale 1 to 1, of nine Silo Units configurated into the Silo area with nine SFE placed in each one, that means loaded with 81 SFE, simulating a similar heat emanation in a real case, and it will test the real operation of the air natural

convection into the Silo for normal operation. Basically this mock-up is necessary to validate the thermal theoretical model, developed by specific software in CNEA.

This task facilitates the final design process to optimize and improve the detailed mechanical engineering.

10. ASECQ PROJECT CHRONOLOGY

First stage — Development of engineering project:

- Start conceptual engineering (mechanical & civil), presentation of technical purpose to NA-SA, January 2010 to June 2010;
- NA-SA independent revision of engineering, June 2010 to December 2010;
- Progress in civil basic & detailed engineering, January 2011 to January 2012;
- "Peer Review Mission IAEA IFMAP CNAI Spent Fuel Dry Storage Project" held from 12 to 16 March 2012 in CNAI site;
- Presentation of preliminary safety analysis to ARN, January 2013.

Second stage — Construction & erection:

- Civil contractor mobilization and work fence, March 2013;
- Start excavation and foundations, June 2013;
- Start building structure of reinforced concrete, December 2013 — in progress;
- Construction of mock-up for tests, December 2014 — in progress.

Third stage — Commissioning in 2017;

Fourth stage — Operation of the installation in 2018.

11. CONCLUSIONS

Due to the time that demands the design and licensing of an independent dry storage facility, and the necessity of solving spent fuel storage capacity to reach the end of design life and life extension of the plant for another ten years more of operation, it was decided to carry out ASECQ as an amplification of Atucha I NPP, considering it like a permanent modification of the original facility with a design lifetime of 50 years. The spent fuel management during the operation life of a NPP is a responsibility of the Operator.

Argentina has not defined yet its fuel cycle back-end strategy, but according to the Strategic Plan for Radioactive Waste and Spent Fuel Management proposed to the National Congress, this definition should be taken before the year 2030. In principle, the fissile material contained in the spent nuclear fuels is considered as a resource. Nevertheless a deep geological repository will be needed for the disposal of the SF or the HLW from its reprocessing, the intermediate radioactive wastes and long lived spent sealed sources. This DGR is currently planned to be started up in 2060.

The construction of ASECQ facility solves the long term storage of spent fuel beyond the life extension of the Plant. It is currently under construction and its starting up is planned for 2018. This design can be considered as an innovative technology in the Argentine Republic.

ACKNOWLEDGEMENTS

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The presenter wants to thank also CNEA authorities for the designation to participate in such event.

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SESSION 6

THORIUM FUEL CYCLES IN A VERY HIGH TEMPERATURE HYBRID SYSTEM

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Abstract

The current growth of the energy demand, the perspective of a pronounced increment for the next decades, added to the near depletion of the fossil fuels has made necessary finding a sustainable energy supply. Nuclear energy is presented as an important energy source to meet global energy needs in the near future, without adverse impacts to the environment. However, it faces substantial challenges to be successful as sustainable energy source. Pebble Bed Very High Temperature advanced systems together with fuel cycles based in Thorium present significant perspectives to assume the future nuclear energy. In this paper the main advantages of the use a Generation IV Very High Temperature Hybrid System (VHTHS) using a variety of fuel cycles based on Thorium (^{232}Th-^{233}U, ^{232}Th-^{239}Pu and ^{232}Th-U) under a deep burn scheme are studied. The conceptual design of the VHTHS composed of a Very High Temperature Pebble Bed Reactor (VHTR) and two Pebble Bed Accelerator Driven Systems (ADSs) is analyzed. Parameters such as the isotopic composition, the Minor Actinides stockpile, the nuclear fuel breeding, the percent fissile fuel and the radiotoxicity of the long-lived wastes are determined in order to know the influence of using Thorium based fuel cycles and ADSs as a second stage in the VHTHS. The MCNPX version 2.6e was used for the neutronic calculations.
8.

1. INTRODUCTION

The current growth of the energy demand, the perspective of a pronounced increment for the next decades, added to the near depletion of the fossil fuels has made necessary finding a sustainable energy supply. Nuclear energy is presented as an important energy source to meet global energy needs in the near future, without adverse impacts to the environment. Furthermore, nuclear energy has been gained international recognition in terms of energy security and greenhouse gas reductions. However, it faces substantial challenges to be successful as sustainable energy source. These substantial challenges are focused on achieving effective nuclear fuel utilization, increasing its environmental benefits, improving its safety performance, and developing an integral long-term management program for the spent fuel. Currently, two main options for the management of the nuclear fuel are used. If the spent fuel is not reprocessed, the nuclear fuel cycle is referred to as an 'open' or 'once through' fuel cycle. If the spent fuel is reprocessed and extracted radioactive isotopes are reused, it is referred to as a 'closed' fuel cycle [1]. However, an option that includes the 'closed' fuel cycle and transmutation strategies for reducing the long-lived radiotoxic materials is presented as one prominent option among the different options for the management of the spent fuel. This kind of management of spent fuel is referred to as an "advanced close" fuel cycle [2].

In order to solve the challenges mentioned above, new concepts of innovative nuclear energy systems known as Generation IV are under research and development. The VHTR considered as an outstanding candidate among the six prototypes of the Generation IV of nuclear energy systems, together with Thorium fuel cycles are studied in order to assume the future of the nuclear energy with a responsible management of the spent nuclear fuel. In addition, as a second stage, the ADS are investigated such an essential option for an advanced transmutation of the minor actinides and some fission products. In this investigation is analyzed the influence of using ADSs as a second in a VHTHS optimized in previous studies [3, 4]. Parameters such as the isotopic composition of the Thorium based fuel cycles, the Minor Actinides stockpile, the nuclear fuel breeding, the percent fissile fuel and the radiotoxicity of the long-lived wastes are determined in order to know the influence of using Thorium based fuel cycles and ADSs as a second stage in the VHTHS. The MCNPX version 2.6e was used for the neutronic calculations.

2. MATERIALS AND METHODS

The Very High Temperature Hybrid System is composed of a VHTR as a first stage and two ADSs as a second stage. The VHTR and the ADSs work in a thermal neutron spectrum, are moderated by graphite and cooled by Helium, both use TRISO coated particles confined in graphite pebbles as fuel elements. In the first stage, the fuel will be burned in the VHTR until the fuel in the nuclear reactor is not capable of sustaining the chain reaction. In the second stage, the nuclear fuel will be moved to the ADSs. When the energy generated in the ADSs is minor than the energy consumed by itself, the fuel will be sent to a geological repository. The goal of the Hybrid System is generate energy and transmute the long-lived radioactive elements into stable or short-lived elements, decreasing the mass and radiotoxicity of the waste, for then send it to a deep geological repository. The main advantages of the inclusion of the ADS as a second step of the VHTHS were analyzed in this investigation for three Thorium based fuel cycles (^{232}Th-^{233}U, ^{232}Th-^{239}Pu and ^{232}Th-U).

The first fuel cycle (^{232}Th-6%^{233}U) is based in a mixture of the fissile isotope ^{233}U and the fertile isotope ^{232}Th. The second fuel cycle (^{232}Th-8%^{239}Pu) is based in a mixture of the fissile isotope ^{239}Pu and the fertile isotope ^{232}Th. The third fuel cycle (40%^{232}Th-60%U enriched at the 8% in ^{235}U) is based in a mixture of ^{232}Th and fresh Uranium enriched in ^{235}U at the 8%. In the three Thorium based fuel cycle, the fuel mass in each pebble (10 g/pebble) and the enrichment in fissile fuel were selected to guarantee an adequate cycle duration in order to reach a discharge burnup of 75 GWd/t in the critical reactor (VHTR).

2.1. Main characteristics of the Very High Temperature Reactor

The VHTR proposed in this paper consists of an upright graphite block shaped like a cylinder with an outer diameter of 7.775 m, an inner diameter of 4.175 m, and a height of 4.939 m. This graphite structure serves as radial, bottom and top reflector. The fuel volume is 17.24 m^3 for a total of 263 465 pebbles. Inside the core was assumed a packing ratio of 0.74. The reactor thermal power is 200 MW(th). A schematic of the proposed VHTR is shown in Fig. 1.

2.2. Main characteristics of the Accelerators Driven Systems

The system is a subcritical reactor driven by a cyclotron of 1 GeV with a proton beam intensity of 10 mA. The spallation target is based on eutectic lead-bismuth in melted state with an innovative geometrical design for the separation window between the beam tube and target. This system was previously optimized in [5, 6]. The effective multiplication factor was minor than 0.950 in order to ensure subcritical operation and limited neutron source power.

This system has a graphite structure serves such a radial reflector of 3.284 m in outer diameter and 4.939 m of height. The core of the ADSs was designed with an inner diameter of 0.310 m, an outer diameter of 2.968 m and a height of 4.939 m. The total volume of the ADS is 20.120 m^3 and the volume of fuel, 8.620 m^3 for a total of 1157.5 kg of mass of fuel. The number of pebbles in the ADS bed is 131730. Inside the core the average power density was 3 MW/m^3 and a packing ratio of 0.74 was assumed. The fuel is burned at a thermal power of 60 MW(th) until the energy generated is minor than the energy consumed by the ADS. A schematic of the proposed ADSs is shown in Fig. 2.

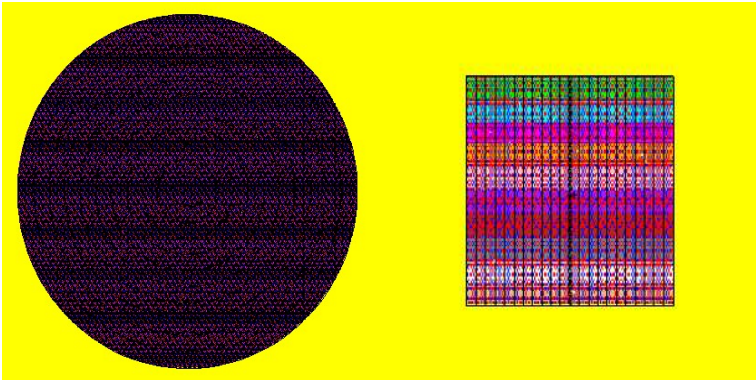

FIG. 1. Longitudinal and cross section geometrical model of the VHTR on the Visual Editor of the MCNPX.

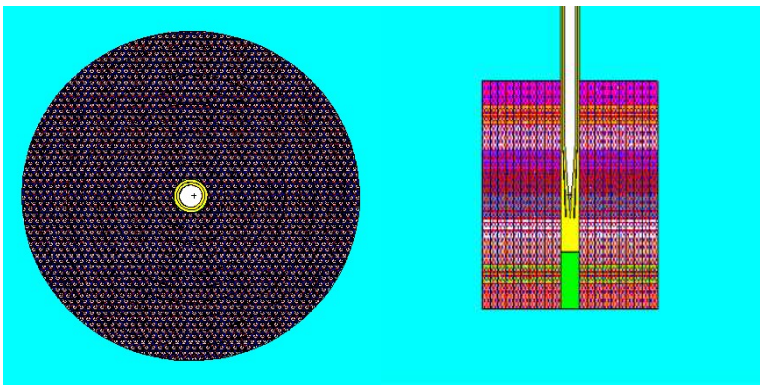

FIG. 2. Longitudinal and cross section geometrical model of the ADS on the Visual Editor of the MCNPX.

2.3. Computer code description

The MCNPX version 2.6e code was the calculation tool used in this study for in-core calculations. The MCNPX code is a transport code, which is intended for computer modeling of time dependent transportation of many types of particles, for different geometries and a wide range of energies [7]. The in-core neutronic calculations were performed with the available library in XSDIR, ENDF/B VI.2 and with an exact model of the geometry of the VHTR and ADSs.

3. RESULTS AND DISCUSSION

For the three fuel cycles the burnup cycle duration in the VHTR was 830 full power days. On the other hand, the burnup cycle duration in the ADSs is different in each studied fuel cycle.

In the first cycle (^{232}Th-6%^{233}U) 210 full power days, in the second cycle (^{232}Th-8%^{239}Pu) 400 full power days and in the third cycle (40%^{232}Th-60%U enriched at the 8% in ^{235}U) 1240 full power days (Fig. 3). The three fuel cycles will be burned under a deep burn scheme and a strategy of continuous recharge.

3.1. Variation of the isotopic composition

Fig. 4 indicate a reduction of the concentration of the fissile isotope ^{233}U in each stage of the Hybrid System for the ^{232}Th-^{233}U fuel cycle. However, the fissile isotope ^{233}U presents a light depletion due to the new ^{233}U obtained from transmutation of ^{232}Th. In the first stage in the VHTR, was consumed the 65% of the fissile isotope ^{233}U, in the second stage in the ADSs was consumed the 4% and a 31% was discharged from the ADSs. The fertile isotope ^{232}Th presents a minor variation with burnup and only a 5% of the initial ^{232}Th was consumed.

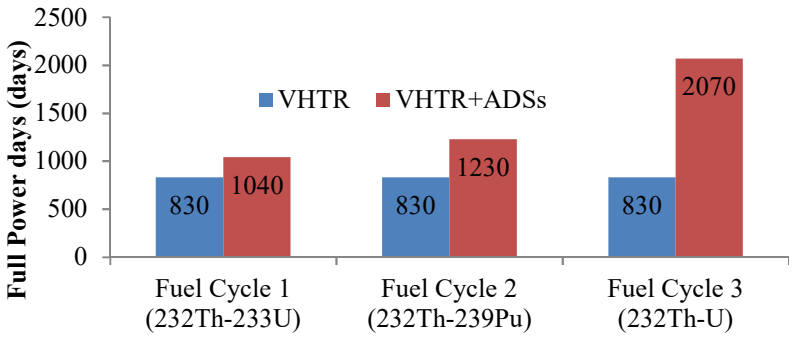

FIG. 3. Fuel cycles vs. burn up cycle duration in the VHTR and ADSs.

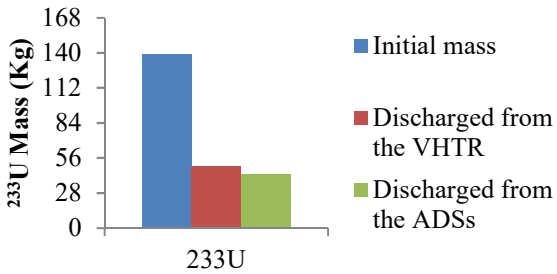

FIG. 4. Variation of the isotope ^{233}U in each stage of the Hybrid System in the ^{232}Th-^{233}U fuel cycle.

Results obtained for ^{232}Th-^{239}Pu fuel cycle show an important reduction of the fissile isotope ^{239}Pu respect to the ^{239}Pu initially loaded. In the first stage in the VHTR, was consumed a 98.25% of the ^{239}Pu, in the second stage in the ADSs was consumed a 1.70% and only a 0.05% was discharged from the ADSs. In this fuel cycle, the fertile isotope ^{232}Th was consumed in a 4.40% (2.72% in the VHTR and 1.69% in the ADSs). In addition, the fissile isotope ^{233}U was obtained from transmutation of the fertile isotope ^{232}Th. The concentration of ^{233}U reached 39.8 kg in the VHTR and increased in a 3.01% in the ADSs (see Fig. 5).

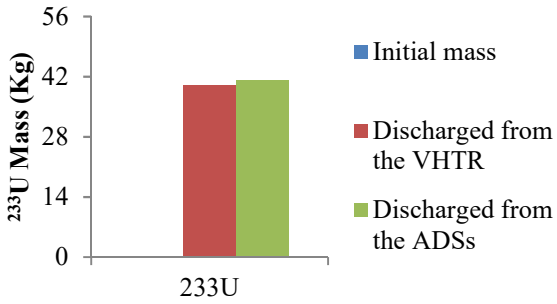

FIG. 5. Concentration of the fissile isotope ^{233}U in each stage of the Hybrid System in the ^{232}Th-^{239}Pu fuel cycle.

Similar results were obtained for the ^{232}Th-U cycle, where the fertile isotope ^{232}Th and ^{238}U presents minor variation with burnup. The ^{232}Th was consumed in a 4% in the first stage and in a 5% in the second stage. On the other hand, the ^{238}U was consumed in a 5% in the first stage and in a 7% in the second stage. The fissile isotope ^{235}U decreased in a 72% in the VHTR and in a 23% in the ADSs, only a 5% of the ^{235}U initially loaded was discharged from the ADSs (see Fig. 6). In the first stage in the VHTR, a concentration of 8.64 kg of the fissile isotope ^{239}Pu was obtained and in the second stage in the ADSs the ^{239}Pu mass was reduced to 6.44 kg (see Fig. 7).

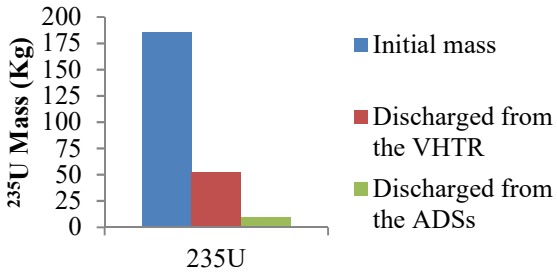

FIG. 6. Concentration of the fissile isotope ^{235}U in each stage of the Hybrid System in the ^{232}Th-U fuel cycle.

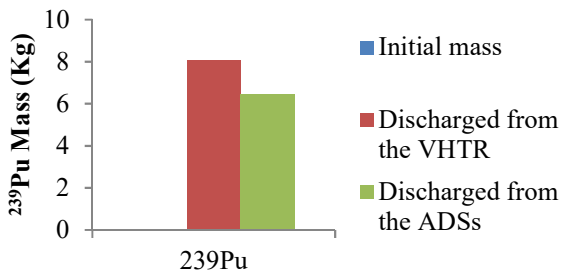

FIG. 7. Concentration of the fissile isotope ^{239}Pu in each stage of the Hybrid System in the ^{232}Th-U fuel cycle.

3.2. Plutonium isotopes stockpile

In the three fuel cycles, the Plutonium isotopes masses presented a growing behavior in the first stage. In the second stage, the inclusion of the ADSs provoked a growth in all isotopes masses except in the ^{232}Th-^{239}Pu fuel cycle; in this case, the ^{240}Pu and ^{241}Pu are consumed rapidly. The plutonium isotopes concentration reached 14.19 g in the ^{232}Th-^{233}U fuel cycle (see Fig. 8), 17.3 kg in the ^{232}Th-^{239}Pu fuel cycle (see Fig. 9) and 26.1 kg in the ^{232}Th-U fuel cycle (see Fig. 10).

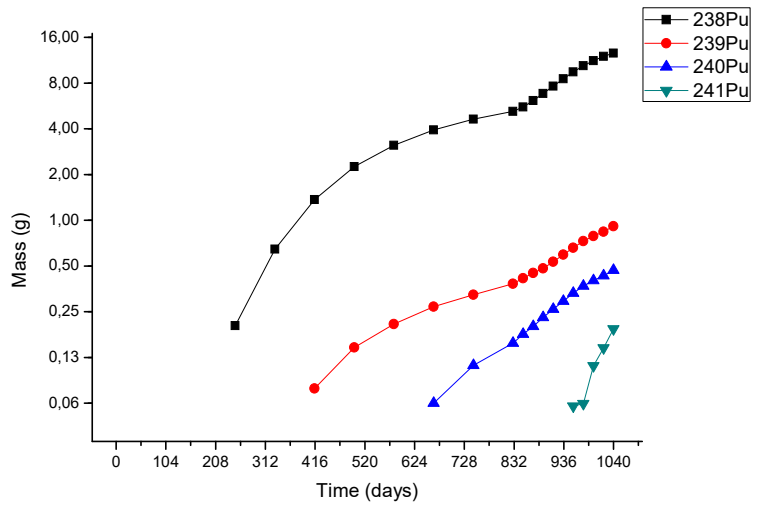

FIG. 8. Plutonium isotopes variation in the Hybrid System (^{232}Th-^{233}U cycle).

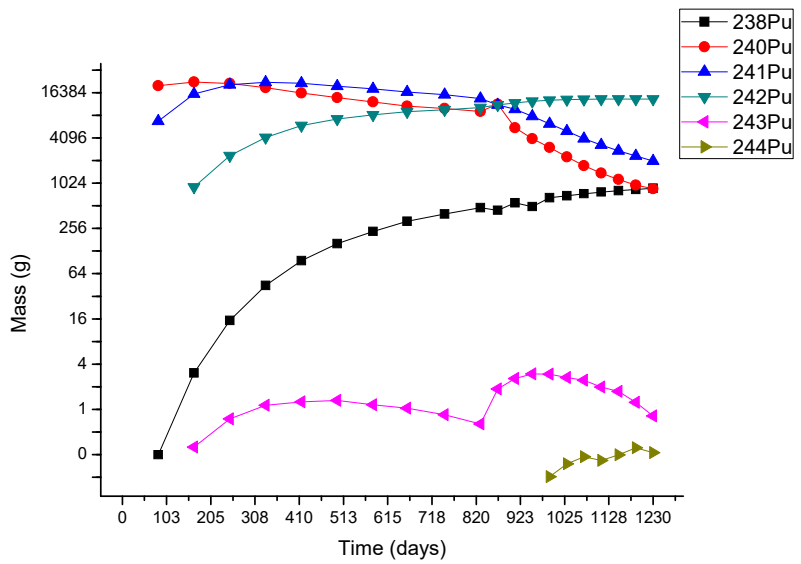

FIG. 9. Plutonium isotopes variation in the Hybrid System (^{232}Th-^{239}Pu cycle).

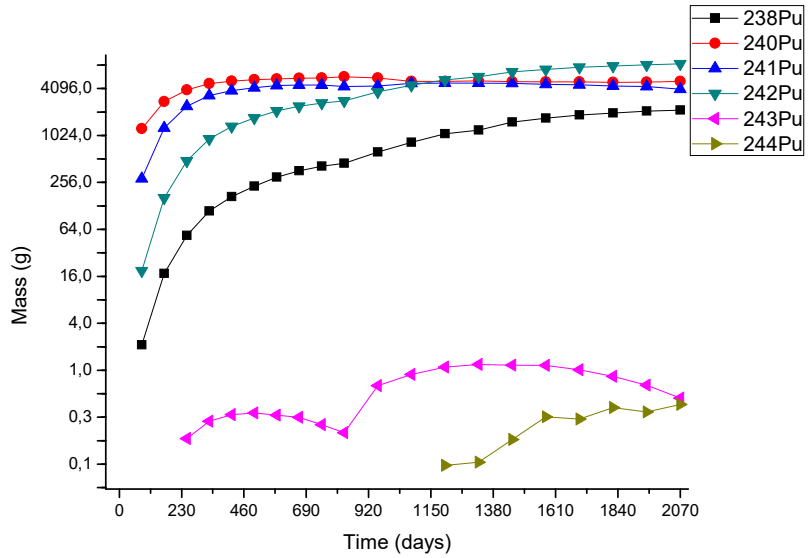

FIG. 10. Plutonium isotopes variation in the Hybrid System (^{232}Th-U cycle).

3.3. Minor Actinides Stockpile

The mass variations of the Minor Actinides in each stage of the Hybrid System as function of time are shown in Fig. 11 for ^{232}Th-^{233}U fuel cycle, Fig. 12 for ^{232}Th-^{239}Pu fuel cycle and Fig. 13 for ^{232}Th-U fuel cycle, respectively.

The inclusion of the ADSs increased the concentration of the Minor Actinides in the three fuel cycles, but in small quantities. In the ^{232}Th-^{233}U fuel cycle was obtained 16.5 g in the first stage and 30.2 g in the second stage. In the ^{232}Th-^{239}Pu fuel cycle was obtained 2.35 kg in the first stage and in the second stage was increased the Minor Actinides concentration in a 36%. The ^{232}Th-U fuel cycle presented the largest concentration of Minor Actinides respect to the three fuel cycle studied. In the ^{232}Th-U fuel cycle was obtained 1.87 kg in the first stage and 6.10 kg in the second stage.

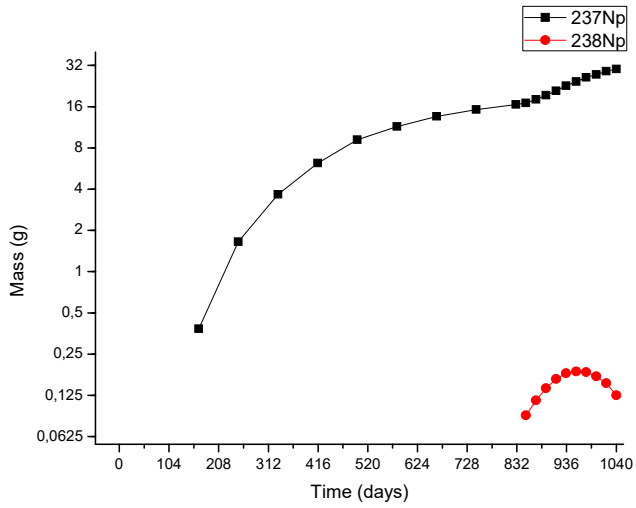

FIG. 11. Minor Actinides mass variation in the Hybrid System (^{232}Th-^{233}U) cycle.

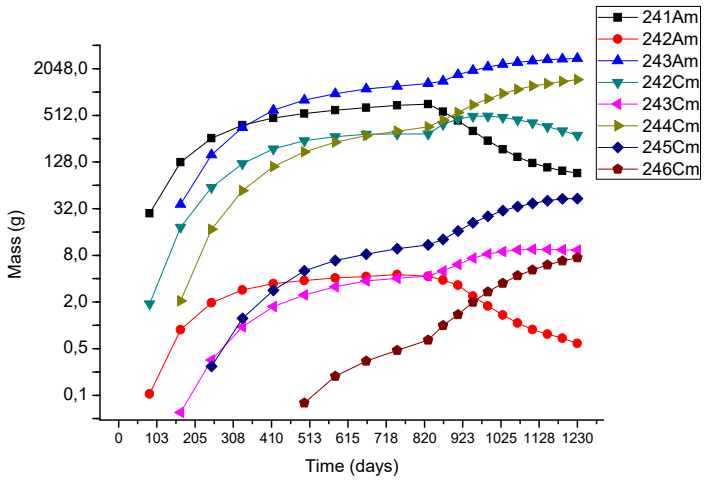

FIG. 12. Minor Actinides mass variation in the Hybrid System (^{232}Th-^{239}Pu cycle).

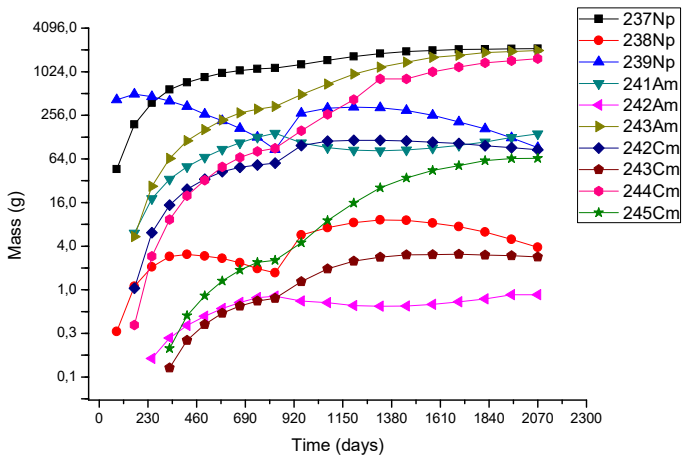

FIG. 13. Minor Actinides mass variation in the Hybrid System (^{232}Th-U cycle).

3.4. Nuclear fuel breeding and percent fissile fuel

Fuel Breeding Coefficient (BC) values (see Table 1) were determined in each stage of the Hybrid System for the Thorium based fuel cycles using Equations defined in [8, 9]. The comparison among the proposed cycles, confirmed that a high fuel breeding in the three cycles in the hybrid system (BC=0.71, 0.47 and 0.98), can be obtained. In addition, the inclusion of a second stage in the ADSs increased the Breeding Coefficient in a 5%, 27% and 36% respect to the first stage in the VHTR.

TABLE 2. BREEDING COEFFICIENT (BC) IN EACH STAGE OF THE HYBRID SYSTEM

Cycle	VHTR BC	VHTR + ADSs BC
^{232}Th-^{233}U	0.68	0.71
^{232}Th-^{239}Pu	0.37	0.47
^{232}Th-U	0.72	0.98

A parameter that allows evaluating the efficiency of a one-through cycle, in which the spent fuel is destined to long term storage, is the final percent of fissile fuel mass (% fissile fuel). Table 2 shows the % fissile fuel values for the three fuel cycles.

TABLE 3. % FISSILE FUEL FOR THE FUEL CYCLES STUDIED

Cycle	VHTR (% fissile fuel)	Reactor + ADSs (% fissile fuel)
^{232}Th-^{233}U	38.06%	33.78%
^{232}Th-^{239}Pu	30.94%	23.98%
^{232}Th-U	44.81%	22.05%

In each case the final percent of fissile fuel mass decrease due to an increase of the burnup of the fuel in the second stage in the ADSs.

3.5. Energetic contribution of each fissile isotope

The contribution of each fissile isotope to the total produced energy in the Hybrid System was determined for the fuel cycles studied. The results of the percent of release energy per isotope in each fuel cycle are indicated in Fig. 14.

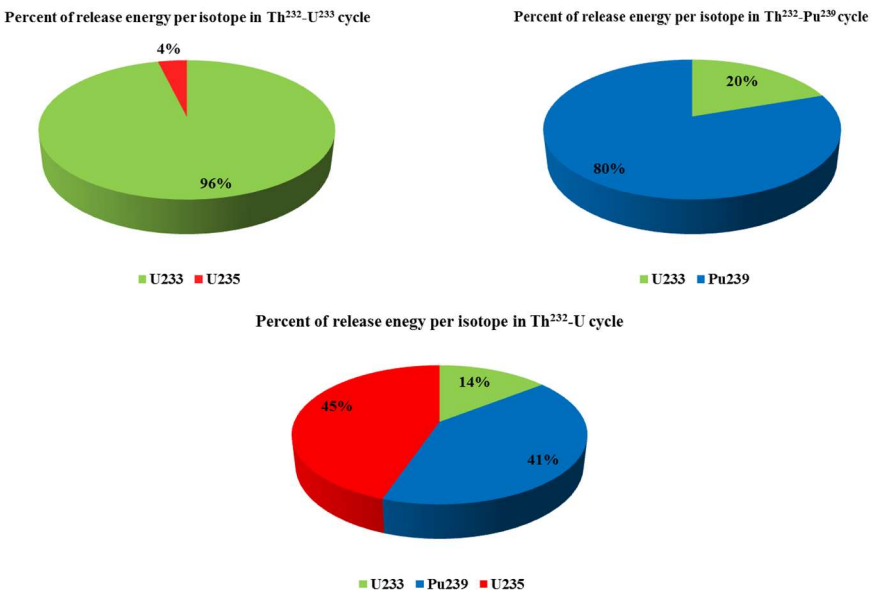

FIG. 14. Energetic contribution of each fissile isotope in ^{232}Th-^{233}U cycle (on the up left), ^{232}Th-^{239}Pu cycle (on the up right) and ^{232}Th-^{233}U cycle (down).

In two of the fuel cycle the major energetic contribution to the release energy is given by the initial fissile isotope. In the ^{232}Th-^{233}U cycle is the fissile isotope ^{233}U and in the ^{232}Th-^{239}Pu cycle is the fissile isotope ^{239}Pu. In the ^{232}Th-U cycle exist a balance between the initial fissile isotope ^{235}U and the reproduced fissile isotope ^{239}Pu.

3.6. Radiotoxicity of the long-lived wastes

The inhalation radiotoxicity in time after that the fuel is discharged from the VHTR is shown in Fig. 17 and Fig. 18 for the VHTR+ADSs.

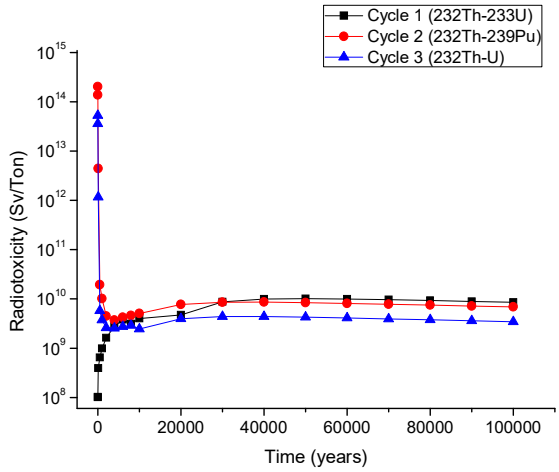

FIG. 17. Variation of the inhalation radiotoxicity, 100 000 years after the fuel is discharge from the VHTR.

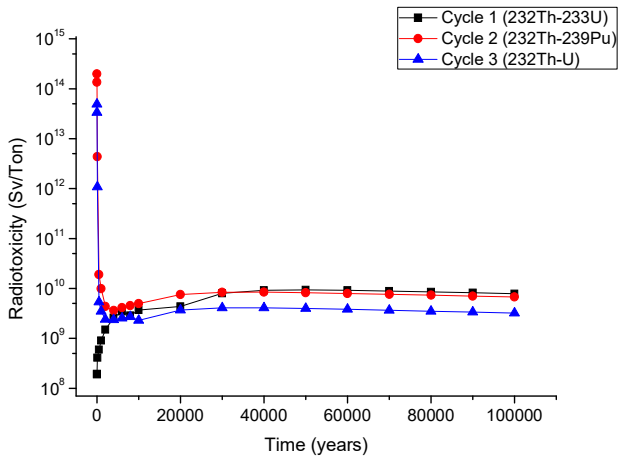

FIG. 18. Variation of the inhalation radiotoxicity, 100 000 years after the fuel is discharged from the ADSs.

In the reactor ^{232}Th-^{239}Pu and ^{232}Th-^{233}U fuel cycle radiotoxicity decreased over time, although in the ^{232}Th-U fuel cycle the radiotoxicity increased. The three fuel cycles reached an equilibrium concentration at the 5000 years with radiotoxicity close to E+10 Sv/ton (see Fig. 17). As can be seen in Fig. 18 the inclusion of the second stage in the ADSs provoked an initial increase in the radiotoxicity of the 49% in the ^{232}Th-^{233}U fuel cycle. On the other hand,

the ^{232}Th-^{239}Pu and ^{232}Th-^{233}U fuel cycles resulted in a decrease of the 2% and 8%. At the end of the studied period (100 000 years), the inclusion of the ADSs caused a reduction in the radiotoxicity of the discharged fuel of the 9%, 2% and 8% in comparison to the results obtained in the VHTR for each fuel cycle respectively.

4. CONCLUSIONS

The results presented in this paper emphasize the main advantages of using Thorium based fuel cycles in a Very High Temperature Hybrid System. Furthermore the main advantages of incorporating two ADSs as a second stage of this Hybrid System. The inclusion of a second stage in the ADSs allowed increasing the burnup cycle duration. The ^{232}Th-U fuel cycle reached the highest burnup cycle duration equivalent to 2070 full power days (830 days to 200 MW(th) in the VHTR and 1240 days to 60 MW(th) in the ADSs).

In the ^{232}Th-^{239}Pu fuel cycle the ^{239}Pu initially loaded was consumed in a 99.95% in the Hybrid System. The concentration of plutonium isotopes presents a growth in the ^{232}Th-^{233}U and ^{232}Th-U fuel cycles. However, the concentration of plutonium isotopes decreased with the inclusion of the ADSs as a second stage for the ^{232}Th-^{239}Pu fuel cycle. The ^{232}Th-^{239}Pu fuel cycle in its first stage reached 36.9 kg and 17.3 kg in the second stage in the ADSs. The inclusion of the ADSs increases the concentration of the Minor Actinides in the three fuel cycles, but in small quantities. The ^{232}Th-^{233}U fuel cycle reached the lowest quantity of Minor Actinides equivalent to 30.2 g. A high fuel breeding was obtained in the Hybrid System in the three fuel cycles (BC=0.71, 0.47 and 0.98). In addition, the inclusion of a second stage in the ADSs increased the Breeding Coefficient in a 5 %, 27 % and 36% respect to the first stage in the VHTR and provoked a decrease in the final percent of fissile fuel mass (% Fissile fuel) in an 11%, 22% and 50% for each cycle respectively.

The inclusion of a second stage in the ADSs caused an initial increase in the radiotoxicity of the ^{232}Th-^{233}U fuel cycle in a 49%, while for the ^{232}Th-^{239}Pu and ^{232}Th-U fuels cycles resulted in a decrease of the 2% and 8% respect to the values obtained in the VHTR. At the end of the studied period (100 000 years), the inclusion of the ADSs caused a decrease in the radiotoxicity of the discharged fuel of the 9%, 2% and 8% in comparison to the results obtained in the VHTR for each cycle respectively.

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DESIGN STRATEGIES FOR DIRECT RECYCLING OF ACR-700 SPENT FUEL

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Abstract

This work is focused on strategies for direct recycling of the 700 MW€ Advanced CANDU Reactor (ACR-700) spent fuel in CANDU-6 reactor. ACR-700 discharges the fuel with a significant amount of fissile isotopes (^{235}U,^{239}Pu and ^{241}Pu). Three strategies are considered for recycling the spent fuel of ACR-700. First strategy is recycling the ACR-700 spent fuel bundles directly in CANDU-6, since the two reactors have the same inner diameter of the fuel channel. Second strategy is removing the central pin (which has residual reactivity plenty of the Dysprosium fissionable poisons) from the ACR-700 spent fuel and re-fabricating the spent fuel bundle into CANDU-6 fuel bundles using dry processing such as the DUPIC fuel cycle.Third strategy is removing the outer fuel pins (which have the fewer amounts of fissile isotopes) and the central pin and re-fabricating the rest two fuel rings into CANDU-6 fuel bundles using the dry processing. The calculations using the MCNPX code showed that the recycled spent fuel in CANDU-6 gives burnup around 3, 6.75 and 13 MWd/kgU for the three strategies, respectively. Normalizing the burnup on the all fuel in the spent bundle, additional burnup of 3, 6.5 and 6.7 MWd/kgU can be obtained for the three strategies respectively. This means that the third strategy gives the higher burnup from the spent fuel. Moreover in the third strategy, only half of the spent fuel bundle is re-fabricated into CANDU-6 fuel bundle which reduces the cost of re-fabrication. Knowing that ACR-700 burns fuel to 20.5 MWd/kgU, the third strategy can increase the burnup of fuel by about 33%. The calculations gave acceptable power distributions on the fuel bundles for the three strategies and acceptable coolant void reactivity compared to reference CANDU-6.

1. INTRODUCTION

Increasing the nuclear fuel burnup has a particular importance to improve the uranium utilization, to reduce the high level nuclear waste and to reduce the amount of plutonium in spent fuel per unit energy which improves the plutonium proliferation resistance.The quantities of spent fuel that are produced by nuclear reactors depend upon the reactor and fuel type and other technical parameters, operational history and the fuel burnup. Current Light Water Reactors (LWRs) uses fuel enriched up to 5wt% and discharges spent fuel with burnup around 50 MWd/kgU and it is aimed to be over 60 MWd/kgU. CANDU reactor is originally designed for natural uranium fuel or slightly enriched uranium fuel. CANDU-6 burns natural uranium to about 7.5 MWd/kgU while the 700 MW(e) Advanced CANDU Reactor (ACR-700) uses 2.1wt%enriched uranium and has an average burnup of 20.5 MWd/kgU[1]. Therefore, the quantities of spent fuel that are produced per unit energy from CANDU reactors is much higher than that produced from LWRs. A typical 1000 MW(e) LWR burnup will generate only around 25 tons of spent fuel per year while, per 1000 MW(e), a CANDU reactor produces, in a year, around 125 tons of spent fuel but with specific activity and decay heat much lower than that from LWRs, which might simplify its later management [2].

The extraction of the fissile isotopes from the spent fuel through the chemical reprocessing [3] and recycling them in the reactors was proposed early when the commercial power plants were built. Since two or three decades, some countries like USA have stopped the fuel reprocessing as a political consideration of weapons proliferation. Also, reprocessing of nuclear fuel is not economically attractive [4]. The Direct Use of Spent PWR Fuel In CANDU Reactors (DUPIC), originally proposed in Korea,employs dry thermal–mechanical processes to convert spent LWR fuel into CANDU fuel. One simple way of DUPIC is cutting the PWR fuel elements into CANDU length (~50 cm), straighten them, and then weld new end-caps to the ends and constricting the CANDU bundles from 48 or 61 element [5]. The general method of DUPIC is re-fabricate the PWR spent fuel in a DUPIC plant such as AIROX [6−8] for CANDU bundles where the cladding is punctured and fission gasses are captured. The spent

fuel pellets are transferred to a furnace and reduced to a fine powder and then milled, shaped and sintered into CANDU fuel pellets. The complex DUPIC processing facility and the transportation ofhighly radioactive materials from the PWR to this facility and from this facility to the CANDU reactor makes the estimation of the economic of DUPIC cycle difficult [8].

In a previous work [9], a new design was proposed for the PWR fuel assembly for direct use of the PWR spent fuel without processing. The PWR spent fuel will be transferred directly (after a certain cooling time) to CANDU reactors which preferably built in the same site to avoid the problem of transportations.The proposed assembly has four zircaloy-4 tubes contains a number of 61- element CANDU fuel bundles (7 or 8 bundles per tube) stacked end to end. Each tube has the same inner diameter of that of CANDU pressure tube. The space between the tubes contains 44 low enriched UO_2 fuel rods and 12 guide tubes. Monte Carlo calculations showed that the burnup can be increased by about 25%. This in turn, beside the improvement of the fuel utilization, will reduce the high level nuclear waste and improve the plutonium proliferation resistance.

This work is focused on strategies for direct recycling of the ACR-700 spent fuel in CANDU-6 reactor.

2. DESIGN CONSIDERATIONS

The high neutron economy CANDU reactors are designed for natural or slightly enriched uranium and on-power refueling. CANDU-6 uses heavy water for cooling and moderation. The Advanced CANDU Reactors (ACR: ACR-700 and ACR-1000) are designed by Atomic Energy of Canada Limited (AECL) as the next generation of the operated CANDU reactors. The ACR design is based on the use of modular horizontal fuel channels surrounded by a heavy water moderator, the same feature as in all CANDU-6 reactors while it used pressurized light water as a coolant. ACR uses minimum lattice pitch (22 cm) with larger calandria diameter (7.8 cm) to displace more moderator for the reduction of heavy water as shown in Fig. 1. This design resulted in a more compact reactor and a reduction of heavy water inventory and therefore reduces the capital and operating cost with high capacity factor [1]. Two important parameters affect the design of the CANDU fuel bundles. These parameters are the power distribution across the fuel bundle and the coolant void reactivity.

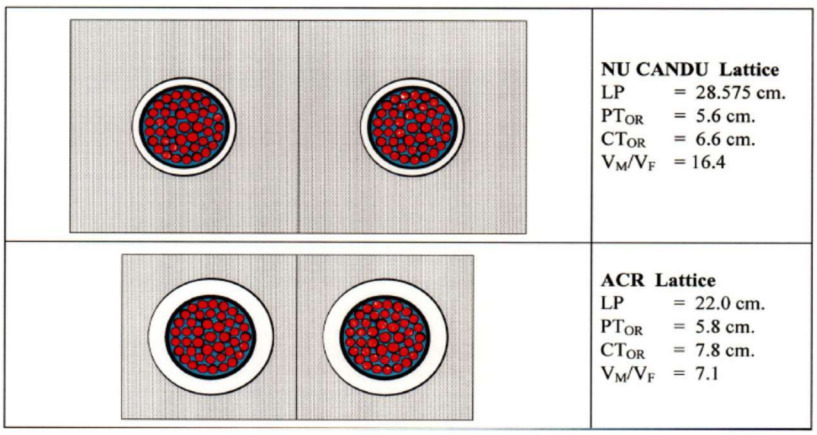

FIG. 1. ACR and CANDU-6 Lattice-Cell configurations [1].

2.1. Power distribution

In CANDU fuel bundles, the power per fuel pin is not uniform throughout the bundle since the fission neutrons are thermalized in the moderator outside the fuel channel. Therefore the fission rate is decreased from the outer pins to the central pin. The bundle power is limited such that the fuel melting will not occur. As a figure of merit, the N equivalent, N_{eq} parameter is a measure of the power distribution across the bundle. The parameter is defined in Eq. (1) as [10]:

$$N_{eq} = \sum_{i} \frac{p_i}{p_m} \tag{1}$$

Where the sum is over all elements *i*, p_i is the element power and p_m is the maximum element power. N_{eq} provides good estimations for fuel pencil diameter of 1 to 2 cm [10].

From this definition, higher N_{eq} means more uniform power distribution. Many parameters affect the N_{eq}. These parameters includes, fuel enrichments, lattice pitch size, fuel pin diameter and number of fuel pins. Increasing the fuel enrichment decreases N_{eq} due to the self-shielding of the fuel pins. CANDU-6 uses 37-element fuel bundles of natural $UO_2$with average power per channel of 5.3 MW. ACR-700 uses 43-element CANFLEX fuel bundle and operated at average power of 6.8 MW per channel. The bundle has two elements size: center pin and inner ring of seven elements with a diameter of 13.5 mm and the outer two rings consist of 35 elements with 11.5 mm diameter. The three fuel rings in the ACR-700 are enriched to 2.1wt% ^{235}U while the central pin is natural UO_2 contains Dy_2O_3 with 7.5wt% Dy in natural Uranium [1].

2.2. Coolant void reactivity

Another important parameter affecting the design of CANDU reactors is the coolant void reactivity. The coolant void coefficient of reactivity is negative in light water reactors, since the coolant and the moderator are the same fluid in the reactor pressure vessel. In CANDU reactors the moderator remains interact when the coolant is lost, and consequently the void reactivity may be positive. Many methods are proposed for the reduction of coolant void coefficient of reactivity [10, 11]. The main two methods are decreasing the moderator volume and use of burnable poisons in the central pin(s) of the fuel bundle since it usually have a proportionality larger contribution to the void reactivity than the outer elements. Marczak[10] listed a number of reasons that make the positive coolant void reactivity with a value greater than the delayed neutron fraction, β in CANDU-6 is acceptable. These reasons includes: (1) the coolant voids during a LOCA at a slow rate ensuring effective shutdown system action; (2) The reactor is divided into two separate cooling loops and a LOCA can generally expected to leave the cooling system; (3) The negative fuel temperature coefficient of reactivity; (4) The long prompt neutron lifetime (compared to that in the LWRs) limits the power increase rate following a LOCA; and (5) two separate and independent shutdown system in CANDU-6 increasing confidence in the ability of shutdown systems to operate effectively. ACR-700 uses light water as a coolant which has very short prompt neutron lifetime and therefore ACR-700 uses minimum lattice pitch with large calandria diameter and dysprosium (7.5% in natural uranium) as a burnable poisons in the central pin resulting in slightly negative coolant void reactivity.

3. STRATEGIES FOR FUEL RECYCLING

ACR-700 design reduces the capital and operating cost of CANDU reactors however it has poor neutron economy compared to CANDU-6. Therefore, ACR-700 will discharge the fuel with a significant amount of fissile isotopes (^{235}U, ^{239}Pu and ^{241}Pu). CANDU-6 and ACR-700 have the same inner diameter of the pressurized tubes (5.17 cm) with the same bundle lengths and the same number of bundles per channel. This in turn enables to recycling the ACR-700 spent fuel bundles directly in CANDU-6. The gradient of fission rate from outer pins to the central pin in the ACR-700 makes the outer ring, which has the higher contribution of the fuel bundle reactivity, has the fewer amounts of fissile isotopes and therefore limits the fuel bundle reactivity at the discharge. Also, the residual reactivity plenty of the Dysprosium fissionable poisons in the central pin decreases the spent ACR-700 fuel bundle reactivity. Two alternative strategies are proposed for direct use of ACR-700 spent fuel:

- Removing the central pin from the ACR-700 spent fuel and re-fabricating the spent fuel bundle into CANDU-6 fuel bundles using dry processing such as DUPIC fuel cycle;
- Removing the outer fuel pins and the central pin and re-fabricating the rest two fuel rings into CANDU-6 fuel bundles using dry processing.

4. SIMULATION

The Monte Carlo MCNPX2.7.0 code [12] is used in this research for the calculations. This code is compatible with MCNP5 [13] with many additional capabilities including material depletion/burnup calculations. All standard evaluated nuclear data libraries used by MCNP can be used by MCNPX 2.7.0. Data libraries containing particle-interaction can be replaced by physics models if the libraries are not available. The program also includes cross-section measurements, benchmark experiments, deterministic code development, and improvements in transmutation code and library tools through the CINDER90 project [14]. The ACR-700 and CANDU-6 lattice cells were simulated. The boundaries of the lattice cells were simulated with reflecting surfaces. Any particle hitting a reflecting surface is specularly (mirror) reflected. Therefore, the calculations were carried out for infinite reactors constructed from these unit cells.The neutron cross section data for the fuel, the coolant and CANDU moderator were recalled from ENDF/B-VII.0 library which is publicly available. The best matching temperature in the library for the fuel is 900 K, for the coolant is 600 K and for the D2O moderator is 294 K. S(α,β) [15] data were recalled for hydrogen and deuterium with best matching temperature in the library.

The fuel in the bundle is represented by four materials: one for each fuel ring and one for the central pin. The simulation was carried out for the following cases (given in Table I):

Case 1: ACR-700 is loaded with its reference fuel Bundle.

Case 2: CANDU-6 is loaded with its reference fuel Bundle.

Case 3: CANDU-6 is loaded with the spent ACR-700 fuel bundle. In this case the spent fuel bundle with its material inventories is simulated in CANDU-6 channel after a decay of 90 days.

Case 4: CANDU-6 is loaded with the spent ACR-700 fuel after removing the central pin and re-fabricating into CANDU-6 fuel bundles. In this case, the material inventories of the three rings are mixed and decayed for 90 days and inputted for the material cards.

Case 5: CANDU-6 is loaded with the spent ACR-700 fuel after removing the outer fuel ring and the central pin and re-fabricating into CANDU-6 fuel bundles. In this case, the material inventories of the rest two rings are mixed and decayed for 90 days and inputted for the material cards.

A criticality calculation (KCODE calculations) with the BURN cards is used to calculate the system criticality and the burnup of the fuel and fuel inventory after each time interval (defined in the BURN cards). The power was given as 5.3 MW for the CANDU-6 channel and it was given as 6.8 MW for the ACR-700 channel. The time intervals were input as 0.5, 1, 2, 4, 10, 20, and then 30 s and at the end of the cycles it were 60s day in the BURN cards.The material inventories are assumed to be decayed for 90 days. A total of 150 cycles per burnup step of which 30 were skipped and 2000 histories per cycle were used to reach standard deviation of the k∞ values around 0.1%. In BOPT card the Tier 3 fission products was used [12].

The coolant void reactivaty as function of the fuel burnup are calculated for the all cases. The material inventories at different fuel burnup were inputted in the lattice cell MCNP files to run 150 cycles of which 10 were skipped and 20 000 histories per cycle to reach standard deviation of about 30 pcm for normal and voided coolant. All the data required for the simulations is collected from Refs. [10, 15].

5. RESULTS AND DISCUSSIONS

The equilibrium CANDU core contains fuel at a range of burnups, from 0 to exit-burnup values.Therefore, the infinite medium reactivity, ρ∞ can be calculated from Eq. (2):

$$\rho_\infty = \frac{\int_0^{exit} \rho_\infty(B)dB}{B_{exit}} \tag{2}$$

Where, $\rho_\infty(B)$ is the infinite medium reactivity at burnup B and B_{exit} is the burnup at the discharge. Fig. 2 gives the calculated depletion histories: $\rho_\infty(B)$ as function of B .The discharge burnup (B_{exit}) of ACR-700 and CANDU-6 are 20.5 MWd/kgU and 7.5 MWd/kgU, respectively.Using Eq. 2 and Fig. 2, ρ_∞ is calculated (numerically) for the reference ACR-700 and CANDU-6 which is found:

$\rho_\infty = 0.0636$ for ACR-700 and $\rho_\infty = 0.0578$ for CANDU-6.

In order to find the discharge burnup of the recycled spent fuel (Cases: 1, 2 and 3), the integration in Eq. 2 is carried out of the burnup until ρ_∞ decreases to 0.0578, since the fuel is recycled in CANDU-6. For case 3 where the ACR-700 spent bundle is recycled directly without any process, the spent fuel can be burnt to less than 3 MWd/kgU. In case 4 where the central pin is removed from the spent bundle and 97% of the spent fuel is recycled, ρ_∞ becomes 0.0599 when the discharged burnup is 6.75 MWd/kgU. In case 5 where the central pin and the outer fuel ring are removed from spent bundle and 52% of the spent fuel is recycled, k_∞ becomes 0.0582 when the discharged burnup is 13 MWd/kgU. Normalizing the burnup on the all fuel in the spent bundle, we find that case 3 gives additional burnup of about 3 MWd/kgU, case 4 gives additional burnup of 6.5 MWd/kgU and case 5 gives additional burnup of 6.7 MWd/kgU which represents an increase in the fuel burnup of 14.6%, 31.7% and 33% for the three cases, respectively as given in Table 1. From these results, we can conclude that recycling ACR-700 spent fuel using strategy of case 5 gives higher burnup than

cases 3 and 4. Moreover only about half of the spent fuel bundle is re-fabricated into CANDU-6 fuel bundle which decreases the cost of re-fabrication.

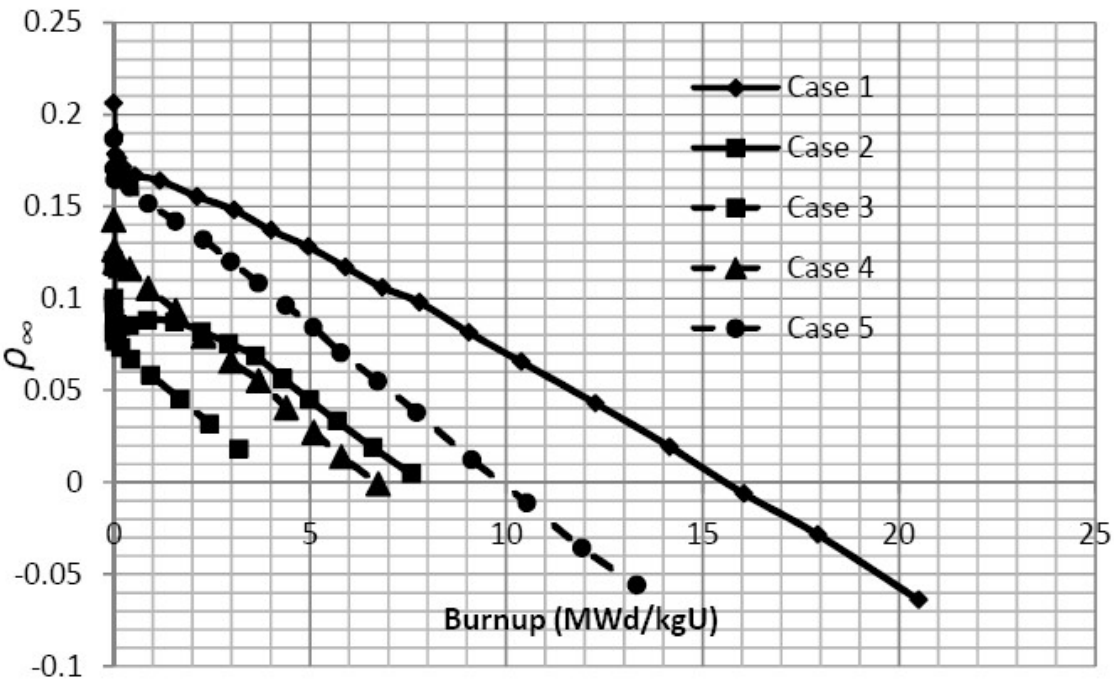

FIG. 2. Depletion histories of studied cases

TABLE 1. SIMULATION RESULTS FOR THE STUDIED CASES

Case	Description	Burnup (MWd/kgU)	% increase in burnup	Mid-burnup $N_{eq.}$	Mid-burnup CVR (mk)
Case 1	ACR-700 loaded with its reference fuel Bundle	20.5	-	39	-1
Case 2	CANDU-6 loaded with its reference fuel Bundle	7.5	-	32.9	12.6
Case 3	CANDU-6 loaded with the spent ACR-700 fuel bundle	3	14.6	34.5	13.5
Case 4	CANDU-6 loaded with the spent ACR-700 fuel after removing the central pin and re-fabricated into CANDU-6 fuel bundles	6.75	31.7	32	12.5
Case 5	CANDU-6 loaded with the spent ACR-700 fuel after removing the outer fuel ring and the central pin and re-fabricated into CANDU-6 fuel bundles	13	33	32.6	13.4

The Neq parameter as function of the burnup for the five cases is presented in Fig. 3.The design of ACR-700 fuel bundles gives higher Neq parameter than the design of CANDU-6 and therefore it is operated at higher power. The hot fuel pin remains in the outer ring of

ACR-700 bundle until about 15 MWd/kgU where it is shifted to the inner ring. In case 3 the hot fuel pin remains in the inner ring. The reference CANDU-6 has N_{eq} parameter around 33 over the all range of burnup. In case 3 and 4 Neq parameter starts with about 31 and increases with burnup. N_{eq} will be 32 and 33 at the mid-burnup where the fuel bundle position will be in the middle of the fuel channel andthe bundle power density is maximum.

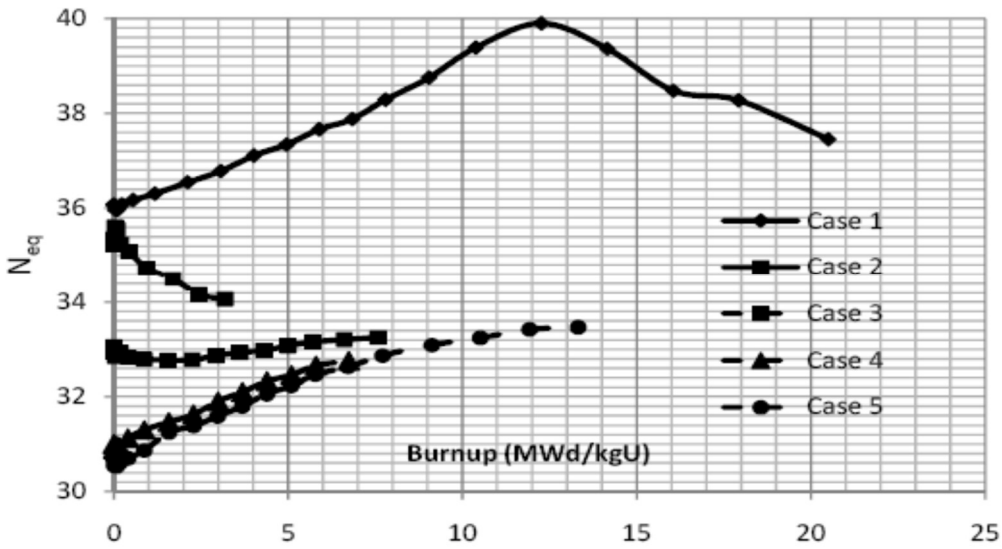

FIG. 3. N_{eq} parameter as function of the burnup for the five studded cases.

Fig. 4 shows the behavior of the Coolant Void Reactivity (CVR) with the burnup for the five cases. The calculations were carried out for none-leakage reactor for the comparison however the change in non-leakage probability during LOCA adds negative reactivity. Use the mid-burnup CVR for the comparison, we find that ACR-700 has a slightly negative CVR while CANDU-6 fueled with natural or spent fuel has a mid-burnup CVR around 13 mk as given in Table 1.

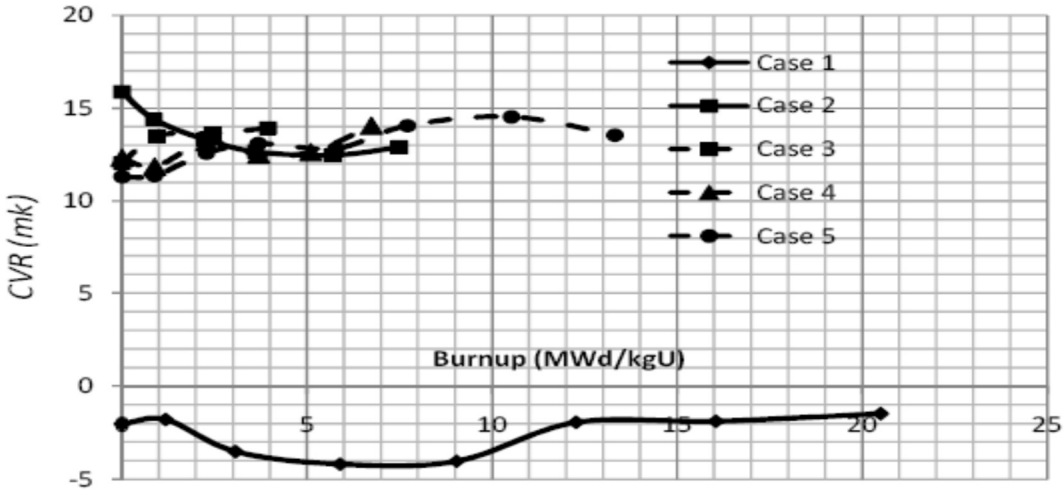

FIG. 4. CVR as function of the burnup for the five studied cases.

6. CONCLUSION

ACR-700 discharges the fuel with a significant amount of fissile isotopes (^{235}U, ^{239}Pu and ^{241}Pu). CANDU-6 and ACR-700 have the same inner diameter of the pressurized tubes with the same fuel bundle lengths and the same number of bundles per channel. This in turn enables to recycling the ACR-700 spent fuel bundles directly in CANDU-6. The gradient of fission rate from outer pins to the central pin in the ACR-700 makes the outer ring, which has the higher contribution of the fuel bundle reactivity, has the fewer amounts of fissile isotopes and therefore limits the fuel bundle reactivity at the discharge. Also, the residual reactivity plenty of the Dysprosium fissionable poisons in the central pin decreases the spent ACR-700 fuel bundle reactivity. Two alternative strategies are proposed for direct use of ACR-700 spent fuel. First is removing the central pin from the ACR-700 spent fuel and re-fabricating the spent fuel bundle into CANDU-6 fuel bundles using dry processing such as the DUPIC fuel cycle. Second is removing the outer fuel pins and the central pin and re-fabricating the rest two fuel rings into CANDU-6 fuel bundles using dry processing.

The calculations using the MCNPX code showed that recycling the ACR-700 spent fuel bundles directly in CANDU-6 gives less than 3 MWd/kgU while the alternative strategies gave 6.75 and 33 MWd/kgU, respectively. Normalizing the burnup on the all fuel in the spent bundle, additional burnup of 6.5 and 6.7 MWd/kgU can be obtained for the two alternative strategies respectively. This means that the second alternative strategy gives the higher burnup from the spent fuel which represents an increase of 33% in the fuel burnup. Moreover in this strategy, only about half of the spent fuel bundle is re-fabricated into CANDU-6 fuel bundle which decreases the cost of re-fabrication. The calculations give acceptable power distributions on the spent fuel bundles and almost the same coolant void reactivity as the reference CANDU-6.

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SESSION 7

SAFETY OF THE DRY SPENT NUCLEAR FUEL STORAGE IN UKRAINE: SCIENTIFIC APPROACH AND RESULTS

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Abstract

The safety of The Dry Spent Nuclear Fuel Storage Facility (DSNFSF) on Zaporizhska NPP in Ukraine consists of three main components – Nuclear Safety, Radiation Safety and Safe Thermal Condition. Therefore for the increasing safety level on DSNFSF it is necessary to know all details and features of equipment operation especially in field of thermal processes. For detailed definition of thermal state of spent fuel assemblies during all period of storage the multi-stage approach was developed. This approach was used for the definition of thermal state of containers with spent nuclear fuel and containers group's thermal state on open storage platform with taking into account the outer factors influence and spent fuel assemblies heat generation capacity changing. In addition all calculations required solving the inverse heat transfer problems. In particular the inverse heat transfer problems were used for the placement of spent fuel assemblies in storage basket for minimization of maximal temperatures and for the identification of equivalent thermal physical properties of compound bodies (storage basket, spent fuel assemblies, fuel rods) for using its in numerical calculations. The results which were presented in this work are used for estimation and the improving of safety level of DSNFSF on Zaporizhska NPP.

1. INTRODUCTION

There are two main ways to store the spent nuclear fuel – in the dry or in the wet storage facility. According to IAEA's data [1] today 73% of the operated storage facilities have a dry type (Fig. 1). The safety of any Dry Spent Nuclear Fuel Storage Facility (DSNFSF) consists of three main components – Nuclear safety, Radiation safety and Safe thermal conditions. This work is devoted to third part of this complex conception, in particular to the definition of thermal state of the container for spent nuclear fuel storage.

The storage system is a complex one and provides some difficulties from thermal point of view at numerical analysis. Firstly there are all mechanisms of heat transfer in it: conductive, convective and radiative. Secondly all heat transfer problems are in complex geometrical form. And finally there are many factors which have inner and outer influence on this system: initial decay heat, the decay heat changes during storage, influence of weather factors (ambient temperature changes, wind influence, solar radiation and precipitation).

In engineering practice the definition of thermal state of containers with spent nuclear fuel is the widespread problem. It can be solved by two ways: natural experiment (full-scale test) or numerical investigations. The full-scale test of thermal processes on DSNFSF is very difficult and awkward problem because requires to observe the norms of nuclear and radiation safety.

More than 250 tons of spent nuclear fuel is produced by Ukrainian NPPs each year. Currently the problem of spent nuclear fuel storage in Ukraine is under process of solving. The temporary storage of spent nuclear fuel until accepting decision about disposal or reprocessing today is used for six WWER-1000 reactors of Zaporizhska NPP.

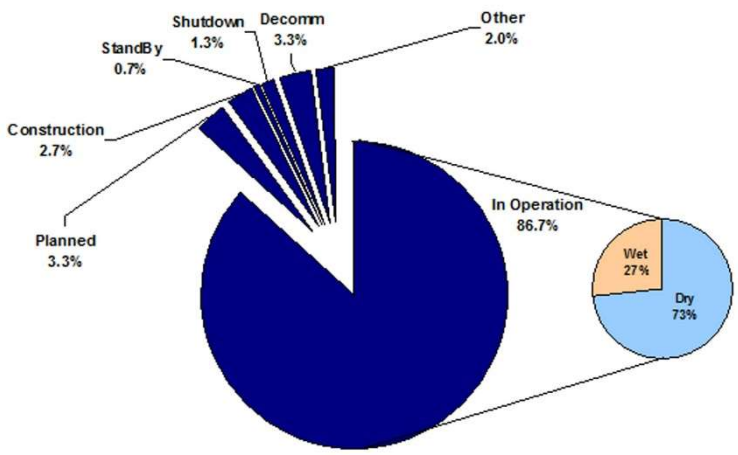


FIG. 1. Spent nuclear fuel storage facilities in the world.

The goal of this work is to describe scientific approach in field of thermal processes investigations which was used for safety estimation of DSNFSF on Zaporizhska NPP.

2. PROBLEM DESCRIPTION

Currently Ukraine has open nuclear fuel cycle which means that the spent nuclear fuel is not reprocessed but stored till making final decision about disposal or reprocessing. The Ukrainian Energy Strategy till 2030 provides for developing and construction the Central Spent Nuclear Fuel Storage Facility in the Chernobyl NPP's region. Moreover this document provides for operation the Dry Spent Nuclear Fuel Storage Facility for the six WWER-1000 reactors on Zaporizhska NPP [2].

The technology of the spent nuclear fuel storage in Ukraine now consists in the following. The spent nuclear fuel assemblies of six reactors of Zaporizhska NPP after storage in water pool are loaded into tight storage basket. The basket is placed inside concrete container and this system is moving to the open-air storage platform (Fig. 2). The basket with spent nuclear fuel is cooled by natural convection. The atmospheric air which comes through the ventilating system cools and removes decay heat from spent nuclear fuel.

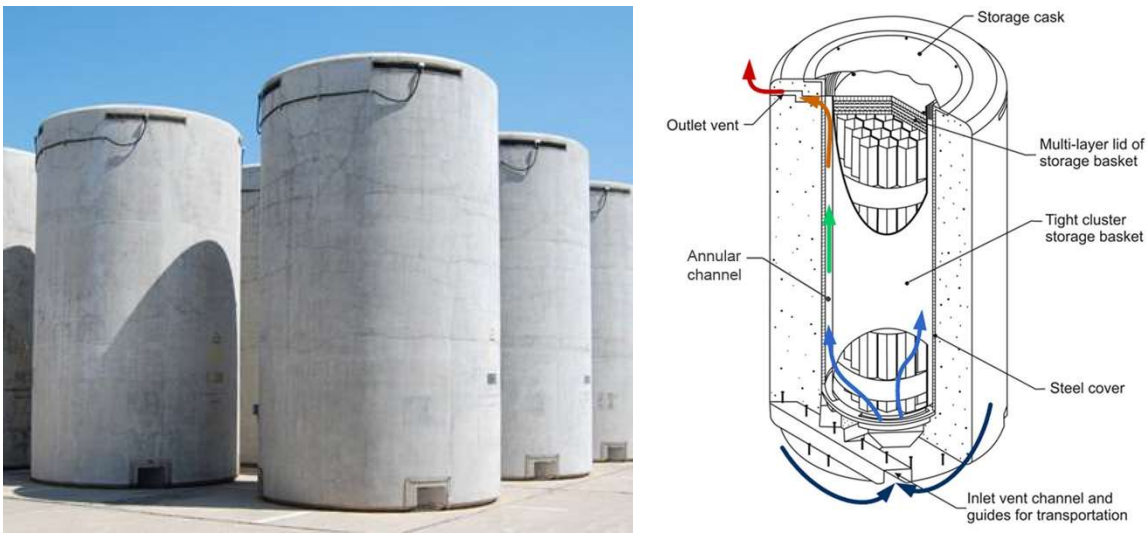


FIG. 2. Storage container.

Storage containers have the passive cooling system, so cooling only by atmospheric air takes place. The thermal safety criteria limit maximum temperature of spent nuclear fuel cladding inside storage basket by 350°C. The ventilating air on an outlet from channels should not be heated more than on 61 degrees. The temperature measurements are carrying out only on outlet of ventilation channels.

The storage facility is designed for 50 years operation. The most dangerous time is the beginning of storage when decay heat is maximal. For safety estimation the decay heat of each fuel assembly is taken 909 W that corresponds to decay heat after unloading from storage pool.

Usually in engineering calculations is used conservative approach which doesn't allow identifying detailed processes in equipment and gives approximate results. This type of calculations were used at safety ensuring for DSNFSF on Zaporizhska NPP [3]. For example in thermal calculations were considered only calm conditions of containers operation which are not typical for Zaporizhe region, the detailed structure of storage basket with spent fuel assemblies was not taken into account and as a result thermal and gas-dynamic processes inside it were not fully considered. So the obtained results include some "temperature margin" which means that calculated temperatures a little bit higher than in reality. These results were called "conservative". But for the high level safety ensuring on DSNFSF it is necessary to know all details and features of equipment operation especially in field of thermal processes. The usage of "temperature margin" will not allow to control all thermal processes which will take place, for example, in accident conditions of operation.

Methodology

For detailed definition of thermal state of spent fuel assemblies during all period of storage the multi-stage approach was developed. At usage of multi-stage approach on each next stage the part of investigated object is considered. The geometrical model of this part has more detailed structure in comparison with previous stage. At each next stage the boundary conditions for part of object are defining on base of thermal fields which calculated on previous stage. On each level of multi-stage approach the solving of conjugate heat transfer problems were used. The definition of thermal fields of solid body and fields of pressure,

velocity and temperature of the cooling or the heating medium at known inner and outer influences is the object of a direct conjugate heat transfer problem.

The developed approach was used for:

(1) The detailed definition of thermal state of containers with spent nuclear fuel;
(2) The definition of containers group's thermal state on open storage platform with taking into account their mutual influence, the influence of outer factors and heat generation capacity changing.

The aforementioned problems were solved numerically. the detailed descriptions of numerical methods, mathematical models, assumption and simplification is described in [4].

In addition all calculations which are mentioned above required solving the inverse heat transfer problems [5]. The inverse heat transfer problems were used for:

(1) The placement of spent fuel assemblies in storage basket for minimization of maximal temperatures;
(2) The identification of equivalent thermal physical properties of compound bodies (storage basket [6] spent fuel assemblies, fuel rods) for using its in simplified models.

3. RESULTS

Since Ukraine has only one DSNFSF the described approach was used for safety analysis of storage containers which are used on Zaporizhska NPP. For example, before construction of second stage of DSNFSF the analysis of thermal state of containers group, their mutual influence and influence of outer factors (the additional protection constructions and weather factors) was carried out. The results for line of ten containers is shown on Fig. 3. In spite of some mutual thermal influence at wind conditions the maximum temperatures in containers remain under dangerous level of 350°C.

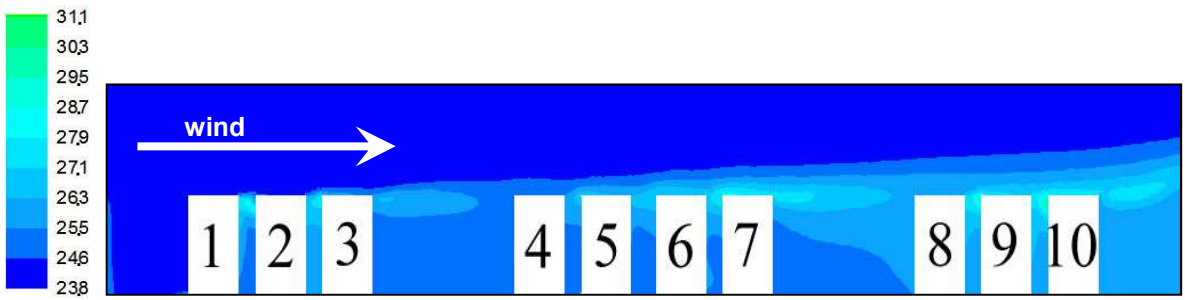


FIG. 3. Temperature field of ambient air which created by the group of containers at wind condition (5m/s) of operation.

Another application of developed approach was in calculation of detailed thermal state of fuel assemblies. It was caused by necessity to store "alternative" fuel assemblies with higher decay heat after storage in pool. For validating the safety of storage the series of calculations were carried out. The temperature fields of the container and fuel assemblies inside at calm condition of storage is shown on Fig. 4. These results allow to detect the hottest fuel assemblies and the hottest fuel rods in each assembly. This information could be used at the

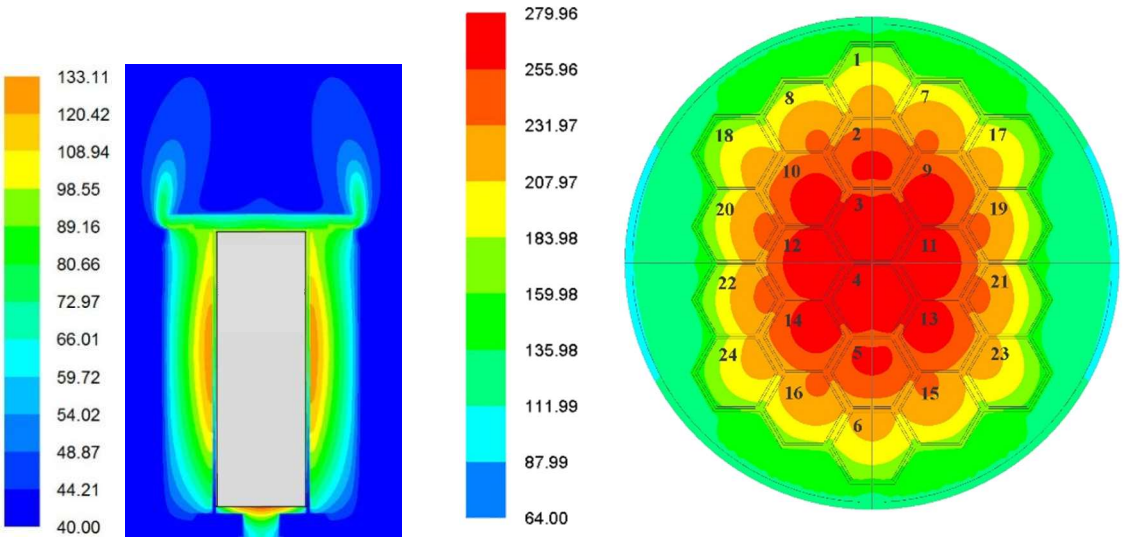
loading of storage basket or at the development of safety precautions during all period of storage.

In engineering practice the developers usually carry out only single calculation of object without any simplifications. It allows to obtain full information about real processes in equipment but doesn't give information about general physical laws which take place in considered object. The scientific approach in thermal analysis of containers with spent nuclear fuel imply the carrying out the series of calculation for different components of the storage system. It requires the replacement some parts of the modelled equipment by the simplified one with equivalent thermal properties. So the determination of equivalent thermal properties is very important problem.

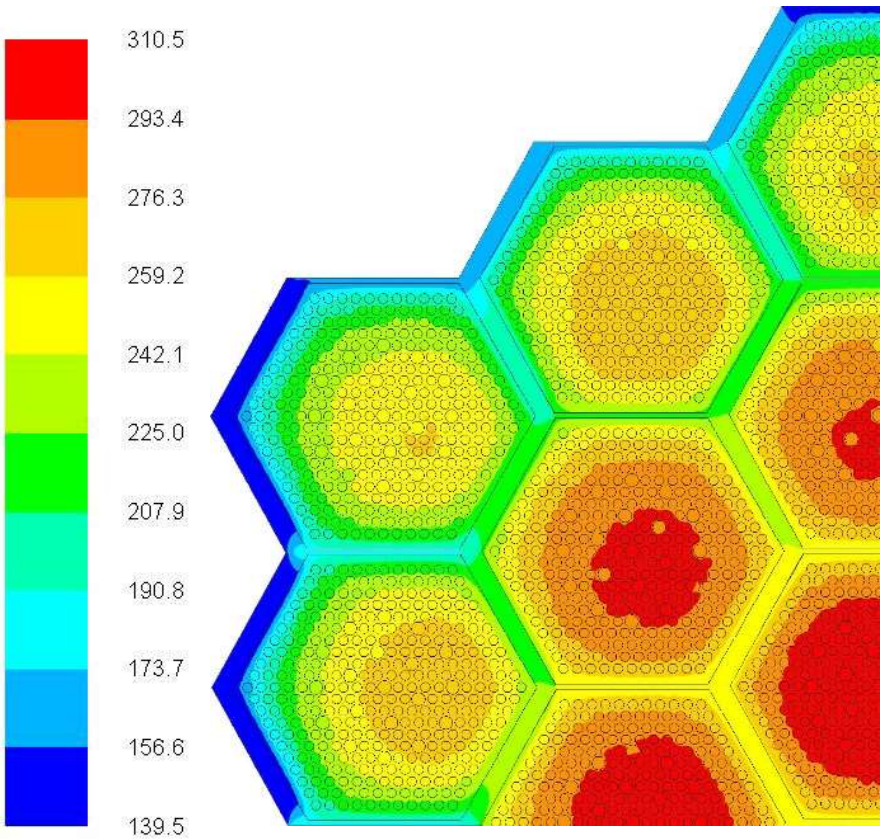
For thermal analysis of group of containers with spent nuclear fuel the storage basket with spent fuel assemblies is replaced by solid body. For the non-time-depended calculations the equivalent heat conductivity of storage basket which presented by solid body has to be detected. The values of equivalent heat conductivity for different time of storage is shown on Fig. 5. The value is changed during time of storage because the decay heat of fuel assemblies is decreased.

The multi-stage approach togeather with the inverse heat transfer problems was used for development of methodology of the spent fuel assemblies placement inside storage basket. The "traditional" and "alternative" fuel assemblies could be loaded inside storage basket but at any types of operational conditions the critical level of maximal of temperatures should not be reached.

The calculations of thermal fields of spent fuel assemblies with different level of decay heat (more and less heated assemblies) were carried out. Some results of these investigations are presented on Fig. 6. The hottest "alternative" fuel assembles was proposed to place near the basket shell. But this type of placement before usage should be checked by the radiation calculation.



(a) (b)



(c)

FIG. 4 Thermal state of storage container: (a) spent fuel assemblies; (b) fuel rods; (c) inside storage basket at beginning of storage.

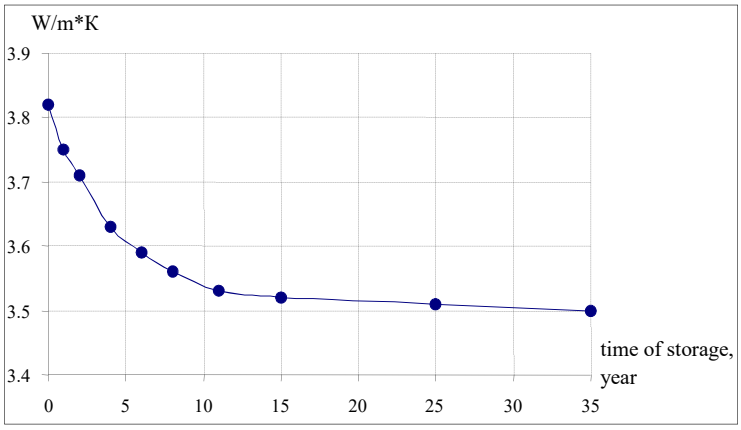


FIG. .5 Equivalent thermal conductivity of storage basket for different time of spent nuclear fuel storage.

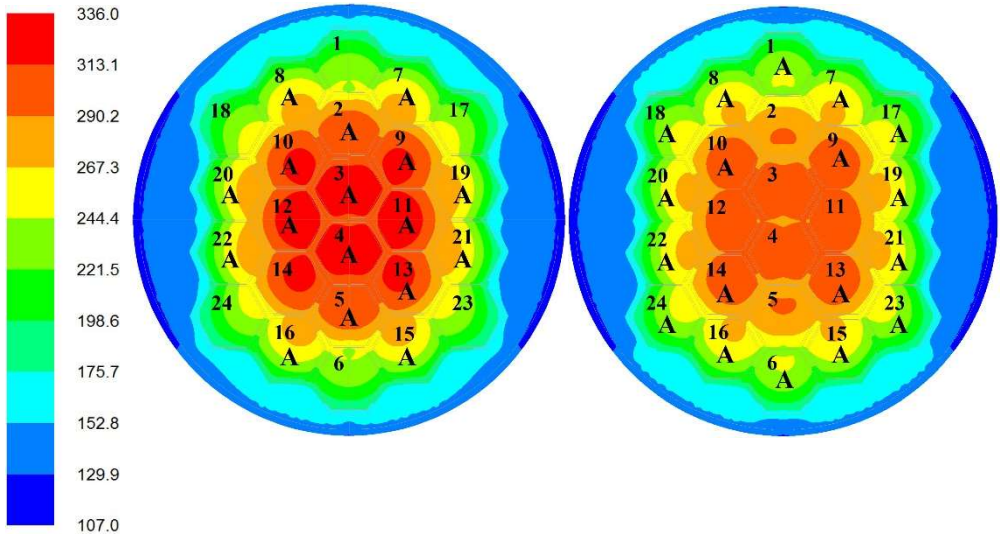


FIG. 6. Temperature field which created by traditional (numbers) and alternative (letter A) fuel assemblies with the higher decay heat after storage in pool.

4. CONCLUSIONS

The results which were presented in this work are used for estimation and the improving of safety level of DSNFSF on Zaporizhska NPP. But described scientific approach which based on solving of direct and inverse heat transfer problems has advantage because can be used not only for equipment on DSNFSF on Zaporizhska NPP. The formed mathematical models and developed methodologies of calculations can be used for other dry storage facilities with ventilated containers. The perspective of this work is in developing universal formulas for the calculations of spent fuel temperatures and developing programs for automatical control of thermal state of pent nuclear fuel during all time of storage.

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PRELIMINARY THERMAL MARGIN COMPARISON BETWEEN A SIMPLE AND A COMPLEX SF MODELS FOR A DRY STORAGE CASK

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Abstract

It is expected that amount of nuclear spent fuels will exceed the capacity of temporary storage pool in each nuclear power plant in Korea after several years. And, demand of the nuclear industry for storage cask commercialization has been increased. According to the demand, KINS (Korea Institute of Nuclear Safety) began a research project to develop appropriate technical guidelines for preparation for licensing of a dry storage cask. As a part of the project, a technical guideline for thermal analysis using CMS (Computational Modeling Software) is expected to develop. Given the growing industry need to store nuclear spent fuel of increasingly higher burn-ups and heat loads in dry storage casks, a conservative analysis model, for example, a homogenized model, is being substituted for a complex model, for example, rod based model, to secure more thermal margin. In this study, there were several assumptions that the PWR nuclear fuel assembly model was designed based on a CE-type-16-by-16 PWR nuclear spent fuel and the fuel assembly has been cooled for 10 years in a temporary storage pool, so the decay heat generation of the spent fuel assembly was 938.8 W, and the PWR dry storage cask model was designed based on a commercial cask which loads 24 PWR spent fuel assemblies. To compare the thermal margin between the homogenized model and the rod model, 4 cases CMS model: 1) 1/4 cask homogenized model; 2) 1/8 cask rod model; 3) one spent fuel assembly homogenized model; and 4) one spent fuel assembly rod based model, were calculated. Consequently, it is expected to contribute an establishment of CFD thermal analysis methodology using the complex analysis model and development of thermal analysis technical guideline using CMS.

1. INTRODUCTION

Nuclear power plants (NPPs) have been operating over 30 years in Republic of Korea and the temporary storage capacity for spent fuels is expected to be full in few years. Furthermore, decision making process to build a permanent disposal facility has been delayed. Even if the decision had been made, it would take several decades to begin operation of the disposal facility. Therefore, dry storage system has been considered as an interim storage system to manage spent fuels in Korea.

Given the growing industry need to store nuclear spent fuel of increasingly higher burn-ups and heat loads in dry storage casks, a conservative calculation model, for example, a simple / homogenized model, is being substituted for a complex/rod to rod model to secure more thermal margin.

In this study, preliminary thermal analyses with a simple / homogenized model and a complex / rod to rod model were performed to fine a thermal margin and additionally, thermal property functions for CMS are supported.

2. THERMAL MARGIN COMPARISON

2.1. Selection of representative PWR spent fuel in Korea

The Korea standard nuclear spent fuel, which has been proposed by KAERI [1], has been selected in this study. And it is assumed that the PWR spent Fuel is cooled for 10 years. Table 1 show the specification of the Korea Standard PWR Spent Fuel.

TABLE 1. THE SPECIFICATION OF REPRESENTATIVE PWR SPENT FUEL

Item (Unit)	Dimension
Assembly array	16×16
Enrichment (wt. %)	4.5
Assembly area (cm^2)	21.4 × 21.4
Length (cm)	453
Burn-up (GWD/MTU)	55
Decay heat per assembly (cooled for 10 years)	938.8 W

2.2. Reference storage cask for PWR spent fuel and modeling

There are various types of dry storage facility for PWR spent fuels, for example, metal canister type, a concrete silo type, dry well type and so on. In this study, MPC-24 multi-purpose cask, manufactured by an American company called Holtec INC, was selected as a reference storage cask.

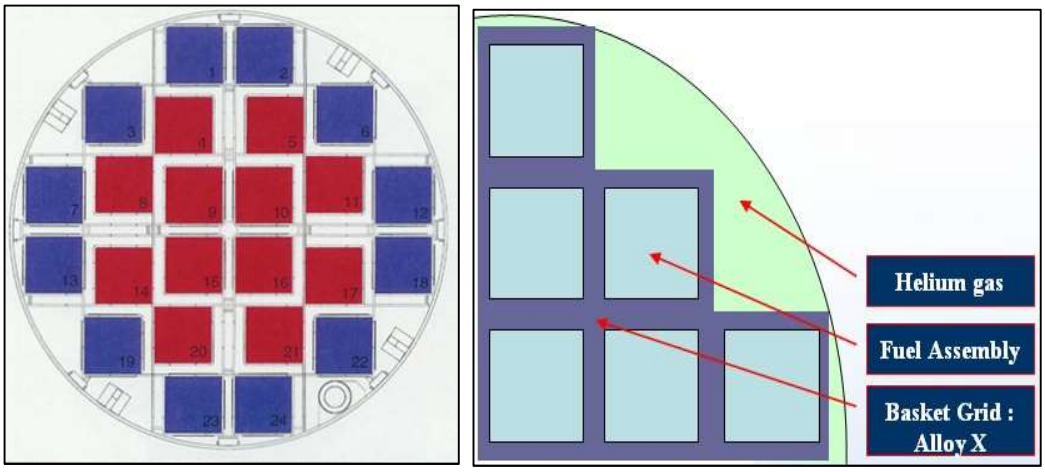

FIG. 1. Cross-section of MPC-24 and 1/4 analysis model.

To use CMS, thermal property functions related to temperature are required. Helium gas and stainless steel property functions are determined using Eqs (1-3):

Helium gas viscosity (kg/m·s):

$$V_H=7.6^{-6}+4.47^{-8}\times T_H-7.82^{-12}\times T_H^2 \quad (1)$$

Helium gas conductivity (W/m·K):

$$K_H=0.042+3.76^{-4}\times T_H-3.21^{-8}\times T_H^2 \quad (2)$$

Alloy X (SS304, SS316, SS304LN, ss316LN) conductivity (W/m·K):

$$K_X=8.617+0.017\times T_X-1.46^{-6}\times T_X^2 \quad (3)$$

2.3. Simple/Homogenized analysis model and thermal analysis results

The cask is composed of a shell, a lid, a grid, dummy weights and heaters. In the cask, helium gas is filled. Helium is an inert gas that has good heat conductivity. Supposing that the influence of natural convection on temperature distribution is small, heat conduction analysis is adopted for this area. Taking account of the radiation effect, an equivalent value of heat conductivity is used for helium. Heat is generated in the heater part homogeneously in the model. The grid is an assembly of stainless steel hollow block. But, in the model, it is assumed that the hollow block is a stainless steel block. [2]

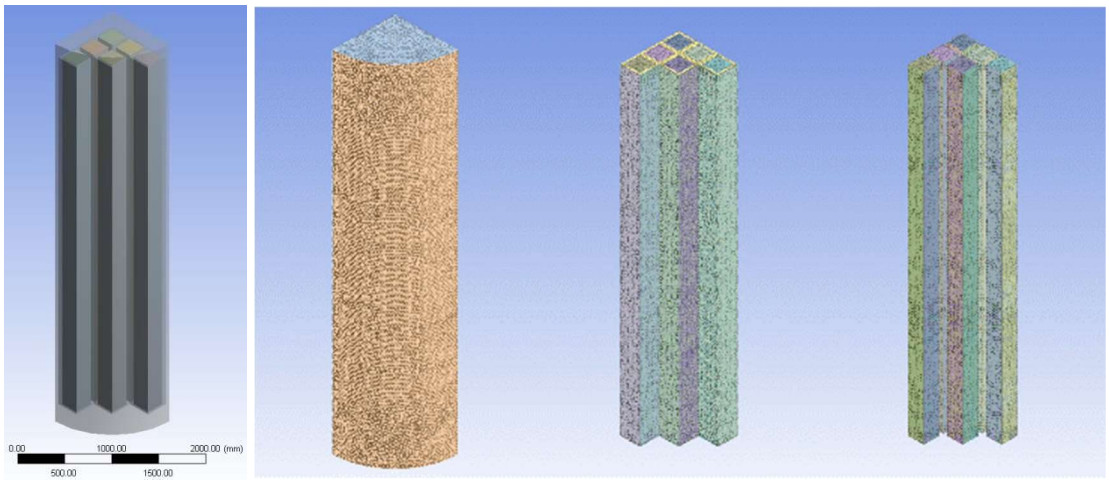

FIG. 2. 1/4 simple analysis model.

According to the ISG-11 (Cladding Consideration for the Transportation and Storage of Spent Fuel, Interim Staff Guidance-11) and NUREG-1567 (Standard Review Plan for Spent Fuel Dry Storage Facilities), the temperature limits of the PWR fuel cladding, within normal operating condition and accident condition, are 400℃ and 570℃, respectively.

In this study, the total decay heat generation is 22.5 kW. Concerning the boundary condition of the inner wall of the cask, fixed temperature (400 K, 500 K, 600 K, 700K and 800 K) conditions are applied to the wall except for the symmetrical plane. The calculation was a steady state analysis. Fig. 3 shows the result of this calculation. The result shows that with the normal operation condition, the limit temperatures at the inner surface of the 24-PWR is 293℃ and with the accident condition, the limit temperatures at the inner surface of the 24-PWR is 494℃.

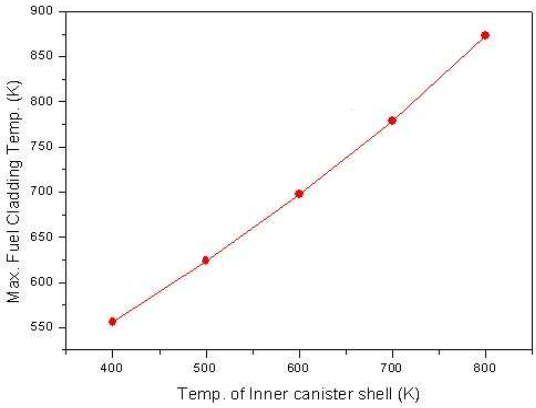

FIG. 3. Result of the thermal analysis with simple analysis model.

2.4. Complex / Rod to rod analysis model and comparison

The complex model is 1/8 symmetric model of the cask (Fig. 4) and it includes 1/8 rods and instrumentation tube of one spent fuel assembly. As boundary conditions, fixed surface temperature condition is given as 500K. The result shows that the complex model has 0.4% thermal margin more than the simple model. In this case, to minimize the convection effect, the model length was intendedly short.

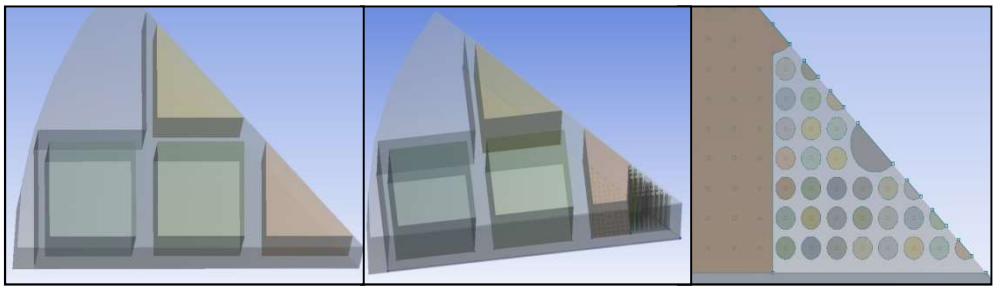

FIG. 4. 1/8 complex model.

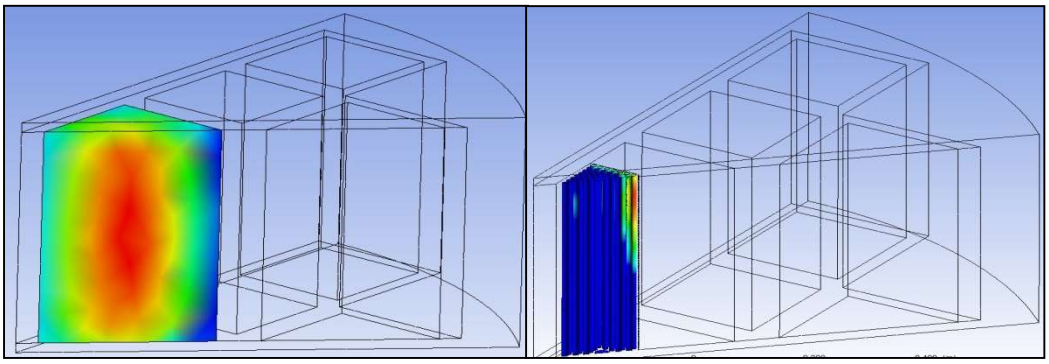

FIG. 5. Result of the thermal analysis with complex model.

In additional case, for whole spent fuel assembly, complex model and simple model were compared in same boundary condition above.

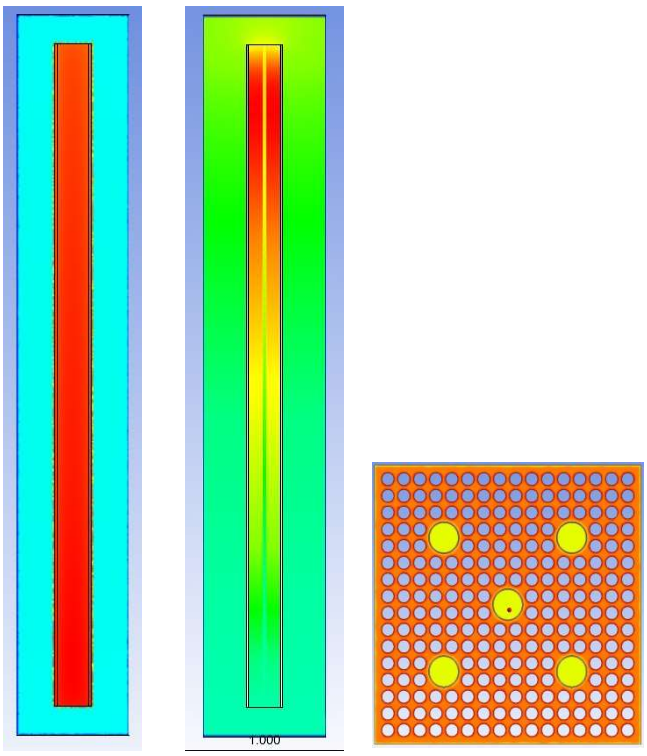

FIG. 6. (1) Simple model; (2) Complex model; (3) Cross-section of complex model.

The result shows that the complex model has 22.5% thermal margin more than the simple model.

3. CONCLUSION

In this study, to compare the thermal margin between the simple/homogenized model and the complex/rod to rod model, 4 cases thermal analyses with CMS model: 1) 1/4 cask homogenized model; 2) 1/8 cask rod model; 3) one spent fuel assembly homogenized model; and 4) one spent fuel assembly rod based model, were performed using CFX 12.0.

The result shows that the complex model secures noticeable thermal margin than the simple model because of convection effect and instrumentation tubes (structural advantage). Therefore, an analysis model like the complex model will be used for licensing. However, due to a real nuclear spent fuel contained structural complex components, for example, spring, dimple, grid vane, and so on, structural influence by the complex components has to be reviewed

Consequently, it is expected to contribute establishment of CFD thermal analysis methodology using the complex analysis model and development of thermal analysis technical guideline using CMS.

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STUDY ON TEMPERATURE ESTIMATION METHOD OF PWR SPENT FUEL CLADDING IN DRY STORAGE

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Abstract

In the Recyclable Fuel Storage Center, which is the first away from reactor (AFR) spent fuel storage facility in Japan, spent nuclear fuel (SNF) from BWRs and PWRs are scheduled to be stored in Dual Purpose Casks (DPCs). In the case of DPC storage in this center, if no abnormality in the external appearance or the inter-lid pressure of DPC is found during the storage period, the DPC and the stored fuel will be considered to be intact throughout the storage period, and the DPC will be shipped without opening its lids for pre-shipment inspection. Therefore, integrity of stored fuel during storage should be confirmed without opening DPC lids. Creep deformation and hydride reorientation are considered as major deterioration mechanisms of fuel cladding during dry storage, and they depend on the cladding temperature history. In this paper we focus on the fuel cladding temperature change during a long-term storage test using a test container which can store two PWR SNF, conducted by the electric utilities in advance of the commissioning of actual storage at the Recyclable Fuel Storage Center, and explain a result of the review for the temperature prediction performance of the analysis tool which we developed as an approach for fuel cladding temperature estimation by measuring the surface temperature of test container. The performance of the tool was reviewed by comparing the analytical results with the measured temperature distributions of the test container in the preliminary heat transfer test.

1. INTRODUCTION

Japan's first AFR dry storage facility for SNF, the Recyclable Fuel Storage Center in Mutsu-shi, Aomori, has completed its construction, and dry storage of BWR and PWR SNF by using DPCs is under safety review based on the new regulatory requirements. In this center, if there would have been no mechanical impact such as DPC overturning or fluctuation of the inter-lid pressure during the storage period, integrity of DPC and stored fuel will be considered to be maintained throughout the storage period, and DPC will be shipped without opening DPC lids for pre-shipment inspection. So the evaluation for integrity of stored SNF is one of the key technologies for DPC storage.

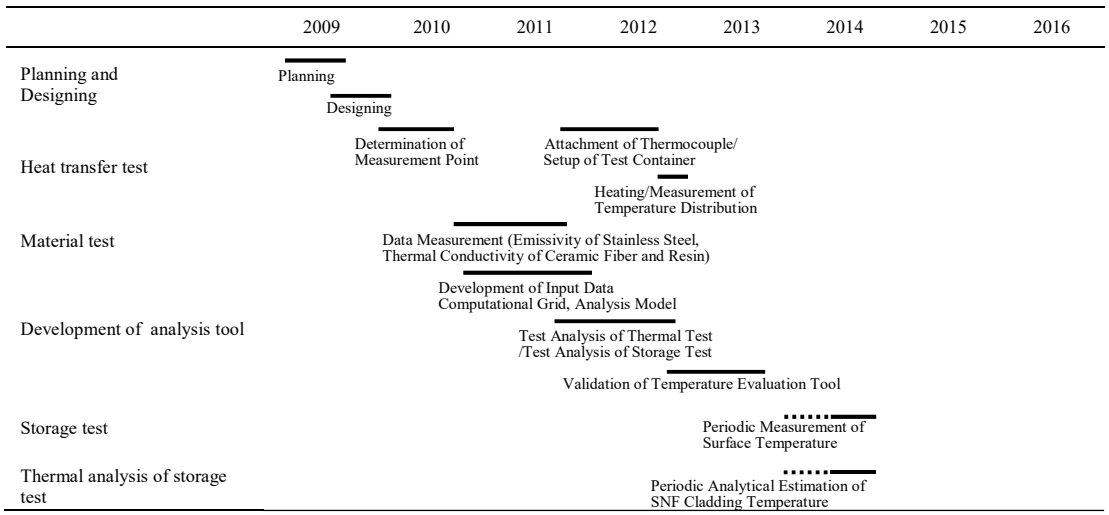
Japan has experiences of at reactor (AR) BWR SNF dry storage in metal casks and of the integrity check of the stored BWR SNF by opening DPC lids. Meanwhile, due to the lack of PWR SNF dry storage experience, the electric utilities planned a long-term storage test campaign in advance of the start of actual storage at the Recyclable Fuel Storage Center. In this test, 42.8 GWd/t and 55 GWd/t SNF are to be stored in the test container; 42.8 GWd/t SNF is to be stored from the beginning of the test and 55 GWd/t SNF will be added from 10 years after the beginning of the test. It is important for PWR SNF shipment after storage without opening DPC lids for pre-shipment inspection that the integrity of PWR SNF will be demonstrated through this test.

Therefore, Regulatory Standard and Research Department, Secretariat of Nuclear Regulation Authority (S/NRA/R) has developed and reviewed the analysis tool independently from the utilities to estimate the cladding temperature through measurement of the surface temperature distribution of the test container with the purpose for comprehension of stored SNF cladding temperature behavior that has major effects on the integrity of SNF during storage. The performance of the tool was reviewed by comparing the analytical results to the measured temperature distributions of the test container in the preliminary heat transfer test with electric heaters.

2. TEST PROGRAM FOR LONG-TERM STORAGE OF PWR SPENT FUEL

A time schedule of the PWR SNF storage test is shown in Table 1. A planning of this test and designing of the test container started in 2009, and the storage test with actual SNF has commenced in 2015. In advance of actual storage test, the heat transfer test was implemented by utilities with the purpose to comprehend heat transfer characteristic of the test container in the beginning of 2014. We carried out the tests to obtain material property of the test container components from 2011 to 2012; the obtained properties were used as the input data to the analysis tool. The development of the analysis tool was started from 2011; analytical models for the heat transfer test and the storage test were provided. The performance review and modification of the tool by comparing the analytical results to the measured temperature distributions of the test container in the heat transfer test were conducted.

TABLE 1 TIME SCHEDULE OF PWR FUEL STORAGE TEST



2.1. PWR spent fuel for the storage test

In the storage test, two SNFs are to be stored; 42.8 GWd/t SNF is to be stored from the beginning of the test and 55 GWd/t SNF will be added 10 years after the beginning of the test. The cooling time of 42.8 GWd/t SNF is 21 years as of 2015 and that of 55 GWd/t SNF will be 12 years as of 2025.

2.2. Test container

The outline drawing of the test container is shown in Fig. 1. Up to two 17×17 type PWR SNF can be accommodated in the test container. The dimensions of the container are; 5.2 m in total height; 1.7 m in the shell external diameter and 2.1 m in the thermal insulator external diameter. To shield gamma and neutron from the SNF, required shielding thicknesses for each container part are guaranteed, and neutron shielding material is installed. And to maintain the SNF subcritical, borated aluminum is used for basket spacer. In the heat transfer test and the storage test, thermal insulators are installed on the surface of the inner container and on the outer shell to raise SNF temperature to the target value. In the initial stage of the heat transfer test which models the storage test with single SNF stored and/or the storage test of single heating and/or single SNF storing, thermal insulator is installed on the surface of the outer shell and the cavity of the container is maintained as a vacuum because the heating rate of the

single SNF is too small to raise the SNF temperature to that in actual dry storage condition. In the second stage of the heat transfer test, the thermal insulator is removed. The ports for internal pressure monitoring, vent, drain are installed in the lid.

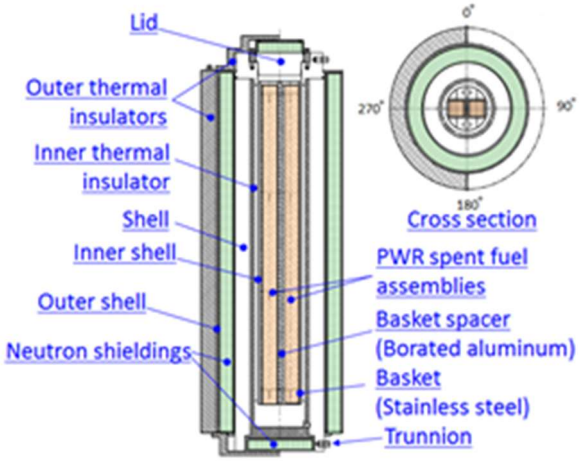


FIG. 1. Outline drawing of test container.

2.3. Heat transfer tests

To comprehend the heat transfer characteristic of the test container, two cases of heat transfer test were conducted using two mock-up heaters to simulate SNF heating rates. As in the storage test single 42.8 GWd/t SNF is stored from the start and 55 GWd/t SNF will be added in 10 years after, the temperatures of the container and the mock-up heater were measured for the single fuel case (Case 1) and for the double fuel case (Case 2). One heater models SNF geometry in detail (Heater-D), and the other is simplified (Heater-S). In Case 1 which simulated the initial stage of the storage test, the heating rate of 520 W was applied to model the heating rate of the 42.8 GWd/t SNF and internal pressure was kept at approximately 2 Pa. In Case 2 which simulated the 10 years after the beginning of the storage test, the heating rates of 440 W and 980 W were applied to model the heating rate of the 42.8 GWd/t SNF and the 55 GWd/t SNF respectively and internal pressure was kept at approximately 70 kPa. The heat transfer test conditions are shown in Table 2. Thermocouples were placed on the container and the heaters as shown in Fig. 2. 123 thermocouples in total were attached to measure the temperatures of the test container and the heaters.

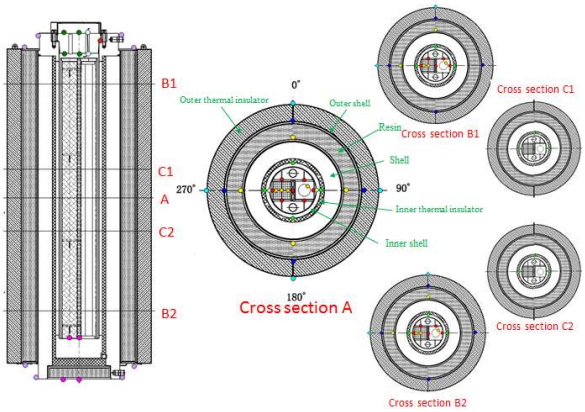


FIG. 2. Thermocouple placement

Each heater simulated only total heating rates of SNF. An axial distribution of the actual SNF heating rate was not modeled by the heaters. The Heater-D modeled SNF geometry and dimension, but the major materials of heater pins modeling fuel rods were stainless steel (cladding) and oxidized magnesium (MgO, insulant). The heater pins were surface-treated to the emissivity of 0.92 (measured value). Thermal conductivity of MgO was approximately 9 $W \cdot m^{-1} \cdot K^{-1}$ (measured value).

TABLE 2. HEAT TRANSFER TEST CONDITIONS

	[Case 1] Single mock-up fuel heating	[Case 2] Double mock-up fuel heating
Heat load	520 W	1420 W(440 W+980 W)
Container atmosphere	Rarefied Helium	Helium
Inner pressure	Approx. 2 Pa	Approx. 70 kPa
Outer thermal insulator	installed	—

2.4. Material tests

In order to provide input data for temperature analysis, the material tests to obtain thermal properties (thermal conductivity and emissivity) of the test container component materials whose properties are unavailable or unreliable in literature were conducted from 2011 to 2012. In the case of thermal conductivity, ceramic fiber used as inner insulator and resin used as neutron shielding material were subjected to the tests. In the case of emissivity, carbon steel used as lid and shell, and stainless steel used as basket were measured.

3. DEVELOPMENT OF ANALYSIS TOOL

The analysis tool has been developed to estimate stored SNF cladding temperature from the surface temperature distribution of the test container. This analysis tool consists of computational codes and sets of input such as physical model, boundary conditions, computational grid, and material properties. The performance of the analysis tool was reviewed by comparing the analytical results with the measured temperature distributions of the heat transfer test.

3.1. Computer codes

In this tool, a thermal fluid dynamics analysis and radiation heat transfer analysis are coupled. A commercial CFD code ANSYS FLUENT [1] is used for thermal fluid dynamic analysis. A code named S-FOKS [2], developed by the NRA for high precision radiation heat flux calculation, is used for radiation heat transfer analysis. The S-FOKS receives the FLUENT calculation result of surface temperature distribution as boundary condition, then calculates surface radiation heat flux and feeds it back to FLUENT. Through iterative calculations, heat conduction, natural convection and radiation heat transfer of the test container will be solved.

3.2. Physical models

3.2.1. Thermal fluid dynamics models

A fluent 3-dimensional thermal fluid dynamics model is used for natural convection calculation of infilled helium gas and air in the gap between inner insulator and shell. model is used as a turbulence model. Since a driving force of natural convection is given by a density difference, equation of state which gives density as a function of temperature is needed to be applied. So equation of incompressible ideal gas pre-installed in FLUENT is applied to helium gas. A Boussinesq approximation model is applied to air. Natural convection of helium inside the SNF assembly is not modeled, became conduction is the dominant heat transfer mechanism there.

3.2.2. Radiation heat transfer model

A radiation heat transfer model used in S-FOKS is able to simulate the radiation through transparent medium like helium and air with high precision. The S-FOKS is able to execute parallel computation with arbitrary number of processors so that the calculation time does not build up excessively even if the number of surface cells is huge.

3.2.3. Thermal conductivity of homogenized fuel model

Since a SNF assembly consists of a numbers of fuel rods, it is difficult to calculate with actual geometry of horizontal cross section of SNF model due to a limitation of computing capacity. For this reason, fuel assembly is modeled as homogenized solid. Separate calculations are made with an actual fuel assembly cross section model to set equivalent thermal conductivities to the horizontal cross sections of the homogenized solid. Examples of the analysis results of cross sectional temperature distribution are shown in Fig. 3 and the calculated thermal conductivities are shown in Table 3. In Case 1 where the cavity of the container is vacuum state, since the effect of thermal conductivity is negligible, a temperature distribution of the fuel assembly cross section is not parabolic shape but flat and there is a sharp temperature gradient on the fringe of the fuel assembly. So, cross section of the fuel assembly model is divided into two parts, central part and outside part. An equivalent thermal conductivity is applied to central part and the equation of radiation heat transfer between parallel plates is applied to outside part. In Case 2 where the helium pressure is sufficiently high, the equivalent thermal conductivity of the homogenized solid is determined by the result of temperature distribution analysis using FLUENT and S-FOKS with 2-dimensional detailed fuel assembly model.

The axial thermal conductivity of homogenized solid is determined by volume averaging of thermal conductivity of each material.

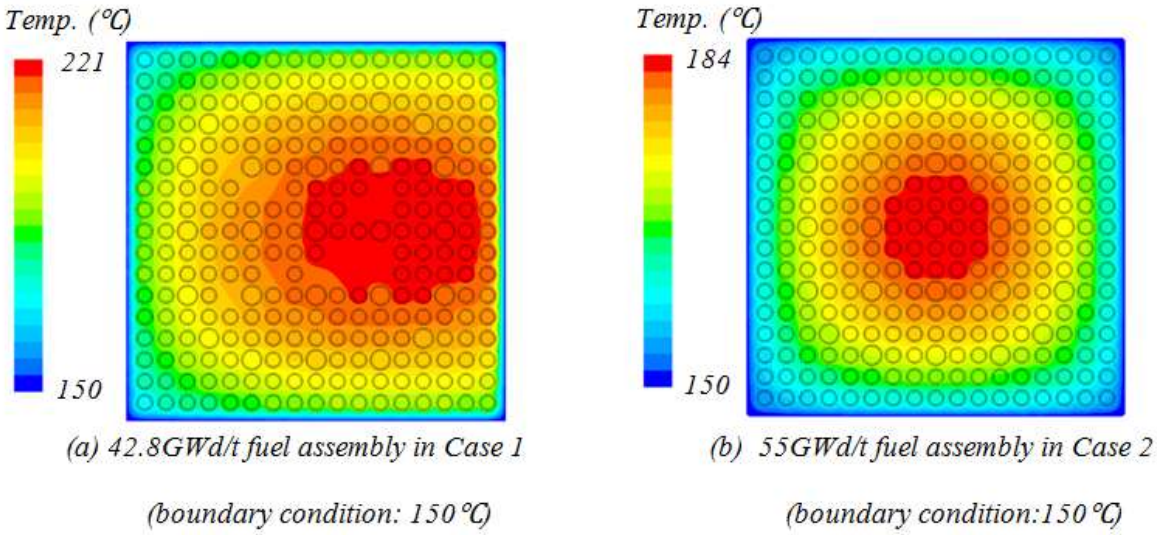


FIG. 3. Examples of temperature analysis results for SNF cross section.

TABLE 3. EQUIVALENT THERMAL CONDUCTIVITY OF SNF CROSS SECTION

Fuel cross-section mean temperature (℃)	Equivalent thermal conductivity ($W \cdot m^{-1} \cdot K^{-1}$)		
	Case 1 42.8 GWd/t fuel	Case 2 42.8 GWd/t fuel	Case 2 55 GWd/t fuel
-50	0.054	0.331	0.329
0	0.068	0.384	0.388
50	0.087	0.443	0.453
150	0.146	0.580	0.606
250	0.242	0.748	0.794

3.3. Boundary conditions

In the calculation using FLUENT, temperature distribution of the outer shell surface of the test container is set as boundary conditions. This temperature distribution is set up based on the data measured by thermocouples attached on the outer shell, lid and bottom plate of the test container and the surface temperature of each cell is set by using bi-linear interpolation.

3.4. Computational grid

The computational grids for the analysis tool used in FLUENT consist of approximately 2.4 million hexahedral computational cells. The computational grids for test container are shown in Fig. 4.

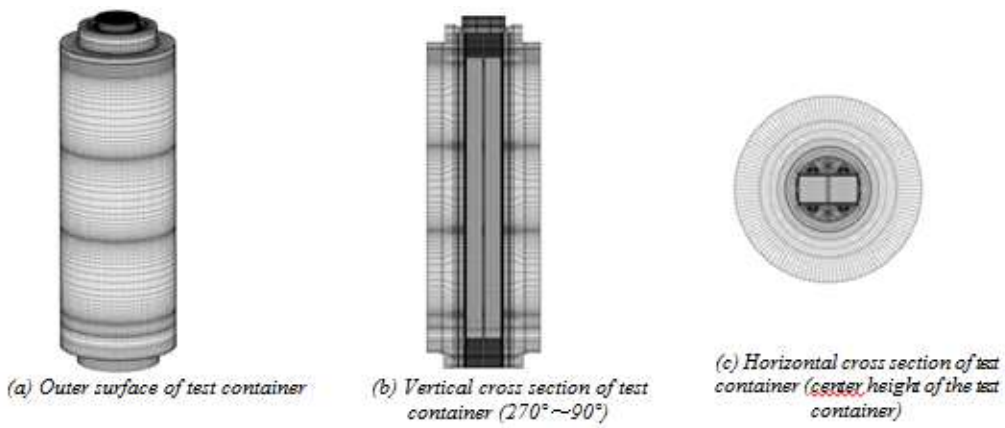


FIG. 4. Computational grids for test container.

3.5. Material properties

The thermal conductivities of following materials are taken from the literature or the test data listed:

- Stainless steel (basket, inner shell, lid, bottom plate and outer shell), carbon steel (shell): JSME Data Book: Heat transfer 5th Edition [3];
- Borated aluminum (basket separate plate): the data provided from the test container manufacturer;
- Ceramic fiber (inner insulator) and resin (neutron shielding material): the measured data in the material shown in Table 4 & 5.

The emissivity of stainless steel and carbon steel obtained from the measured data in the material test are shown in Table 6. Helium and air are treated as fluid; the specific heat, density, thermal conductivity and viscosity coefficient of the gasses obtained from the data listed in JSME Data Book: Heat transfer 5th Edition [3] and JSME Data Book: Thermo-physical Properties of Fluids [4].

TABLE 4. THERMAL CONDUCTIVITY OF CERAMIC FIBER AND RESIN

Temperature (℃)	Thermal conductivity ($W \cdot m^{-1} \cdot K^{-1}$)
	Ceramic fiber
28	0.0285
101	0.0363
202	0.0478
301	0.0602

TABLE 5. THERMAL CONDUCTIVITY OF CERAMIC RESIN

Temperature (°C)	Thermal conductivity ($W \cdot m^{-1} \cdot K^{-1}$)
	Resin
35	0.884
83	0.829
141	0.742

TABLE 6. EMISSIVITY OF STAINLESS STEEL AND CARBON STEEL

Temperature (°C)	Emissivity	
	Stainless steel (rough surface)	Carbon steel
28	0.284	0.021
101	0.290	0.026
202	0.298	0.035

4. RESULTS AND DISCUSSIONS

4.1. Cases and conditions of analyses

In advance of the heat transfer test, the preliminary analysis was conducted to evaluate the causes of the difference between measured and calculated temperature distribution. The models and the conditions of analysis were modified based on this evaluation.

4.2. Result of Case 1 (single mock-up fuel heating case)

4.2.1. Result of preliminary analysis

The preliminary analysis of Case 1 was conducted with the container surface temperature distribution shown in Fig. 5 as a boundary condition. The temperature distribution was set by bi-linear interpolations from the measured temperatures. The analysis results of temperature profiles are shown in Fig. 6 comparing with the measurements results. In this case, since the pressure in the test container was quite low (approximately 2 Pa, measured value), the thermal conduction and the natural convection in the cavity between top of the basket and inner surface of the lid was negligible. Therefore, the axial heat transfer in the top region of the container is prevented by its low thermal conductivity and low convection because of the low pressure (approximately 2 Pa, measured value). At the underpart of the test container, since a

ceramic fiber insulator layer exists, axial heat transfer was also disturbed significantly as with the case of the top. Consequently, as shown in Fig. 6 (a), the axial temperature distribution of the Heater-D is an axially symmetrical shape and has a peak at near the center height of the test container (cross section-A). The analysis result reproduced macroscopic characteristic of the actual temperature distribution and the temperature cross section-A agreed roughly with the measured data.

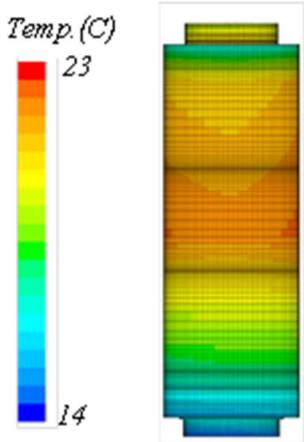
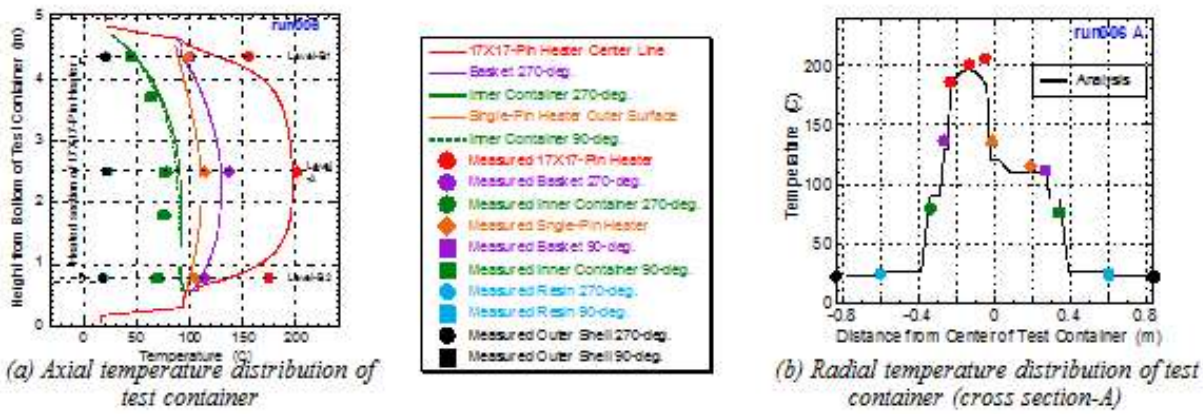


FIG. 5. Boundary condition of container surface in Case 1.

On the other hand, in the case of top and bottom of the Heater-D, the analysis results show 30℃ to 35℃ lower than the measured temperatures. According to Fig. 6 (b), the temperature difference between analysis result and measured temperatures of the central part at the cross section-A is within approximately 4℃, while for the temperature distribution in radial direction, the analysis result showed parabolic shape whereas measured temperatures showed asymmetric shape with peak shift toward basket separate plate.



(a) Axial temperature distribution of test container

(b) Radial temperature distribution of test container (cross section-A)

FIG. 6. Comparison of preliminary analysis result and measured temperature distribution data of Case 1.

4.2.2. Result of verification analysis

In the verification analysis, following items of the analysis model were modified based on the evaluation for the causes of the gap between the preliminary analysis result and measured data:

(1) Consideration of air natural convection in the gap between inner thermal insulator and shell:

In the preliminary analysis, the effect of natural convection in the gap area was disregarded. In the verification analysis, as addressed in 3.2.1, fluid model was added to the analysis model so that the effect of the natural convection was able to be reflected to the analysis.

(2) Modification of equivalent conductivities for top edge and bottom edge of Heater-D:

By reflecting detailed structure of top edge and bottom edge of heater, the equivalent thermal conductivities were set smaller.

(3) Modification of homogenized model for Heater-D:

In the preliminary analysis, cross section of the Heater-D model was divided into two parts: central part and outside one. For each part, different equivalent thermal conductivities were used. As stated in 3.2.3, the equation of radiation heat transfer between parallel plates was applied to outside part. This enables the analysis to model the emissivity difference between basket and basket separate plate to the temperature distribution.

(4) Modification of basket separate plate emissivity:

The emissivity on the surface of basket separate plate was reduced from 0.15 to 0.04 (lower limit of aluminum emissivity data listed in literature [3]).

The comparison between the verification analysis result and measured temperature data is shown in Fig. 7. As shown in Fig. 7 (a), the temperature difference between the analysis and the measured temperature of top edge and bottom edge of the Heater-D decreased to 10℃. As shown in Fig. 7 (b), the temperature difference between the result of analysis and the measured temperature data of central part of the Heater-D at the cross section-A decreased to 2℃. The temperature distribution in the Heater-D shows that the peak shifted toward basket separate-plate as observed in measured temperature distribution.

4.3. Result of Case 2 (double mock-up fuel heating case)

4.3.1. Result of preliminary analysis

A preliminary analysis of Case 2 was conducted with the container surface temperature distribution shown in Fig. 8 as boundary condition. The temperature distribution is set in the same manner as 4.2.1. A comparison of the analysis result and the measured temperatures is shown in Fig. 9. In this case, since the helium pressure in the test container was approximately 70 kPa, the thermal conduction and natural convection in the cavity between top of the basket and inside surface of the lid is effective. For this reason, axial heat transfer increased in the top of the inner shell compared to Case 1. Axial temperature distribution of

the Heater-D is asymmetric shape which has a peak at a level lower than cross section-A. The analysis result reproduces macroscopic characteristic of the temperature distribution.

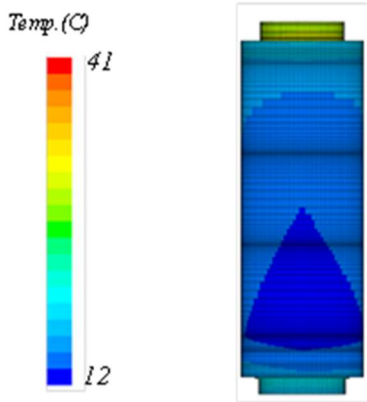
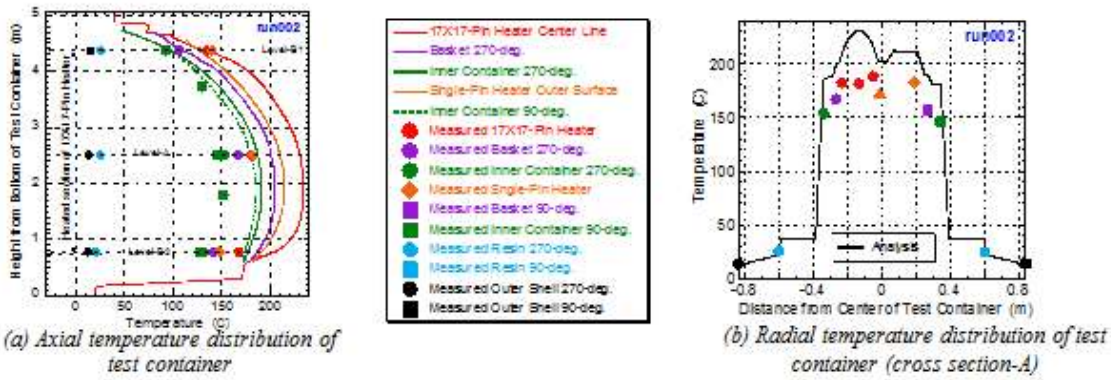


FIG. 8. Boundary condition of container surface in Case 2.

Fig. 9 (a) shows that the analysis results roughly agree with the measured data at the top of the test container. But at center height and lower part of the test container, inner shell temperature considerably exceeds measured data. The temperature at the center of the Heater-D at the cross section-A is 50℃ higher than the measured data. According to Fig. 9 (b), the analysis result of the radial temperature distribution of the Heater-D at the cross section-A has a parabolic distribution with a peak at center, while measured temperature shows flat distribution and the temperature at center was slightly lower than that of the outside part.



(a) Axial temperature distribution of test container

(b) Radial temperature distribution of test container (cross section-A)

FIG. 9. Comparison of preliminary analysis result and measured temperature distribution data of Case 2.

4.3.2. Result of verification analysis

In the verification analysis, the analysis model was modified as follows based on the results of consideration of the cause of the difference between the result of preliminary analysis and measured data.

(1) Consideration of air natural convection in the gap between inner thermal insulator and shell:

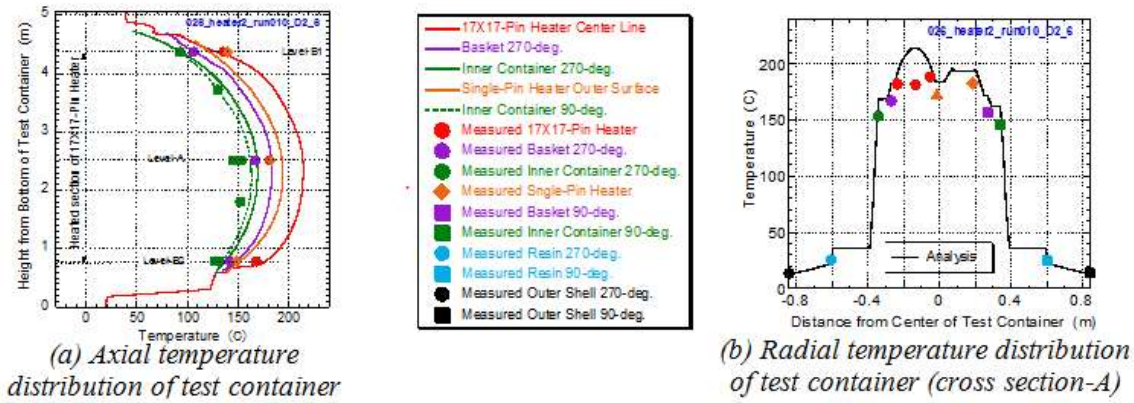
As with the Case 1, the effect of natural convection in the gap area was disregarded in preliminary model, but the fluid model was added to the verification analysis model so that the effect of the natural convection was able to be reflected to the analysis.

(2) Modification of homogenized model for the Heater-D bottom edge and emissivity of basket separate plate:

The same modifications as Case 1 were done.

As a result of the analysis with modified model the temperature of the inner shell at the center height of the test container (cross section A) notably decreased and the temperature difference between the analysis result and the measured data was reduced because of the consideration of air natural convection. But at the inner shell, the difference between the analysis and the measured is still large. As shown in Fig. 6(a), such temperature difference also occurred in Case 1 in the lesser degree. It was attributed to unexpected large heat transfer occurred from the inner shell to the shell at the bottom part of the test container. Therefore, the result shown in Fig. 8 was reviewed and the potential heat transfer path was investigated. As a result, the contact areas of shell and four bosses attached around the bottom plate of inner shell were found as a potential heat transfer path. Since the smallest gap between the top surface of the bosses and inner surface of the shell is 1.5 mm, even slight misalignment of the inner shell with the height of approximately 4 m would cause the contact of the bosses to the shell. Furthermore, the temperatures of bottom plate near the boss were 70℃ in Case 1 and 120℃ in Case 2, whereas the temperature of the shell was 20℃ in Case 1 and Case 2. Making a prediction of the gap between the bosses and the shell under the temperature condition like this, the gap distances will be reduced to approximately 0.5 mm due to the thermal expansion of the bottom plate to indicate potential contact to the shell.

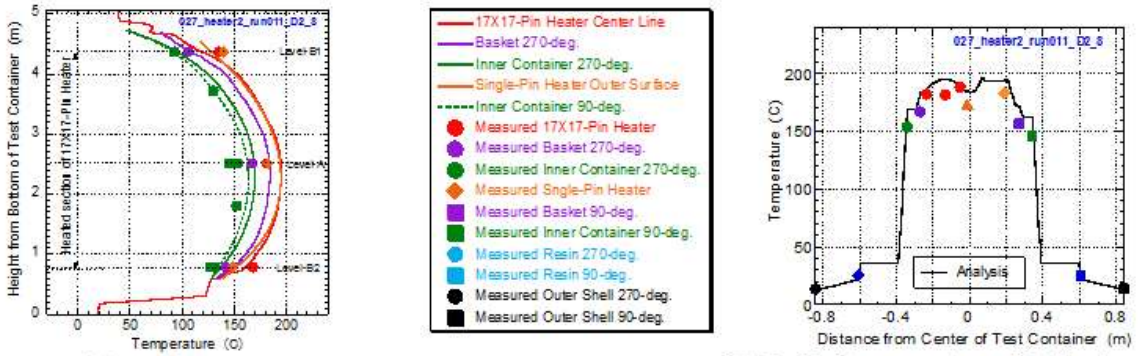
Based on above consideration, a follow-up analysis with the assumption that the contact of the four bosses with the shell was made to result that, as shown in Fig. 10, the temperature distribution at bottom part of the test container well agrees with the measured temperature distribution. However, according to Fig. 10 (b), the temperature of Heater-D center part at cross section-A is still approximately 35℃ higher than the measured temperature. It was found that the large difference between the analysis results and the measured data was caused by the significantly flattened temperature distribution of the measured data. Therefore, an effect was surveyed with the model of tentatively triplicated equivalent thermal conductivity of the entire axial length of heating part. As a result of this analysis, the temperature difference of the Heater-D center part at the cross section-A decreased to approximately 14℃ as shown in Fig. 11. Meanwhile, in this analysis, since the model was also modified with consideration to the gap between the basket and the side surface of the top gap above the Heater-S, the temperature difference of the analysis results and the measured data decreased at the top surface of the Heater-S compared to Fig. 10. On the other hand, as shown in Fig. 12, the radial temperature distribution of Heater-D at the bottom end (cross section-B2) is not in flat shape like that at the cross section-A, but parabolic shape as ordinarily expected. The same shape is observed at the top edge (cross section-B1). From these reasons, the flattening of temperature distribution as seen in the cross section-A can be said it occurs only at local parts of the Heater-D. It is difficult to identify the cause of the flattening, but since the temperature measurement points at the cross section-A is nearby the support grid of the heater pins, a possibility of the radial direction heat transfer via support grid can be pointed out.



(a) Axial temperature distribution of test container

(b) Radial temperature distribution of test container (cross section-A)

FIG. 10. Comparison of verification analysis result and measured temperature distribution data of Case 2.



(a) Axial temperature distribution of test container

(b) Radial temperature distribution of test container (cross section-A)

FIG. 11. Comparison of verification analysis No.2 result and measured temperature data in Case 2.

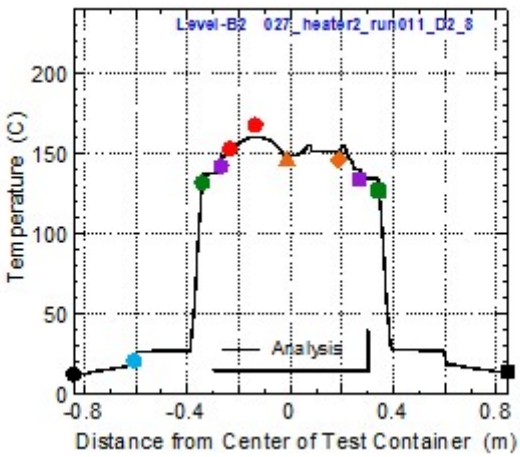


FIG. 12. Comparison of verification analysis result and measured temperature data at cross section-B2 in Case 2.

4.4. Summary of Case 1 and Case 2 test results

As a result of the comparative review of the analysis tool, the cladding temperature of the stored fuel can be predicted by measuring the surface temperature distribution of the test container. The review of the results of Case 1 and Case 2 are as follows:

Case 1:

- The difference between calculated result by this tool and the measured temperature is within 2°C;
- In case of the storage test, this tool will be able to predict the fuel cladding temperature with an uncertainty of 10 °C;
- As a preposition to the prediction, surface temperatures of the test container must be specified as boundary conditions with the same precision as the heat transfer test, and an uncertainty of decay heat and material properties must be excluded.

Case 2:

- In the heat transfer test, unexpectedly large heat transfer from the inner shell to the shell is presumed;
- A significant flattening of temperature distribution in radial direction of Heater-D is observed;
- Contact areas between inner-shell bosses and shell are assumed as a heat transfer path;
- A tripled equivalent thermal conductivity is tentatively input as a radial thermal conductivity of homogenized solid for a flattening of temperature distribution in radial direction of Heater-D.

As a result of above modification, the temperature difference between the analysis results and the measured data of the Heater-D decreased to approximately 15°C.

In future, further investigations for the phenomena in Case 2 are needed to finalize the analysis tool for reasonable estimation of fuel cladding temperature.

5. CONCLUSION

In advance to the storage test of PWR SNF, an analysis tool has been developed to estimate the SNF cladding temperature analytically from the measured temperature distribution of the test container surface. The performance for the temperature estimation by the tool was reviewed based on the measured data from the heat transfer test using electric heaters as mock-up fuel. The result of performance review showed that in the initial stage of the storage test with single SNF stored under extremely low internal pressure, this tool can estimate the cladding temperature of SNF with uncertainty of approximately 10°C. In the second stage of the storage test with double SNF stored in helium atmosphere, the uncertainty of estimation is expected to be smaller than approximately 15°C. The effect of the local contact of the container parts on the temperature distribution at the bottom part of the test container and the effect of radial direction heat transfer via SNF support grid on fuel temperature are studied to confirm the accuracy of temperature estimation by this tool.

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LOW COST GAMMA AND NEUTRON RADIATION SENSORS FOR REAL-TIME CASK MONITORING

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Abstract

A low-cost array of modular sensors for online monitoring of radioactive waste was developed at INFN-LNS. We implemented a new kind of gamma counter, based on Silicon Photo Multipliers and scintillating fibres, that behaves like a cheap scintillating Geiger-Muller counter. Front-end electronics and an FPGA-based counting system were developed to handle the field data, also implementing data transmission, a graphical user interface and a data storage system. Tests in a real radwaste storage site have shown quite encouraging results. We also developed a low-cost technique for thermal neutron detection not making use of ^{3}He, also suitable for online real time monitoring of spent fuel casks. As a neutron converter we used ^{6}LiF, being the neutron capture cross section of ^{6}Li very well-known and with only an alpha and a triton in the exit channel. We can deposit wide and thin layers of converter onto several different substrates, to be placed on top of solid state detectors or scintillators capable of efficiently detecting the decay products. Tests with neutron sources and with neutron beams have proved the full feasibility of these sensors. A combination of several units of the aforementioned gamma and neutron detectors can be exploited for spent fuel cask monitoring in place and/or during transportation, in order to contribute in the assessment of possible cask ageing with radiation loss, and to prevent possible cask tampering by ensuring the continuity of knowledge. Battery-operated versions of the sensors, with wireless data transmission, are currently under development.

1. INTRODUCTION

The term radioactive waste, or shortly radwaste, indicates waste containing radioactive material. It is usually produced as a residue of nuclear processes, as in nuclear fission reactors, even though industrial processes and medical applications may also give rise to radioactive waste. Contrary to chemical waste dangerousness, radioactivity decreases with time, therefore one could in principle seal and isolate the waste for some time until its radioactivity poses no relevant risk. Unfortunately the time scale we are talking about can range from a few hours to years for medical and industrial radwaste, and up to tens of thousands of years for high-level waste from nuclear fission reactors and nuclear weapons decommissioning. Moreover, the scientists obviously trust the exponential decay law of radioactivity, whereas this is not always the case for laypersons, politicians and the general population. As mishandled radwaste could contaminate the environment or, far worse, be used for the production of weapons, all over the world there are regulations prescribing how the radwaste has to be handled and stored.

Whichever its origin, the radwaste ends up into special drums which have to be placed into short/medium term repositories where they are periodically inspected by operators making use of more or less sophisticated tools and detectors. As it is customary, in several locations inside a repository a few specific radiation detectors and dosemeters allow to keep under control the local environment. As for the final destination of the long-lived radwaste, their confinement into geological repositories is still a matter of debate.

Nowadays the modern radiation detection and information technologies allow to set up granular distributed systems to collect activity data in real time, and to make them available by means of suitable databases and network connections. This is why at INFN-LNS we have proposed a system for the online real-time monitoring of the radioactive waste repositories, consisting of an array of many radiation sensors to be deployed all around the radwaste drums

[1, 2]. In order to be suitable for mass deployment these detectors have to be small, inexpensive, robust, easy-to-use and reliable. The sensors we are going to describe in this paper possess all these listed features. A logical scheme of the system is shown in Fig. 1.

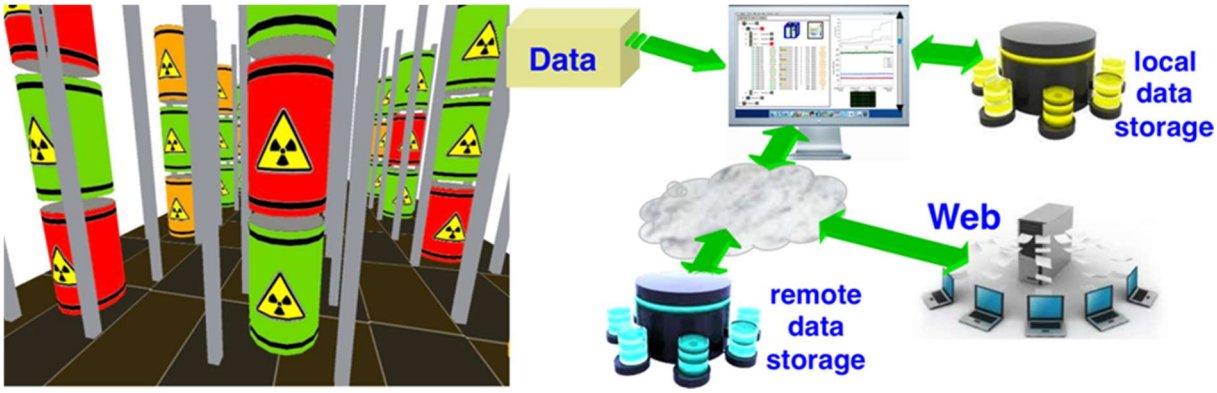

FIG. 1. Logical scheme of the radioactive waste repository monitoring system.

2. DETECTION OF GAMMA RAYS

The composition of the radwaste can be very different from case to case, and in general it is categorized according to its activity from Very Low Level Waste (ILLW) to Low Level Waste (LLW), Intermediate Level Waste (ILW) and High Level Waste (HLW). However, from the monitoring standpoint we can make a distinction between two main categories, apart from those arising from the activity, depending on the origin of the waste. Radwaste originated directly from the nuclear fuel, either reconditioned or not, is expected to emit neutrons, whereas other radwaste basically emits only gamma rays (alpha and beta radiation is stopped inside the material itself, unless the waste is dispersed in the environment). This means that in order to monitor the radwaste one has to focus mainly on detectors for gamma rays and neutrons.

As for the gamma ray detection, there is a wide variety of devices and techniques on the market, and typically a detector is from moderately to highly expensive, therefore not well suited for a granular monitoring. By exploiting the recently born silicon photomultiplier (SiPM [3, 4]), along with plastic scintillating fibers, we have developed a new gamma ray counter capable of fulfilling this task [5]. The scintillating optical fiber [6] is coupled at each end to a SiPM whose single photon sensitivity, along with the left-right coincidence constraint, allows to achieve a reasonable sensitivity to gamma radiation even though using a thin 1 mm diameter fiber. Simulation results are in perfect agreement with the measured behavior, and any improvement in the SiPM development technology will reflect immediately into an improvement of the detector performance.

The basic idea, in order to obtain a detector small, cheap, robust, easy-to-use and reliable, was to use well known materials and properties, avoiding too sophisticated technologies which would lead to expensive and delicate components. We decided to rely on scintillators, representing a well consolidated yet economic technology for radiation detection. In particular we chose the plastic scintillating fiber, which can be produced in large quantities at low cost, with very reproducible geometrical and physicochemical properties. Moreover, a radiation detector based on a scintillating fiber can be sized rather easily between few centimeters and

few meters long, still employing the same components, therefore being better suited to fit and watch wider objects (drums) than a typical localized detector [7].

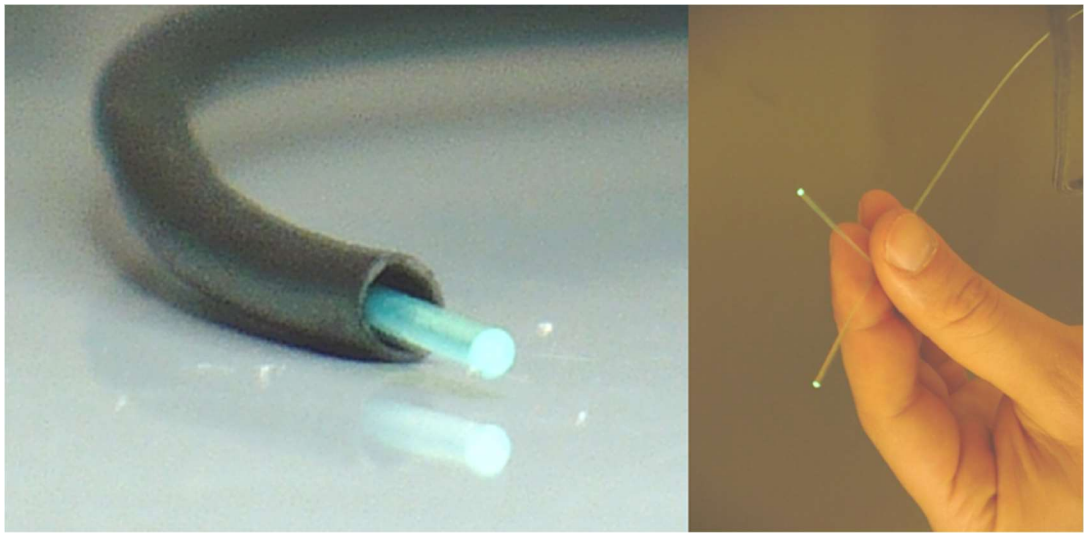

FIG. 2. A sample of 1 mm diameter plastic scintillating fiber.

FIG. 3. A gamma detector sample, with scintillating fiber and the two SiPMs installed inside a thin aluminum pipe. The same detector can be hosted into stiff or flexible plastic pipes.

The plastic scintillating fibers have the advantage of coming in a predefined shape that is also suitable for immediate geometric modifications: one can cut them to the desired length and arrange them around the source at will, even flexing or looping them. One might argue that a 1 mm thick plastic scintillator is very poorly sensitive to the gamma radiation, which is basically the ubiquitous radiation coming out of the radwaste. However, even though a gamma ray deposits a small amount of energy in the fiber, the small number of scintillation photons thus produced can still be detected by the single-photon-sensitive SiPM detector. The expected gamma rays from radwaste have energy below 2 MeV, typically well represented by ^{60}Co (1.17 and 1.33 MeV) and ^{137}Cs (0.66 MeV) radioisotopes. We performed a series of numerical simulations using the GEANT4 Monte Carlo tool, finding that at these gamma energies the average energy deposited on the 1 mm fiber, calculated for events depositing at least 50 keV, is around 180 keV thus producing about 1500 scintillation photons. Under these conditions the expected average number of photoelectrons produced on each SiPM is a few tens, that are easily detected and discriminated from the intrinsic dark noise of the photosensor. The corresponding intrinsic efficiency with the same 50 keV threshold, i.e. the probability that a gamma ray crossing the fiber deposits at least 50 keV, is of the order of

≈0.5%, a small but useful value that allows to operate at high activity without saturating the detector. A representative bench test, whose results are plotted in Fig. 4 was performed with radioactive sources. The test showed the expected behavior for the detectors, with a good S/N ratio and background below 1 count/s. A test with twelve detectors around four low-activity drums, performed in a real repository, showed very promising results: even though the activity was very low, the difference in counting rates with and without drums was of the order of 50 standard deviations (Fig. 5).

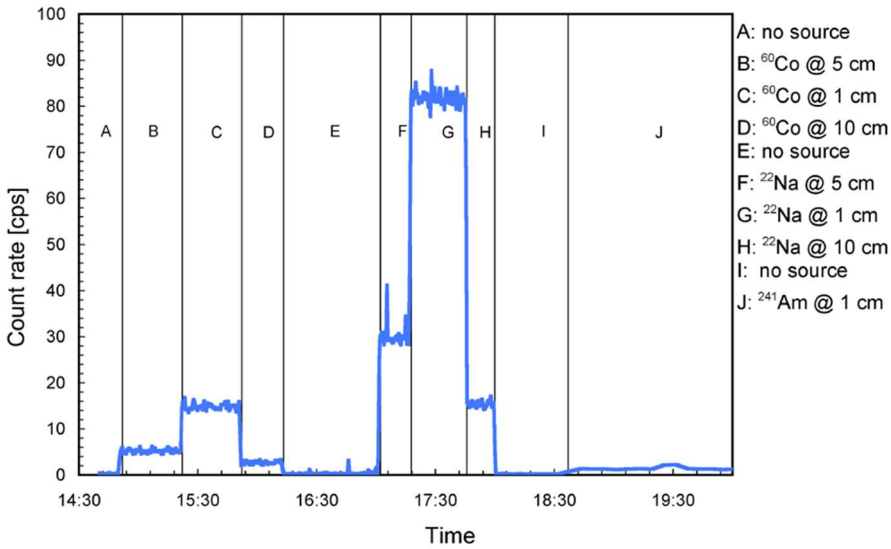

FIG. 4. Bench test of a sample detector with radioactive sources.

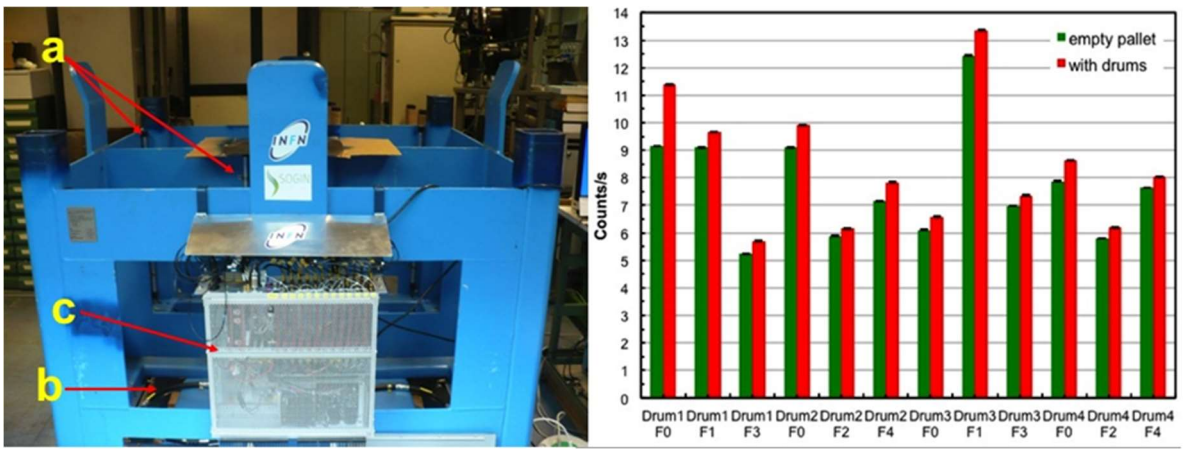

FIG. 5. Left: prototype of a 4-drum platform equipped with 8 linear detectors (a), four circular detectors (b), and the support electronics (c). Right: summary of a one-day test with and without radwaste drums. The difference in counting rates was of the order of 50 standard deviations.

3. DETECTION OF THERMAL NEUTRONS

For the detection of thermal or low energy neutrons, as a viable alternative to ^{3}He tubes, semiconductor-based detectors can be used in combination with a neutron reactive film, usually called neutron converter, which converts neutrons into charged particles [8, 9]. The main characteristics of a converter, suitable for use in combination with a solid state detector,

are large neutron absorption cross section, small gamma ray emission, and presence of decay channels into energetic charged particles. The two nuclear species commonly considered as suitable neutron converters are ^{6}Li (natural abundance ≈7%, thermal neutron cross section ≈940 b) and ^{10}B (natural abundance ≈20%, thermal neutron cross section ≈3840 b). The corresponding neutron capture reactions involved are:

$$^{6}Li + n \rightarrow {}^{3}H\ (2.73\ \ MeV) + \alpha\ (2.05\ \ MeV)$$

and

$$^{10}B + n \rightarrow {}^{7}Li\ (1.01\ \ MeV) + \alpha\ (1.78\ \ MeV)\ \text{ground state, BR} \approx 6\%$$

$$^{10}B + n \rightarrow {}^{7}Li\ (0.84\ \ MeV) + \alpha\ (1.47\ \ MeV) + \gamma\ (0.48\ \ MeV)\ \text{excited state, BR} \approx 94\%$$

We decided to choose 6Li because after capturing a neutron it has a unique decay channel, with no gamma rays, with a higher available kinetic energy and with lighter particles produced, therefore easier to detect than in the case of ^{10}B. We use it in form of LiF, enriched at 95% in ^{6}Li, that is a very stable and nonexpensive salt which we can deposit uniformly as very thin layers onto rather large substrates up to hundreds of cm^{2}. Based on these assumptions we have shown that inexpensive solid state neutron detectors with good n/gamma discrimination can be built. We have chosen to use independent detector and converter, which allow a better modularity and easy reconfigurability [10].

The produced charged particles, namely ^{3}H and ^{4}He, lose some of their energy passing through the converter; the energy loss depends on the converter material itself, the particle type and its energy, and therefore the film thickness must be carefully chosen. Moreover, the reaction products entering the detector must have sufficient energy in order to produce signals large enough for particle discrimination from the gamma background.

The dependence of the spectrum shape on the converter thickness has been reported in several papers [11, 12] where it is evident that a thinner film ensures a better separation between the reaction products and the background, although the detection efficiency becomes smaller.

In [13] we sketched the conversion mechanism of a neutron inside the ^{6}LiF layer. As the two reaction products are emitted in opposite directions, only one of them may enter the detector. The case shown makes use of a double-sided silicon pad detector, 3 cm x 3 cm, 300 µm thick, with a neutron converter on each face. For a bench test we placed the detector of [14] inside a neutron moderator made from polyethylene blocks, along with an AmBe neutron source. Such a source, besides producing neutrons with a broad spectrum centered around 4-5 MeV, produces also as many high energy gamma rays (≈ 4.4 MeV), and a huge amount of ≈59 KeV low energy gamma rays. We performed such a test, under different configurations, in order to disentangle the detector behavior. In [15] we superimposed three typical spectra obtained with the detector under the following conditions: no neutron source (area 1), converters installed; with neutron source but no converter (area 2); with neutron source, converters installed (area 3). What immediately sticks out is that the spectrum without source, only due to environmental background gamma (and cosmic) rays, lays entirely below about 0.3 MeV. When exposing the detector without neutron converters to the neutron source, the contribution to the spectrum comes from low energy gamma rays and from protons in (n,p) reactions on silicon [16]. In this case we can see that such a contribution lays below ≈ 1 MeV. Conversely, when installing the two converters and exposing the sandwich to the neutron source, we see a totally different spectrum shape, due to the detection of alpha and triton particles, coming from the neutron capture on ^{6}Li followed by the decay of ^{7}Li into alpha and triton. The alpha contribution in the spectrum is between 0.6 MeV and 1.7 MeV, whereas the triton

contribution is between 1.7 MeV and 2.7 MeV, exactly as expected from calculations and simulations, and both represent a perfect signature of neutron detection.

A similar technique can be fruitfully applied to scintillators, and due to the usually lower cost of scintillators one can decide to deposit the converter layer directly on the scintillator itself, or to make use of an additional substrate. For our first prototypes we have been using strips of CsI(Tl) or Pilot-U scintillators, either with direct deposition of LiF or by wrapping them with an aluminum foil onto which the converter was deposited. A patent application for this family of new neutron detectors was filed in 2013 [17]. These detectors can be fruitfully exploited for permanent real-time online monitoring of nuclear spent fuel casks, thus they can represent a viable tool to ensure the continuity of knowledge after sealing the casks, that is the main ingredient both for safety and security.

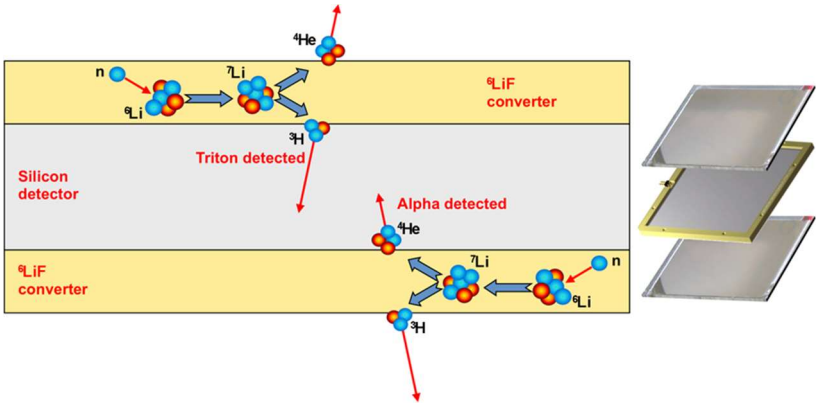

FIG. 6. Sketch of a thermal neutron detector consisting of a double-sided silicon detector coupled with two ^{6}LiF neutron converters. As the neutron energy is very low, the reaction products are emitted in opposite directions in the laboratory frame, and only one is detected.

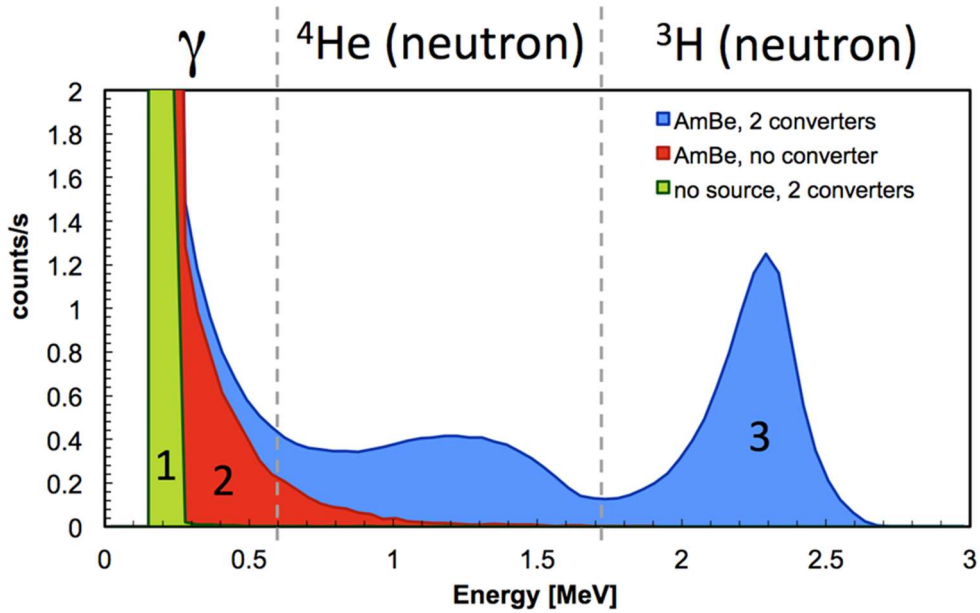

FIG. 7. Three typical spectra obtained with the detector under the following conditions: no neutron source, converters installed (area 1); with neutron source but no converter (area 2); with neutron source, converters installed (area 3). The gamma contribution to the spectrum is roughly below 0.6 MeV; the alpha contribution between 0.6 MeV and 1.7 MeV; the triton contribution between 1.7 MeV and 2.7 MeV.

4. THE DMNR SYSTEM

In view of implementing a mesh of radiation detectors to be placed around radwaste drums inside repositories for online monitoring purposes, we plan to employ the gamma ray sensor, based on a scintillating fiber, and the solid state technique for thermal neutron detection. The tests showed very encouraging performance and features, which can be summarized as follows:

- Possible deployment as a mesh of detectors;
- Robustness & flexibility;
- Low intrinsic efficiency, which ensures a good dynamical range;
- Reasonable geometrical efficiency;
- Reliability, because of the intrinsic simplicity of the devices;
- Possible position sensitivity;
- Ease of handling;
- Reasonable radiation hardness: tests have shown that after several Gy of high energy electrons and gammas the gamma detector is still working, even though its resolution degrades. The same can be said for the solid state thermal neutron detectors;
- Last, but not least, the described detectors are rather inexpensive.

The front-end electronics for the prototypes was entirely developed at INFN-LNS, and we are now establishing links with private companies for the engineering of more compact solutions to be commercialized. We also developed the computing infrastructure, which allows on the one hand to exploit a powerful online console containing many interactive tools for real time automatic monitoring and control, and on the other hand to interrogate the relational database where the full history of the repository is recorded. It is worth to mention a useful feature of the console, namely the possibility of a virtual navigation among the drums (see [18]), which are highlighted in green, yellow or red according to the measured activity, and which can be selected with a simple click in order to obtain detailed data about them. The online data and the database can be easily made available for web access to duly authorized people (or to the general public?).

5. CONCLUSION AND PERSPECTIVES

In the near future the nuclear decommissioning will become a quite relevant activity, as most of the 435 active nuclear reactors all over the world (as of 2013) are older than 30 and 40 years. This implies that new repositories will have to be put in operation, and the general public will require more transparency and more detailed data, also considering that the official quantity of the current worldwide radwaste is 30 million m^3 [19]. The implementation of a modern monitoring architecture could represent a relevant step in the direction of a better acceptability of the repositories for the population, as well as an additional and transparent tool both for the control authorities and for the general public.

ACKNOWLEDGEMENTS

The fiber gamma detector was initially developed with the support of INFN and Ansaldo Nucleare. After a few years of development and test we are now going to instal a prototype monitoring system inside a real radwaste repository located in a former nuclear power plant in Sessa Aurunca (CE, Italy), in collaboration with the nuclear decommissioning company Sogin. The development of more sophisticated thermal neutron detector units is being pursued in collaboration with people from the Joint Research Center in Ispra and from DG ENER Euratom in Luxemburg.

We are grateful to Carmelo Marchetta from INFN-LNS staff for his efforts in producing the neutron converters. We are indebted with Annafrancesca Mariani, Severino Alfieri, Mario Iorio and Luca Savino from Sogin for their support in the test at the radwaste repository. Last but not least, we owe a lot to our colleague and friend Marco Ripani, responsible of the INFN-Energy strategic project, for his continuous support.

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A NOVEL APPROACH FOR MONITORING HIGHLY ACTIVE WASTES

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Abstract

The Fukushima-Daiichi nuclear accident is currently one of the largest challenges to the nuclear industry. Due to damaged infrastructure many of the reactor cooling systems were taken off-line and consequently fuel loaded into each reactor has possibly melted or indeed become relocated or damaged due to the original earth quake. At present the location and condition of the highly active (HA) waste fuel is unknown and understanding this is a significant barrier to decommissioning the stricken reactors. Createc Ltd. have developed a novel gamma camera imaging system, called N-Visage that utilises a slot collimator which, when compared to traditional pin-hole camera systems, exhibits a significantly higher SnR whilst maintaining a lower overall collimator mass. As a result N-Visage can maintain high-resolution imaging whilst remaining in a small and light overall package – often a significant advantage during remote deployments. This system was designed to monitor medium activity infrastructure with an upper limit of 1 Gy/Hr, already far beyond the range of most other gamma camera systems and has already been extensively deployed within Fukushima-Daiichi and has had far-reaching success. At present the limit to the imaging system used within N-Visage is the detection mechanism used within the collimator, where a very sensitive Cadmium Zinc Telluride (CZT) detector 'whites-out' at a comparatively low fluence-rate. Traditionally the way to solve this is to increase the mass of the collimator to restrict both the solid angle of the aperture and the overall background to reduce the overall fluence-rate. This however nullifies significant advantages of the N-Visage system and would make the deployment of the system within the Primary Containment Vessel (PCV) awkward, if not impossible. In this paper we present a feasibility study that investigates the possibility of extending the working envelope of the N-Visage detection mechanism toward 10 000 Gy/Hr; the estimated upper-bound of the gamma environment within the Reactor Pressure Vessel (RPV) of the stricken reactors. The experimental results have shown that by using either a variant of highly pure silicon or a $CdWO_4$ scintillator and by limiting the active electronics within the deployed system, it is technically achievable to deploy an N-Visage system within the reactor. Work is currently underway to design a highly radiation tolerant variant of the N-Visage gamma imaging system, that once deployed within the reactors will help to ascertain the location and condition of the HA fuel waste and remove this fundamental barrier toward decommissioning.

1. INTRODUCTION

Gamma imaging is being increasingly deployed in the civil nuclear industry for radiological monitoring and surveillance and it is now widely accepted as a reliable method for providing location tracking of radioactive sources and partial radiological characterisation. Different types of gamma cameras vary in regard to their detecting, collimating and readout principles and their levels of success may vary [1, 2, 3]. That is not only because different configurations affect key characteristics such as field of view and sensitivity; the physical size, weight, scan times and radiation hardness are equally important factors that can determine the fitness of a system for a certain purpose. Tokyo Electric Power Company (TEPCO), the utility company responsible for the Fukushima-Daiichi plant, has for example recently attempted to deploy a standard 150 kg gamma imaging system within a PCV of one of the stricken BWR reactors but have had little success. Interestingly the determining factor for abandoning this attempt was the excess weight of the gamma camera [4].

The N-visage gamma camera is a fast-imaging, compact and lightweight 17 kg system that can rotate 360 degrees and has already been successfully deployed within Medium Activity (MA) areas at several locations within the Fukushima-Daiichi reactor site. The results of one such case study are shown in the figure below.

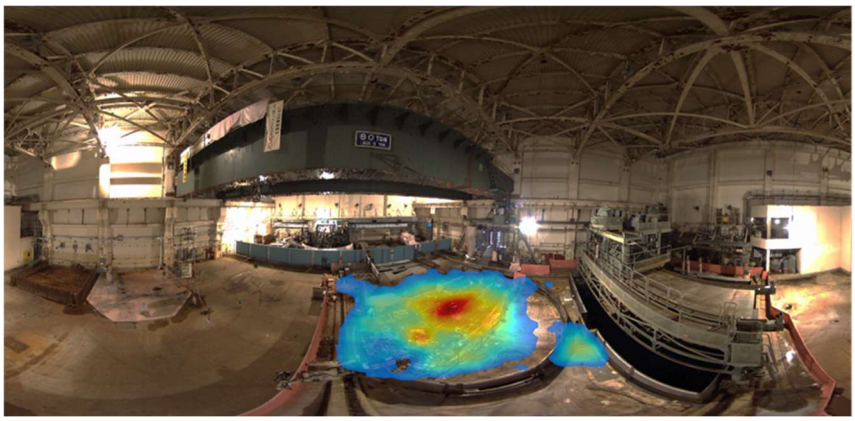

FIG. 1. Using a gamma camera to locate the source of radiation at Fukushima Daiichi. The coloured area indicates the location of the most intense radiation sources in the scene.

Despite its success, N-Visage has not been previously used at extreme environments like the ones present in the PCV, mostly due to the limitations of the current sensitive CZT detector and associated electronics. TEPCO have estimated that the dose rate apparent within the PCV can typically reach scales of around 10 000 Gy/hr and in order for N-Visage to be utilised in such radiation-hard environments its radiation tolerance would have to be significantly increased.

In this report we present a feasibility study regarding the development of a gamma imaging device to enable Highly Active (HA) inspection. More specifically, we focus on the selection of highly dose-tolerant and mechanically robust gamma radiation detectors that would be compatible with the N-Visage imaging process. Further development of the N-Visage imaging system into a highly radiation tolerant variant that can be deployed within the PCV will enable the environment to be fully imaged, helping to locate the fuel debris and ascertain the condition it is currently in. This would help TEPCO plan for safe fuel removal of the stricken reactors a very significant barrier to the current decommissioning process being undertaken.

2. THE N-VISAGE IMAGING METHOD

A direct approach to single detector imaging utilises a cylindrical single-aperture collimator that restricts the exposure of the spectrometer or dosimeter to a small angle. The image is then constructed pixel by pixel where each pixel corresponds to a distinct measurement. The limited field of view results in monitoring a considerable amount of background radiation due to gamma-ray leakage of the collimator. A typical approach to remedy this effect is to use a high attenuation collimator, which in turn implies greater amounts of material and an inevitable increase in collimator mass. The increased effectiveness of the collimator however, ultimately results in a weakening of the signal of interest, usually compensated by increasing the system and collimator slit physical dimensions. This increases the collimator's solid angle, the absolute efficiency and the strength of the signal at the expense of resolution.

The N-Visage Imaging Method achieves a higher signal in the detector and a lower collimator mass by replacing the standard cylindrical collimator shape with a slot [5]. In this scheme, the solid angle is determined by both the width and the length of the slot whereas the resolution only depends on the width. It is therefore possible to increase the solid angle, thus achieving a

better signal to background ratio, without reducing the resolution or increasing the collimator mass.

For the purpose of illustrating the N-Visage imaging process, we describe the simplest and hence sub-optimal physical realisation of the imaging system which consists of a spherical collimator with a hollow centre containing a radiation detector, and a slot cut halfway through the sphere to form an aperture. An image is formed after the slot-collimator is rotated at different poses as shown in Fig. 2. Apart from a small maximum dimension this configuration has the advantage of measuring the radiation incident from every angle and subsequently analysing the full radiation dose. In this particular implementation the collimator diameter is less than 100 mm and it enables the entire gamma camera to fit in a 110 mm cylindrical package.

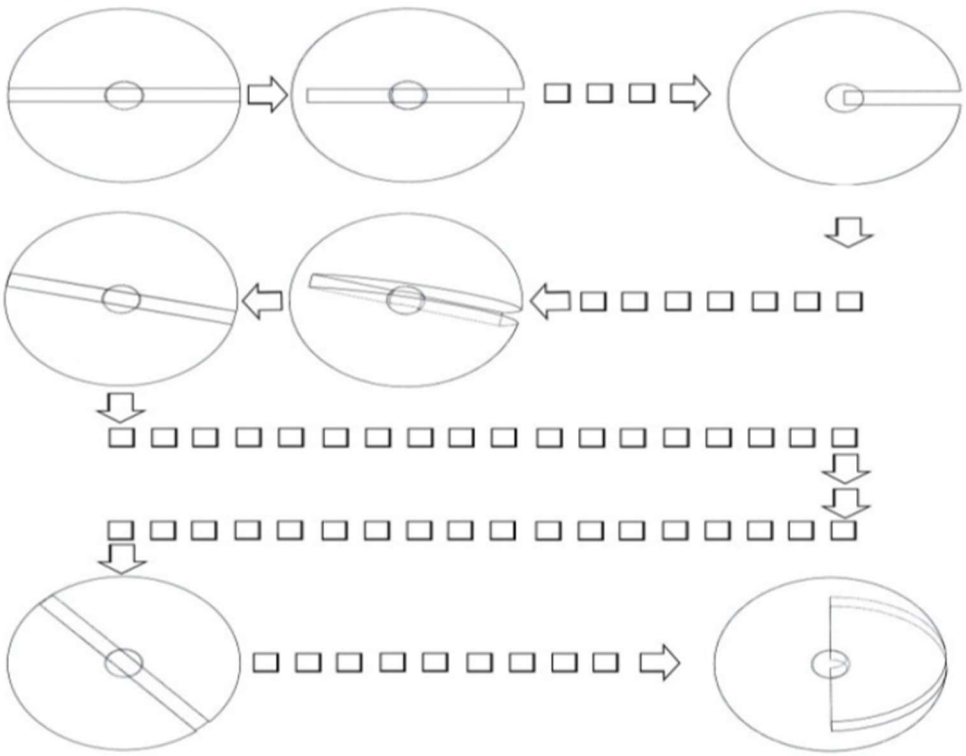

FIG. 2. Example slot-collimator scan pattern. Squares indicate the presence of intermediate poses that have not been drawn. Poses are ordered so that the step between consecutive poses requires only a small rotation.

The traditional cylindrical collimator approach builds the resulting image pixel by pixel, where each pixel corresponds to a distinct measurement. The slot collimator however, employs an indirect imaging approach which forms the final image using standardised reconstruction techniques with a non-negativity constraint; done to ensure a realistic radiometric solution. For each pose, a spectrum is recorded and converted to a single value by summing counts around characteristic photopeaks. Each reading d_i is then expressed as a weighted sum of each pixel s_j, i.e $d_i=A_{ij}s_j$ and the reconstruction process is used to solve the linear inverse problem

$$\boldsymbol{d} = A\boldsymbol{s}. \tag{1}$$

The entries of the systems matrix A encode the attenuation effect of the collimator for a specific pose with respect to radiation arriving from a particular. Iterative solving schemes that can incorporate a positivity constraint such as the Algebraic Reconstruction Technique, or the Projected Landweber method can then be used to estimate the distribution of incident gamma photons relative to the camera from such an underdetermined system of equations. This then allows the distribution to be placed on a 360x180^{o} 2D panoramic image of the environment and create the final gamma camera image.

3. RADIATION SENSOR SELECTION

Indirect measurement, moving collimator gamma-ray imagers such as N-Visage have an intrinsic advantage in that they can more or less use any point radiation measurement device as their sensor, as long as a suitable collimator can be engineered and manufactured it will work. More importantly for high-radiation imaging this type of device does not require the development of sophisticated pixelated detectors or the use of complex readout electronics which would require significant protection from the high-dose radiation. Due to the intense gamma field any device selected would need to be capable of running in 'current-mode' where no energy discrimination is undertaken due predominately to the large quantity of pulse pile-up.

From literature, three general classes of detector were deemed appropriate for the application; scintillators, semiconductors and gas ionisation detectors. In general however gas ionisation chambers tend to be more fragile than the other two types and given the likely deployment mechanisms to be used on-site, this type of detector was not given serious consideration. The other two identified types of detector have distinct strengths and weaknesses: scintillation type devices are fundamentally optical rather than electrical and are therefore immune to electrical noise, which can be a serious limiting factor when making sensitive measurements given the electrically harsh environment of most nuclear plants. Semiconductors generate only weak electrical signals which can be easily drowned out, especially over the long signal cable runs required to isolate the detector electronics from the high levels of radiation; whereas the scintillator would remain fairly impervious.

The strength of semiconductors is that the long cable runs remain significantly more unaffected to high-dose uptake when compared to optical fibre. When fibre-optic cable is irradiated significantly it tends to darken, adding non-linearity to the response of the detector at the read-out electronics as the fibre starts to attenuate the signal. Therefore using fibre it will be hard to maintain a given detector calibration for more than a few minutes within the active environment. There are two methods of reducing this effect: utilise 'radiation-hard' fibre which will extend the life-span of the cable but will not remove it entirely, or minimise the total exposure of each device by replacing it in-situ at a given total dose and therefore special consideration must be made within the final device to allow this to occur. These methods however, would not be required for the semiconductor detection mechanism.

From this initial literature-based search an extensive yet focused search of suitable detectors was performed combining both academic text and manufacturer catalogues/capability with a view to testing devices within an available irradiation chamber. The selection of the devices is detailed below and the testing of the realised detectors is detailed in section 4.

3.1. Selection metrics

During the search for a suitable detector there are six fundamental attributes that were deemed important to the overall gamma imaging system;

Sensitivity — The smallest signal change that the detector can measure.

Sensitivity Range — The range of signal that the detector can measure, in this case the range of gamma-ray photon kinetic energies that the detector can measure.

Required Support Electronics — The amount and complexity of the electronics required to retrieve the signal from the detector. As the deployment environment will be particularly harsh, any electronics may either need to be shielded or alternatively removed from the deployed sensor altogether and placed in an external control box away from the harsh environment.

Thermal and mechanical robustness — As the detector is to be deployed remotely and in a harsh industrial setting, it is required that any solution is robust enough to withstand the inevitable strains placed upon it during deployment.

Radiation Hardness — As the dose-rate of the environment is so high, it is likely that any detector will become damaged to some degree due to the ionizing radiation. Any detector that is chosen must be 'radiation hard' meaning that it will continue to function when both the instantaneous and accumulated dose become large.

A full list of the considered devices is not appropriate for the purpose of this report, however when compiling the list, two suitable types of device have emerged:

High purity semiconductor — Cimpoca [6] has shown that these devices can be manufactured to withstand significant instantaneous and accumulated dose. The major problem with these type of detectors for this type of implementation is that a pre-amplifier is traditionally required close to the detector – not ideal given the environmental conditions.

Micro-scintillator — Seo [7] and Hyung [8] have shown that certain kinds of small scintillating crystals like BGO and $CdWO_4$ can withstand very high doses. These crystals are coupled to a standard fibre optic cable meaning that no electrical signal is required within the harsh deployment environment and therefore no electrical noise will be induced into the cable.

Two detectors were chosen to test; one scintillation based and the other semiconductor.

3.2. Description of $CdWO_4$ scintillator

The selected scintillation material was Cadmium Tungstate ($CdWO_4$). This material has a high density (7.9 g/cm^3) enabling small detectors to be relatively efficient - an important factor when designing the moving collimator. The light yield from $CdWO_4$ is approximately 40% that of NaI(Tl) with an emission spectrum peaked at 470 nm, the material is also non-hydroscopic and relatively mechanically robust making it suitable for deployment within harsh environments.

The $CdWO_4$ material was to be implemented into a 'micro-scintillator' detector; these devices have a small cross-section resulting in a relatively low efficiency for a given dose-rate, well suited to the high gamma environment estimated within the PCV. The crystal is then coupled to a long fibre cable which would enable the associated read-out electronics to be placed outside the environment, away from any damage effects. With a few very specific exceptions, this type of detector implementation is not available off-the-shelf therefore a custom design was implemented. This device utilised a 5 mm diameter × 10 mm length

$CdWO_4$ crystal encased into a simple cylindrical aluminium can which was subsequently bonded to a 3 mm diameter, 10 m length PMMA fibre. The completed device used for radiation hardness testing is shown in Fig. 3.

FIG. 3. The final bespoke $CdWO_4$ detector. It comprises of a 5 mm diameter by 10 mm crystal encased into an aluminium tube. This was then coupled to a 10 m length of 3 mm diameter PMMA fibre.

The major drawback to using $CdWO_4$ is its relatively long light decay time when excited by gamma energy; where a mixture of two differing components are apparent, one at 1.1 μS (40%) and the other at 14.5 μS (60%). This afterglow effect, where light continues to be emitted for some time after the initial event, could be a significant issue as it will limit the response time of the material to varying radiation fields. Thus, after glow will limit the speed that the spinning collimator can move and thus the length of time taken to resolve a given gamma image – a value which at the extreme may exceed the lifetime of the detector within the PCV environment. In later sections we detail a simple test of the $CdWO_4$ detector that estimates the speed of response using a simplistic slot collimator.

3.3. Description of the high purity silicon diode

High purity silicon diode detectors for high dose applications are well established and are generally available off-the-shelf in suitable packages and are thus relatively cheap. The device selected was a radiation hardened Si-007D Si diode detector from Eurorad, which was placed inside a TO39 package and connected to the read-out scheme by a 10 m length of RG59 coax cable. The detector is shown in Fig. 4.

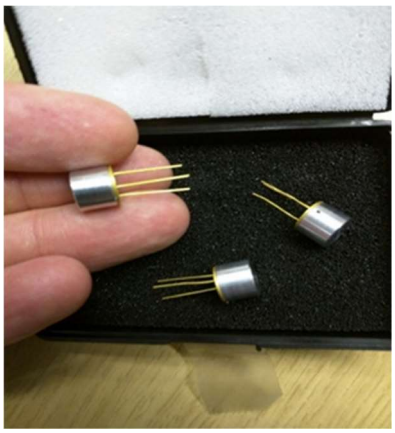

FIG. 4. The Si-007D Si Diode Detector was mounted inside a standard TO39 package and was connected to the read-out scheme by a 10 m length of RG59 coax cable.

As the radiation intensity within the PCV is high the detector can be run in a photovoltaic mode of operation meaning that the resultant current can be read directly from the diode without the need for either applying a reverse bias voltage or pre-amplifier. However the drawback to using this type of configuration is that as the contact potential at the P-N junction is low, typically less than 1 V, the current from the device is very small, typically in the order of pA. Therefore there is a significant potential for this signal to be drowned out by noise and also niche read-out electronics are required.

4. RADIATION HARDNESS TESTING

Once selected, the detectors were tested within a hostile high-dose radiological environment that mimicked the environment within the reactor PCV as closely as possible. For this, an Irradiation chamber at the Dalton Institute, Westlakes Cumbria was used, which uses a series of ^{60}Co sources located within 3 tubes which sum to an activity of 535 TBq. The chamber is shown in Fig. 5 and by adding lead collimation and positioning samples within the chamber, the apparent dose-rate can be varied from approximately 20 Gy/hr to 21 000 Gy/hr.

FIG. 5. The Irradiation Chamber at the Dalton Institute – The rear tubes are used to expose ^{60}Co sources during an irradiation. Maximum dose rate available is 22 000 Gy/hr, when a specimen is placed at the front of the chamber.

Two key metrics were evaluated using the chamber:

Maximum dose-rate operability — The Detectors have a maximum dose-rate that they can sustain and remain functionally useful. Ideally this will be when the device output (either light or current, dependent upon device) becomes saturated and no longer increases when the dose-

rate is increased or alternatively when the output becomes non-linear i.e. when the output decreased as dose-rate increases.

Total dose exposure/tolerance – The high dose-rate apparent means that the devices will unavoidably become damaged by radiation at some point and no-longer function as required. It is important to ascertain the maximum dose that the devices can withstand and remain approximately within a given calibration.

To evaluate these metrics a simple experiment was performed which varied the dose-rate within the irradiation chamber at steps from its minimum to its maximum dose-rate values. By recording the resultant output of each detector a simple first-order response function can be evaluated. If each device is subsequently given an additional dose uptake and the simple experiment subsequently repeated then the response of the detector can be compared and an idea of performance degradation can be obtained. Ideally, this response function experiment was to be performed at 0, 0.5 and 1 MGy of total dose uptake for each of the detectors which represent the detector being placed at the most hostile location of the PCV for a period of 100 hours.

4.1. Silicon Diode results

The results obtained from the Silicon Diode detector are shown in Fig. 6. The data shows that the Silicon Diode is linear from 20 Gy/hr up to 21 000 Gy/hr, which is the upper limit of the Irradiatior, with the detector very close to the ^{60}Co sources. Also, the data is repeatable as shown by very similar results from the two runs. This data has proven that the Silicon Diode detector mechanism is capable of working within the specifications for the imaging project and is a feasible option to use within the final device.

The results for total dose uptake tolerance for the Silicon Diode detector are shown Figure 6. This data shows that the detector can function within the hostile environment for significant periods of time with only a ~30% drop in output due to radiation damage. This makes the radiation damage effect negligible over the course of a gamma scan (in the order of minutes) meaning that the detector could be used for a full survey campaign. This would remove the need to change the detector out in-situ, reducing the complexity of the final realised device making it easier to use and potentially making it cheaper to make. Finally, the radiation damage within an actively deployed diode could be simply calibrated away by measuring accumulated dose and adjusting the expected response. If it is found that an active measurement is required however it is thought that this could be achieved by simply measuring either the voltage drop or dark current apparent over the diode, a technique which is relatively well known in literature.

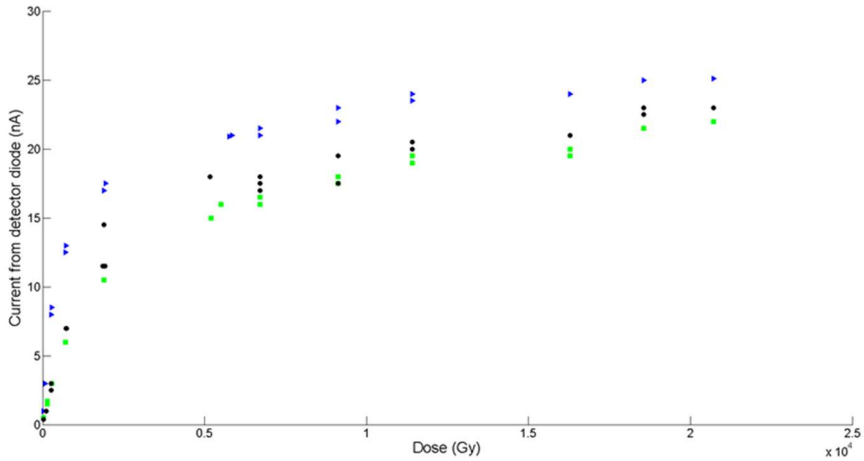

FIG. 6. Results from Silicon Diode detector at 0 Gy (Blue), 0.5 MGy (Black) and 1 MGy total dose. Drop in Data at approximately 16 000 Gy/hr is due to external influences and was consistently low through-out all experimentation.

4.2. $CdWO_4$ results

The results from the $CdWO_4$ crystal are shown in Fig. 7. As can be seen the detector exhibits an almost perfectly linear response to dose-rate from approximately 300 up to 21 000 Gy/hr as well as exhibiting good repeatability. It would therefore be feasible to use this detector in the final imaging system. The response of the detector to dose-rates below 300 Gy/hr were not measureable however, this is likely due to a lack of light collection efficiency within the read-out scheme used to conduct this study. Further work is currently being undertaken to optimise this read-out scheme and is expected to lower the detection limit of the detector to below 100 Gy/hr.

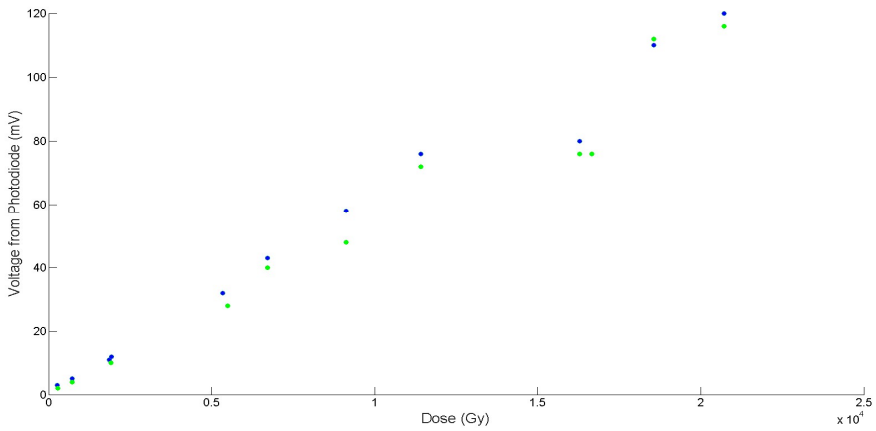

FIG.7. Results from $CdWO_4$ detector at 0 MGy total dose. Blue 'x' are values from a first 'run' with increasing dose-field whilst green 'o' are from a second run with decreasing dose field. Drop in Data at approximately 16 000 Gy/hr is due to external influences and was consistently low through-out all experimentation.

The $CdWO_4$ crystal was subsequently exposed to 0.5 MGy total dose. However during the exposure the connecting fibre snapped meaning that no light was apparent when the second test was performed, therefore no further information could be obtained using the $CdWO_4$

crystal. The fibre was subsequently stripped and inspected, the condition of which is shown in Fig. 8. It is likely that this issue would not arise over the course of a given deployment however further research is required to understand this fully.

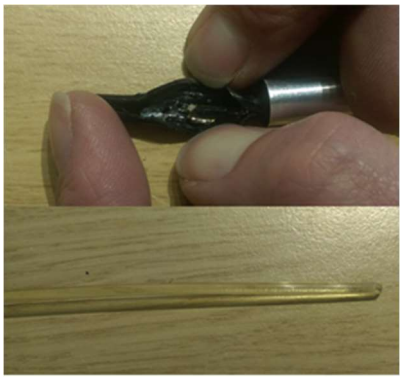

FIG. 8. The $CdWO_4$ fibre optic cable having been stripped of the outer sheathing. As can be seen the cable has shattered due to embrittlement caused by radiation damage effects. Also notice the small amount of darkening within the fibre.

4.3. $CdWO_4$ detector response time

A very crude estimate of the response of the $CdWO_4$ detector was required to understand if the afterglow of the scintillation mechanism mentioned would cause a significant problem. This was done by utilising a very simple spinning vertical slit collimator implemented within a lead filled aluminium canister. The ideal response from this make-shift collimator was then estimated by simply rotating it every 45° and sampling the detector read-out. The result from this is shown Fig. 9.

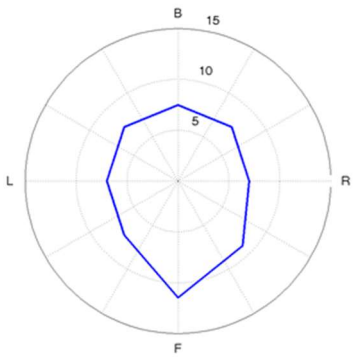

FIG. 9. Polar plots of collimator response for $CdWO_4$ Scintillation detector (mV). F – Front (w.r.t ^{60}Co source), B – Back, L – Left and R – Right.

The collimator was then rotated at 1 Hz, an estimate of the scanning speed of the final gamma camera device and the read-out electronics recorded to estimate the detector speed of response.

FIG. 10. Response of the $CdWO_4$ scintillator read-out scheme when the collimator was travelling quickly. The voltage division is set to 5 mV and the time base is 250 ms. Note Triangular waveform showing collimator response function.

Fig. 10 shows the result from this simple experiment. Given the resolved shape of the collimator response, it is likely that 1 Hz is not the limit for the response and therefore it is likely that the afterglow effect apparent within the $CdWO_4$ crystal will not be a significant issue if it is selected for use in the realised device.

5. CONCLUSION

The feasibility study detailed in this report have demonstrated that it is indeed possible to extend the operating envelope of the N-Visage imaging technique up to the estimated 10 000 Gy/hr required for deployment within the PCV. This result has identified two potential solutions to the detector sensitivity problem currently limiting the upper dose-rate that the N-Visage system can be deployed in and thus removes a significant barrier to using it within the stricken Fukushima-Daiichi reactors. Current research effort is focused on the engineering challenge of producing a variant that meets the challenge within the extremely harsh environment found within the PCV. This research is focused on optimising key areas such as the detector, collimator and system geometry interface as well as the associated system read-out schemes. Other research is focused on minimising the active electronics required within the system, especially within any part of the system likely to be deployed within the PCV done to ensure the device remains as robust as possible. Also, remote deployment arrangements as well as radiation hard environment laser rangefinder measuring devices are being evaluated to ensure that a full system evaluation is performed. Significant progress has been made with these topics and our initial results are very encouraging; this information will be disseminated in subsequent reports as more data becomes available and the final design of the system progresses.

ACKNOWLEDGEMENTS

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