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Ageing Management Programmes for Spent Fuel Dry Storage Systems

Final Report of a Coordinated Research Project

AGEING MANAGEMENT PROGRAMMES FOR SPENT FUEL DRY STORAGE SYSTEMS

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AGEING MANAGEMENT PROGRAMMES FOR SPENT FUEL DRY STORAGE SYSTEMS

REPORT OF A COORDINATED RESEARCH PROJECT

INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

At the end of 2024, there were 416 nuclear power plants in operation. More than 400 000 tonnes heavy metal of spent fuel has been discharged from nuclear power plant operation to date, with about 300 000 tonnes heavy metal of that total currently stored in various storage facilities using wet and dry storage systems. Approximately 106 000 tonnes heavy metal of this inventory is in dry storage, and in recent years the share of dry stored fuel has increased relative to wet storage.

With the start of construction of the world's first geological disposal facility for spent fuel in Finland, a clear pathway to disposal is now available. Although similar projects in France and Sweden are also at an advanced stage, most IAEA Member States are still decades away from constructing such facilities, resulting in prolonged storage durations for spent fuel until disposal becomes possible. For storage systems to operate safely over long periods of time, potentially beyond their original design lifetime, ageing management programmes are vital, and these programmes need to be specifically designed for the storage systems to which they are applied.

To help address this need, the IAEA initiated the Coordinated Research Project on Ageing Management Programmes for Spent Fuel Dry Storage Systems in 2015. The project aimed to gather national experience in developing and applying methodologies for the implementation of ageing management programmes specifically for dry storage systems.

This publication is based on the results of the coordinated research efforts carried out by participating Member States during the project period from October 2016 to October 2021.

The IAEA is grateful to the project participants for their contributions to the drafting and review of this publication. The IAEA officer responsible for this publication was C. Gastl of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Nuclear power is utilized in many countries as a low carbon baseload power source for electricity. A key challenge related to nuclear power is the overall fuel cycle, namely, long term management of the spent nuclear fuel (SNF) after it has been removed from the reactor. Storage of SNF is a vital fuel cycle step in every country utilizing nuclear power generation or running nuclear research reactors. According to the Power Reactor Information System managed by IAEA, the number of operational reactors is 416 (as of Dec. 2024), with 211 permanently shut down. Management of spent fuel varies by country; in some countries, spent fuel is transported to centralized storage facilities for storage and/or reprocessing, while in others, spent fuel is stored on site in either spent fuel pools (SFPs) or dry storage systems (DSSs). According to Fig. 1 approximately 430 000 t of SNF has already been produced worldwide, and while about one–third has been reprocessed, the remaining is stored.

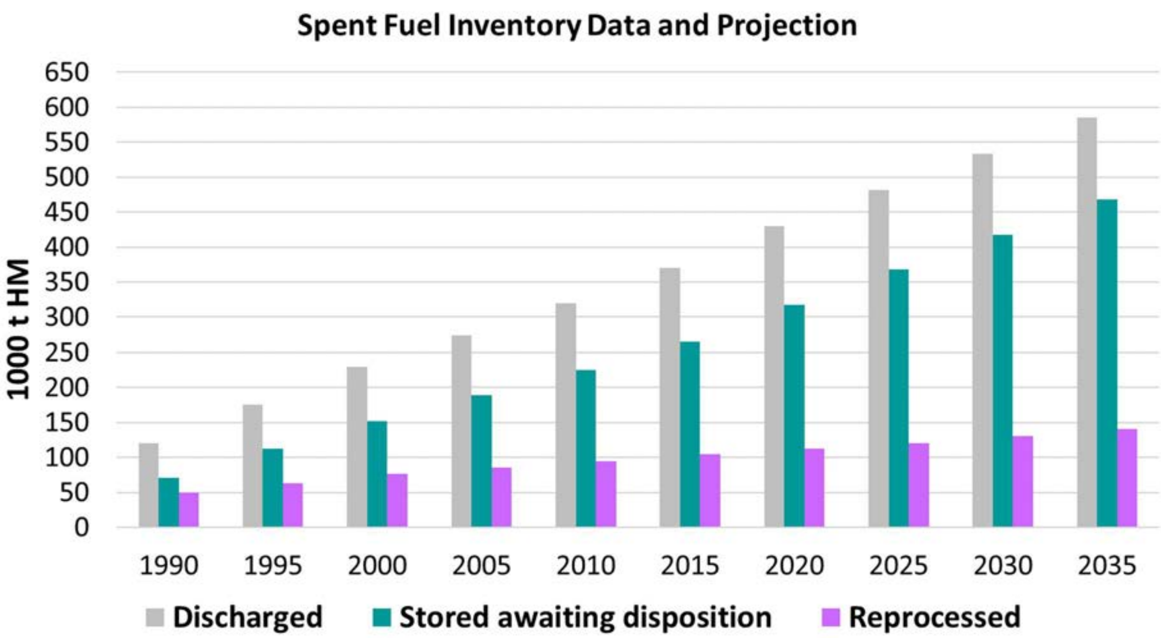


FIG. 1. Spent Fuel Arisings. Accumulation of spent fuel in storage vs spent fuel reprocessed.

According to the IAEA Safety Glossary, the storage of spent fuel is defined as “the holding of spent fuel in a facility that provides for its containment, with the intention of retrieval” [1].

Several storage systems have been developed and deployed that fulfil all necessary safety requirements.

According to the data from the 7th review reports submitted by contracting members of the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management (Fig. 2), the majority (65%) of stored fuel is kept in wet storage. The dry storage of SNF (accounting for around 35% of the inventory) is taking place either in storage facilities on the reactor site or in dedicated off-site facilities.

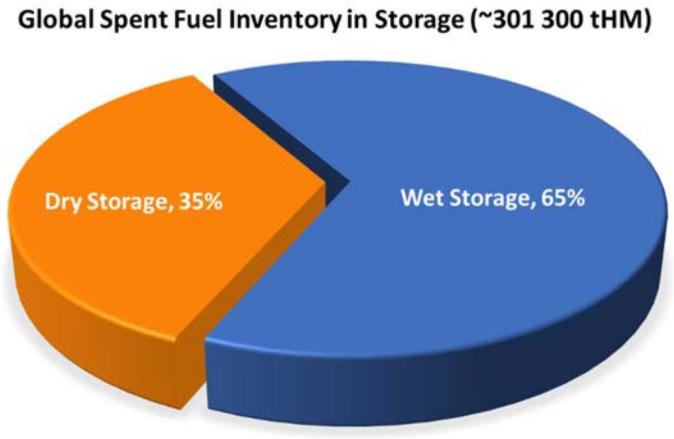


FIG. 2 Distribution of SNF among different types of storage locations in 2019.

Eventually, the ultimate disposal of fuel from a once through fuel cycle or high level waste after one or more recycling cycles, will consist of transportation to and emplacement in a deep geological repository (DGR). Several countries around the world have begun the process to design, license, and construct a DGR, however, in others predicted dates (initially) for the availability of DGRs, have passed, e.g., in the year 2012 in Germany [2], and surface storage has had to be continued.

Owing to the delayed commissioning and operation of these DGRs, utilities will need to operate their storage facilities for longer durations than originally planned. Wet storage has been used for several decades, and ageing management of these facilities is generally well understood. In the meantime, many utilities have opted for DSSs, largely because of their passively safe design, to manage SNF in the interim until disposal facilities become licensed and operational. Many DSSs already have extended licenses for continued operation, and it is anticipated that more DSSs will be required for continued SNF storage for periods significantly longer than their original licensed periods (typically 20–40 years), and that they may be needed for 120 years or longer [3]. For the extended operations of nuclear power plants (NPPs) far beyond their originally licensed lifetimes, ageing management programmes (AMPs) have already been shown to be a key tool for safely extending operational duration. Therefore, the expectation is that in the case of DSSs, AMPs will also be a key enabler to verify continued safe operation [4] at these facilities to cover the time needed in each country to develop and implement its disposition pathway. Since in some countries dual purpose casks (DPCs) licensed for both storage and transportation are used for storage of SNF, the IAEA Transport Safety Standard Committee (TRANSC) has started a working group to discuss AMPs for transport packages (see also Section 2.3). Furthermore, in the current IAEA Specific Safety Requirements SSR-6 rev. 1 Regulations for the Safe Transport of Radioactive Material [5], AMPs are applied to transport packages also.

1.2. OBJECTIVE

This publication reports the findings of Coordinated Research Project T21028 Ageing Management Programmes for Dry Storage Systems (AMP–DSS). These findings will support MS in implementing their specific AMPs. As it also contains AMP examples from different countries already implemented and experience with it, it is also a good resource for operators searching for reference cases.

1.3. SCOPE

This publication is limited to dry storage systems and their AMPs. It captures national approaches and the usage of results from AMPs. A systematic approach on how to generate an AMP depending on the specific dry storage system is presented and some examples of operational experience are presented. Technical options for monitoring, mitigation and repair were gathered and summarized.

1.4. STRUCTURE

This report is organised in nine main sections and conclusions are provided in Section 10.

Section 2 provides a brief introduction describing the currently available dry storage systems. Section 3 consists of an overview of national approaches towards ageing management, a description of the utilisation of ageing management results in licenses extension and information regarding transport after storage. Section 4 describes the stepwise methodology of generating an AMP and provides examples for selected dry storage systems. Section 5 details the necessary classification of SSCs and describes those for the different types of dry storage systems. Section 6 provides information about some specific ageing mechanisms for different materials used in dry storage systems. Section 7 highlights operational experience provided by organizations participating in the CRP. Section 8 shows the currently available technologies for monitoring, mitigation and repair. Section 9 is focused on the need to update existing AMPs and ensure the results of any conducted inspections are considered. The Appendices cover national approaches to ageing management, a concept for post storage inspection and research contract report summaries.

2. SPENT FUEL DRY STORAGE SYSTEMS

Various dry storage systems for commercial SNF from power reactors are available, each having specific features and advantages as described in the latest IAEA publication [6] providing an overview about spent fuel storage options and systems. There are different factors that affect the choice of technology, including the type of fuel to be stored and the expected quantity.

To distinguish among the different technologies depicted in Fig. 3 [6], the first step is to differentiate between casks and storage structures. Casks are able to obtain a type B(U) package design approval for transportation according to IAEA SSR–6 Rev. 1 [5], while storage structures are not intended to be transported. For casks, two different types are currently used based on the cask body material either metal or concrete with a metal liner. Both metal and concrete cask types can store SNF either in canisters (e.g., HI–STAR 100) or as bare fuel when equipped with a basket (e.g., CASTOR V19, ENUN 32P, HI–STAR 80, HITZ 52B, TN 24E, TUK–137T.A). For storage structures, the first differentiator is the equipment and complexity of the structure. A storage building (typically referred to as a vault), along with additional equipment like inbuilt cranes, may be used to store SNF either in canisters (e.g., Fort St. Vrain, United States of America (USA), and Mining Chemical Complex, Russian Federation) or as bare fuel (e.g., Paks, Hungary). The other type of storage structure is a rather simple unit, with SNF held in canisters. Storage units are available in either vertical or horizontal (e.g., NUHOMS) orientation. The vertical storage units are deployed as either single units or multiple units. The single storage units can be ventilated (e.g., HI–STORM, MAGNASTOR, VSC–24) or not ventilated (e.g., AECL Concrete Silo, INVAP). Multiple storage units are differentiated into above ground types (e.g., MACSTOR) and below ground types (e.g., HI–STORM UMAX).

The examples displayed here do not represent any recommendation or preference of the IAEA but have been chosen for demonstration purposes only. Each Member State has to pick the storage system(s) that best serves its needs with respect to the anticipated disposal policy and associated timescale.

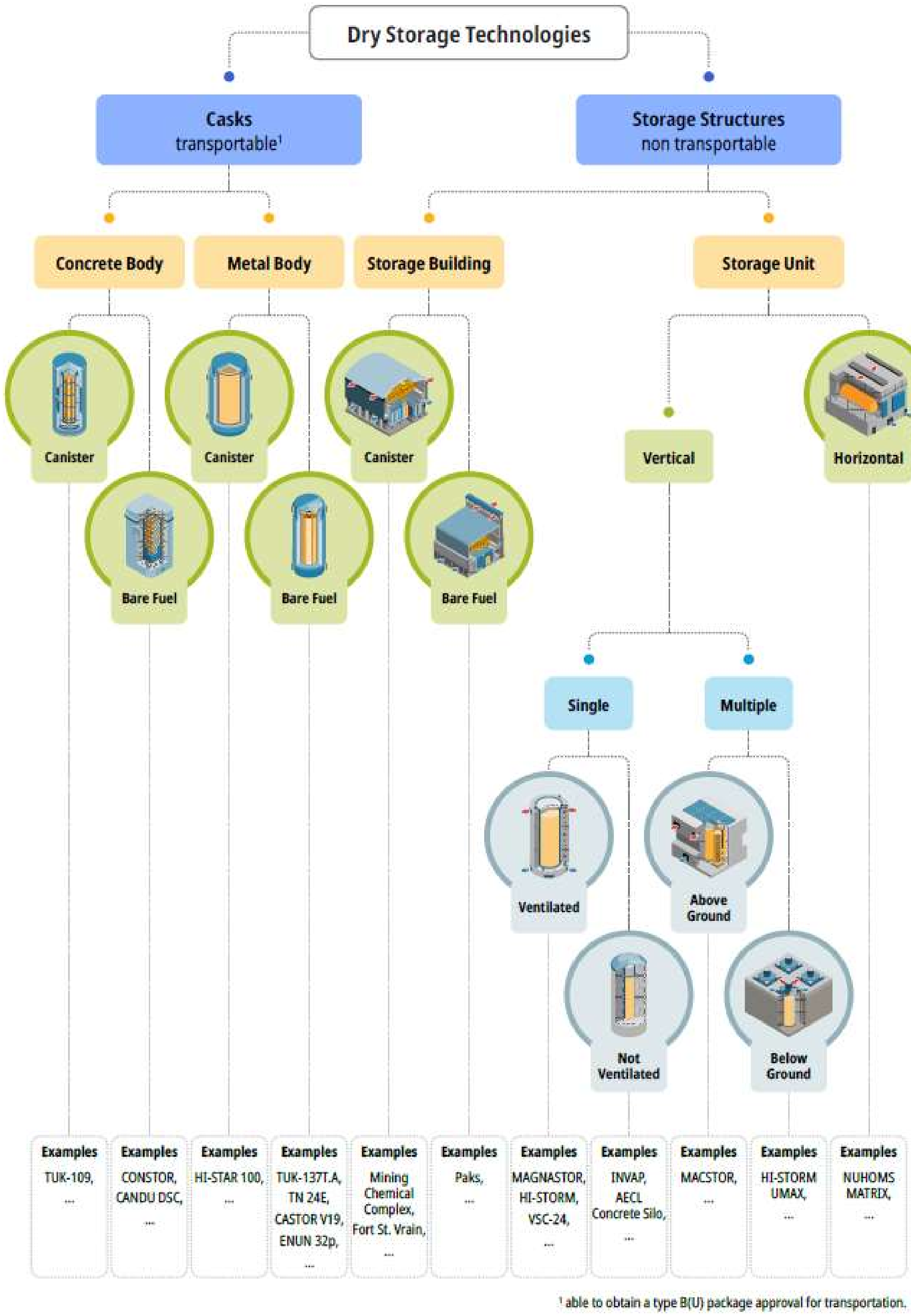


FIG. 3 Dry storage technologies [6].

From the diversity of examples provided in Fig. 3, it can be assumed that different technologies based on different materials and storage environments would require separate and specific AMPs. For this reason, this publication will focus on approaches for developing AMPs for all current dry storage technologies.

The distribution of SNF across DSSs already used to store SNF is depicted in Fig. 4 (based on data given in 2019 Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management 7th contract reports) and shows the need for development of approaches for all current systems.

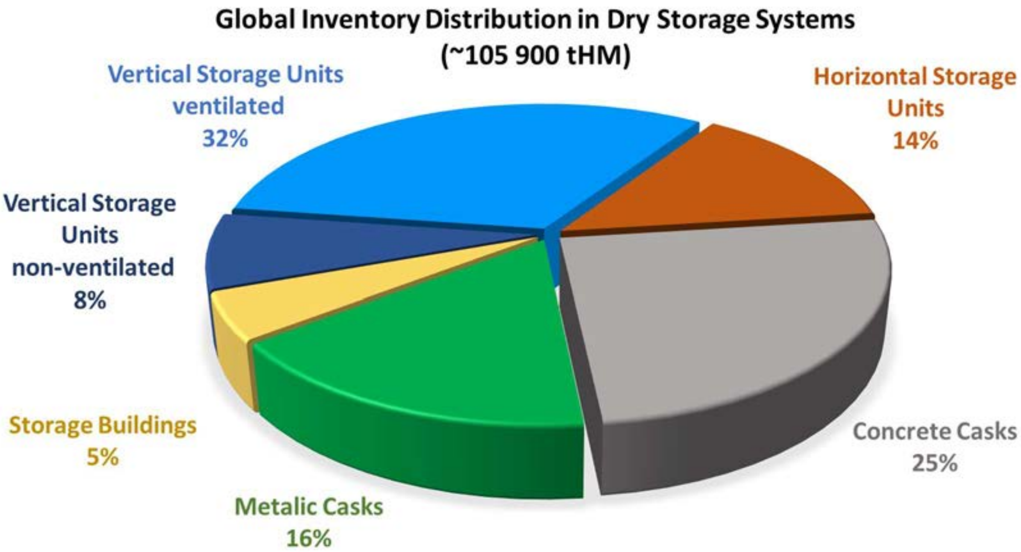


FIG. 4 Current distribution of SNF among different dry storage systems [6].

2.1. METAL OR CONCRETE CASKS WITH BOLTED LIDS

Bolted cask systems are typically an all in one design with the cask providing shielding, confinement, criticality protection, heat removal, and retrievability. The primary components for ageing management in a bolted cask system include bolts, seals, trunnions, and the outer surface coating as illustrated in Fig. 5.

The cask body or shell is typically made of carbon steel or ductile cast iron, and polymer materials or concrete for neutron absorption; these all have different ageing mechanisms and different non–destructive examination (NDE) requirements. Pitting/localized corrosion, degradation of seals and bolting, and accelerated general corrosion are typical ageing mechanisms of concern in the case of metallic casks. All casks have to be manufactured in accordance with the quality assurance (QA) system of the manufacturer. Additionally, every individual cask is controlled by an independent authorized body or regulator. After manufacturing, a certificate of conformity with the approved type is issued and a copy is submitted to the regulatory body. All deviations from the approved type have to be recorded, assessed and approved. This step ensures that the starting condition of the cask is defined.

FIG. 5 Metal cask scheme for dry storage of spent fuel assemblies (left); TN–32B example (right) (courtesy of Orano NPS).

When DPCs, licensed for both transport and storage, are used, the impact of ageing on transport packages has to be considered especially when transport after storage is intended [5]. Casks are stored in a dedicated cask storage facility. An indoor or outdoor concrete pad where individual metallic or concrete casks are placed, with monitoring systems and physical protection including, in the case of indoor facilities, a concrete building. These facilities usually do not provide any safety functions. However, indoor storage facilities contribute to the long term safety of stored casks by providing weather protection and may provide additional shielding.

2.2. STORAGE UNITS

A storage unit is typically a two part system, with the welded canister providing certain design functions (structural, confinement, criticality safety, etc.) while the concrete overpack provides shielding and additionally may support seismic functions. The overpack also contains air inlet opening and outlet vents, and associated pathways for decay heat dissipation if needed. A horizontal storage unit (HSU) and a vertical storage unit (VSC) are shown in Fig. 6. The primary locations for ageing management of a welded canister system are the locations in and around welds. Areas of interest also include internal support structures, standoffs/channels, coatings, and concrete.

FIG. 6 Examples of storage units comprising horizontal (upper) and vertical dry cask storage systems (lower). (Upper image courtesy of Arizona Public Service/Palo Verde. Lower image courtesy of Xcel Energy).

DSS manufacturers typically have several models that employ various design differences based on site specific needs or advances in materials, processes, or technology. Currently, most canisters are manufactured from 304 or 316 stainless steel, while cask based systems are manufactured from a combination of neutron absorber resin or polyethylene and carbon steel or cast iron. Both systems tend to utilize aluminium boron based neutron absorber materials because of their high thermal conductivity.

2.3. STORAGE BUILDINGS

Storage buildings (see Fig. 7 as an example), are reinforced concrete buildings (above or below ground) containing an array of storage cavities to contain one or more SNF assemblies or other high level radioactive waste. Shielding is provided by the exterior (concrete) structure. Heat removal is normally accomplished by forced or natural convection of air over the exterior of the fuel containing units or storage cavities, with the air subsequently exhausted directly to the outside atmosphere or the heat dissipated via a secondary heat removal system. Examples: Paks (Hungary), HABOG (Netherlands), or MCC (Russian Federation).

FIG. 7 Mining Chemical Complex RBMK storage facility (Courtesy of Tenex).

3. NATIONAL REGULATORY APPROACHES TO AGEING MANAGEMENT OF DRY STORAGE SYSTEMS

Extended abstracts on national approaches to ageing management guidance were provided by Argentina, Czech Republic, Germany, Japan, Pakistan, Spain, and the USA in December 2018. A summary of these extended abstracts, with additional questions and comments, was discussed during the second Research Consultancy Meeting (RCM–2) at Argonne National Laboratory, April 29—May 3, 2019. The complete final abstracts on national approaches to ageing management guidance can be found in Appendix I. Important topics covered there are as described in 3.1 to 3.3.

3.1. OVERVIEW OF APPROACHES TOWARDS THE DEVELOPMENT OF AGEING MANAGEMENT GUIDANCE

The national approaches to ageing management guidance vary depending on the country's fuel cycle policy (reprocessing or disposal); technology (wet or dry storage), and subsequent transportation via road, rail, or waterways. Choices of dry storage locations (at reactor or away from reactor) and storage environment (indoor or outdoor) also vary, as reflected in each country's approach to ageing management guidance. The need to store spent fuel in DSSs longer than initially intended is recognized; therefore, effective AMPs are required to reduce risk, ensure public safety and health, and protect the environment.

3.2. EXTENDING BEYOND ORIGINAL LICENSING BASIS (DESIGN INTENT)

Using accumulated data, the integrity of dry metallic casks and spent fuel for a period of up to 60 years has been evaluated in Japan [7]. In addition, a concept of alternative means of visual inspections for post storage transportation without opening the cask lids has been described;

see Annex II for a full description. More details on how to update an existing AMP are addressed in Section 9, Update of AMPs.

3.3. TRANSPORTATION AFTER STORAGE

In the case of canisterized systems, the transportation overpack is designed according to the transport regulations in effect at the time. DPCs may be used for SNF storage [8]. This storage method needs to comply with the transport safety regulations in effect at the time of license renewal during storage or transport, perhaps decades in the future [9].

The Package Design Safety Report (PDSR) has to clearly state how the design meets all safety requirements for transport and the regulatory requirements for an ageing management system. The revision to IAEA SSR–6 [5] introduced the following requirements to be applied to shipment after storage:

- Paragraph 106, "Transport comprises all operations and conditions associated with, and involved in, the movement of radioactive material; these include the design, manufacture, maintenance and repair of packaging, and the preparation, consigning, loading, carriage including in transit storage, shipment after storage, unloading and receipt at the final destination of loads of radioactive material and packages" [5];
- Paragraph 503 (e), "For packages intended to be used for shipment after storage, it has to be ensured that all packaging components and radioactive contents have been maintained during storage in a manner such that all the requirements specified in the relevant provisions of these Regulations and in the applicable certificates of approval have been fulfilled" [5];
- Paragraph 613A, "The design of the package has to take into account ageing mechanisms" [5];
- Paragraph 809 (f), "If the package is to be used for shipment after storage, a justification of considerations to ageing mechanisms in the safety analysis and within the proposed operating and maintenance instructions" [5];
- Paragraph 809 (k), "For packages which are to be used for shipment after storage, a gap analysis programme describing a systematic procedure for a periodic evaluation of changes of regulations, changes in technical knowledge and changes of the state of the package design during storage" [5].

Concerning paragraph 613A, for shipment after storage, consideration of ageing mechanisms is important because of the long period between loading and the start of shipment after storage. The consideration of the impact of ageing on the package needs to be supported by AMPs and time limited ageing analyses (TLAAs). The AMPs address ageing effects by including prevention, mitigation, condition monitoring, and performance monitoring for effective ageing management.

Concerning paragraph 809 (f), justification of considerations of ageing mechanisms, and paragraph 809 (k), the gap analysis is a periodic assessment of whether the package design complies with the current transport regulations. The frequency of the assessment depends on the national regulations and can range from five years for a cask design currently in production to several decades for casks that are no longer in production and are, during this period, only used for storage. This gap analysis considers changes in the regulations, changes in technical knowledge, and changes of the state of the package due to ageing during storage, resulting in the identification of gaps and research actions to fill the gaps.

A transport package design approval is normally issued for a period of several years. At the end of the approval period, the operator of a storage facility (user of the DPC) will need to make sure the package design owner applies for an extension to the package design approval. It is important that an identified, suitably qualified and experienced organisation retains the design authority status for the DPC, because technical questions and the need for further technical justification of the design may be requested by the competent authority as part of the package approval certification process. Renewal of the package design approval certification for the next period will require demonstration of compliance with the transport safety requirements of the transport regulations extant at the time of the certification process.

4. METHODOLOGY FOR GENERATING AN AGEING MANAGEMENT PROGRAMME

The following Section outlines a methodology to be used in generating an ageing management programme (AMP).

4.1. OVERVIEW OF THE STEPS NECESSARY TO GENERATE AN AGEING MANAGEMENT PROGRAMME

To generate an AMP for a DSS, certain steps are performed to ensure that the AMP includes all activities necessary to maintain the intended functions of the storage SSCs. Figure 8 shows the step by step approach, including the scoping evaluation (outlined in Section 5), ageing management review (AMR) (Section 4.3), and the development of the AMP (Section 4.4). In some cases, a TLAA (Section 4.5) may be an alternative approach to demonstrate that an ageing effect will not detrimentally affect an intended safety function. A periodic update of the AMP is undertaken as described in Section 9.

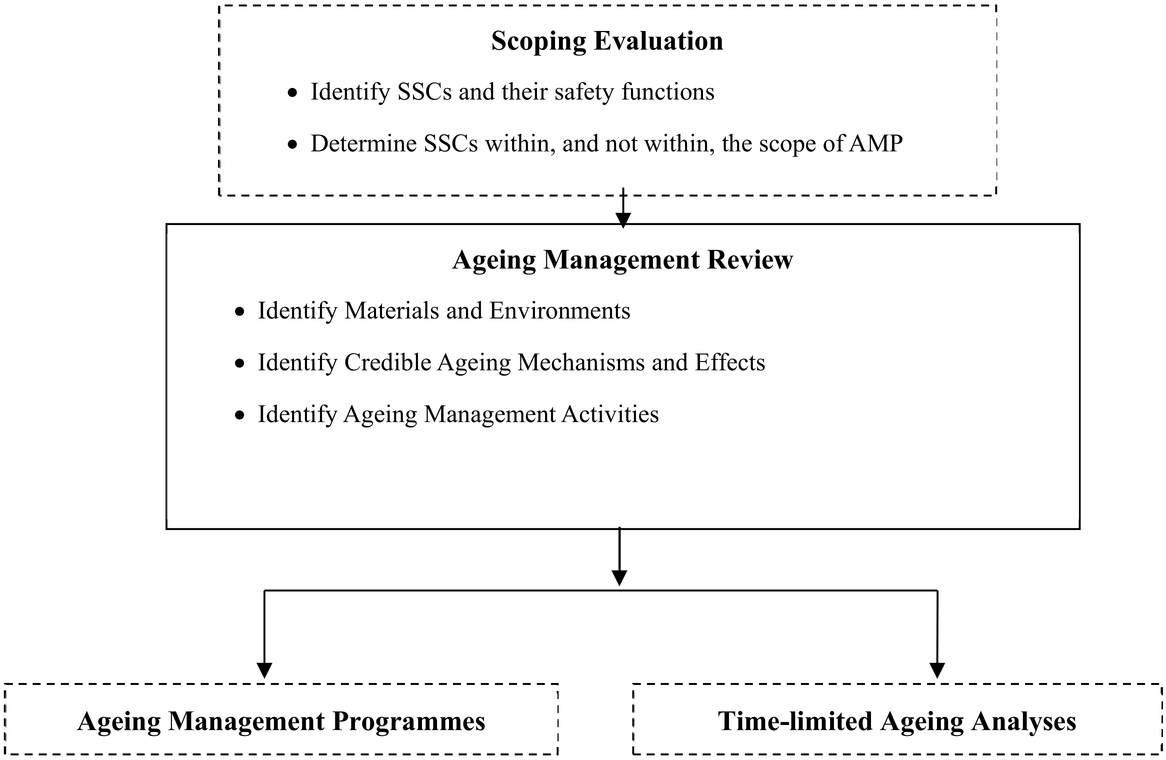


FIG. 8. Stepwise approach to generating an AMP.

4.2. AGEING MANAGEMENT REVIEW

An AMR is conducted to assess all in scope SSCs to determine whether there are ageing effects that could adversely affect the SSCs' ability to perform their intended safety functions. If so, activities are identified to manage those ageing effects. The AMR is implemented using established procedures and methods in accordance with national regulatory requirements (see also the description of the national approaches in Annex I). An AMR approach consists of the following steps:

- Identify materials and the service/operating environment (e.g., humidity, temperature, salt content) for each in scope SSC, referring to the information that formed the basis of the SSC's safety assessment and licensing;
- Identify credible ageing effects (those that could affect a safety function if not addressed by an ageing management activity). Section 6 provides the technical underpinning for ageing effects. In addition, extensive guidance on ageing effects has been published by MS [2];
- Identify AMPs for managing ageing effects. Section 4.2 provides the guidance for developing an AMP that effectively monitors and controls the credible ageing effects.

Depending on national requirements, some ageing effects may not require an AMP if the effects can be shown, by analysis, to be incapable of adversely affecting a safety function for the operating period. Such analyses are usually TLAAs, i.e., existing design basis analyses with a time dependent component. Some examples of such analyses are provided in Section 4.4.

4.3. AGEING MANAGEMENT PROGRAMMES

In accordance with the results of the AMR, an AMP specific to dry storage SSCs can be developed. It monitors and controls degradation of SSCs identified in the scoping evaluation so that ageing effects will not result in a loss of intended safety functions. The AMP includes inspection and/or monitoring activities to detect ageing effects, and acceptance criteria indicating when corrective actions are needed to manage ageing effects, so that the DSS and the storage facility continue to perform their safety functions. Section 4.3 provides example AMPs.

4.3.1. Ageing management programme elements

An AMP has nine elements as outlined in the following subsections.

4.3.1.1. Scope of the ageing management plan, based on understanding of ageing

The scope of an effective programme clearly defines the SSCs and their intended functions that are the subject of the AMP. These SSCs are those that were identified in the AMR as potentially being adversely affected by ageing.

4.3.1.2. Preventive actions to minimize and control ageing effects

Identification of preventive actions. This programme element describes in detail the activities for prevention of ageing. These activities are defined as those needed to prevent or minimize the initiation and progression of degradation:

- They typically include the use of physical barriers (e.g., coatings to prevent corrosion, or sealant to prevent water ingress) or operating practices (e.g., ground dewatering to control settlement of a foundation) that prevent or slow down potential degradation of an SSC;
- It is important to ensure that any proposed preventive action will not result in an unintended consequence for the ability of an SSC to fulfill its intended function(s);
- This programme element may not be relevant for condition or performance monitoring programmes that do not rely on preventive actions.

4.3.1.3. *Detection of ageing effects*

Identification of parameters to be monitored or inspected. This programme element identifies the ageing effects that the programme manages and provides a link between the parameter or parameters that will be monitored and how the monitoring of these parameters will ensure adequate ageing management.

In an effective condition monitoring programme, the parameter monitored or inspected will be capable of detecting the presence and extent of ageing effects. Some examples are detection and sizing of cracks and identification of the depth of corrosion degradation.

In an effective performance monitoring programme, a link will be established between the degradation of the particular SSC intended function(s) and the parameter(s) being monitored. An example is the measurement of radiation to detect the degradation of radiation shielding materials.

Effective technology (inspection, testing and monitoring methods) for detecting ageing effects before failure of the SSC:

- Ageing effects need to be detected before there is a loss of the SSC intended function(s). This programme element clearly states when, where, and how programme data are collected (i.e., all aspects of activities to collect data as part of the programme);
- For condition monitoring programmes, justification is provided, including codes and standards referenced, to show that the method or technique (such as visual, volumetric, or surface inspection) and frequency are sufficient to detect the ageing effects in advance of the loss of SSC intended function. A programme based exclusively on detecting SSC failures is not regarded as an effective AMP;
- Where sampling is used to represent a larger population of SSCs, the basis for the sample size is provided. The sample size is based on such aspects of the SSC as the specific ageing effect, location, existing technical information, SSC design, materials of construction, service environment, or previous fabrication, operating, or failure history. The samples are biased toward locations most susceptible to the specific ageing effect of concern. Provisions for expanding the sample size when degradation is detected in the initial sample are also included;
- In an effective performance monitoring programme, the 'detection of ageing effects' programme element discusses and establishes the monitoring methods that will be used for performance monitoring. In addition, the 'detection of ageing effects' programme element also establishes and justifies the frequency with which these performance monitoring activities will be implemented;

— In an effective prevention or mitigation programme, the 'detection of ageing effects' programme element discusses and establishes the monitoring methods that the programme will use to monitor the preventive or mitigative parameters that the programme controls; and justifies the frequency with which these monitoring activities are performed.

4.3.1.4. Monitoring and trending of ageing effects

Monitoring and trending activities are described, and they provide a prediction of the extent of degradation and thus enable timely corrective or mitigative actions.

A description is provided showing how the collected data are evaluated. The methodology for analyzing the monitoring or test results against the acceptance criteria is described. This programme element may also include trending for a forward look. Trending is a comparison of the current monitoring results with the original baseline (condition of SSCs at the beginning of storage). If this information is not available, previous monitoring results can be used to make predictions. The description includes an evaluation of the results against the acceptance criteria, and a prediction regarding the rate of degradation, to confirm that the next scheduled inspection or monitoring activity will occur before a loss of SSC intended function.

Although ageing indicators may be quantitative or qualitative, ageing indicators are quantified, to the extent possible, to allow trending. The parameter or indicator trended is described.

4.3.1.5. Mitigation of ageing effects

Activities that mitigate further degradation when degradation has been observed are described, including operations and maintenance actions.

Mitigating actions occur when the condition of the SSC is still within the bounds of the acceptance criteria. An example of a mitigating action is replacing a confinement boundary seal before it has reached the end of its useful life, or cleaning deposits from a canister surface.

4.3.1.6. Acceptance criteria

Acceptance criteria against which the need for corrective action is evaluated:

— The quantitative or qualitative acceptance criteria of the programme, and their basis, are described. The acceptance criteria, against which the need for corrective actions is evaluated, ensure that the SSC intended function(s) is consistently maintained under all design basis conditions. The programme includes a methodology for analyzing the results against applicable acceptance criteria;

— The acceptance criteria for an effective AMP could be specific numerical values or could consist of a discussion of the process for calculating specific numerical values to define conditional acceptance criteria to ensure that the SSC intended function(s) will be maintained under all design basis conditions. Information from available references may be cited;
— If qualitative acceptance criteria are used, the criteria are sufficiently clear to practicably ensure that a singular decision is derived on the basis of the observed condition, avoiding the use of ambiguous phrases (e.g., significant, moderate);
— It is not necessary to justify any acceptance criteria taken directly from the design basis information included in the final safety analysis report (FSAR), the storage facility technical specifications, or other codes and standards incorporated by reference into the applicable regulations; they are already part of the design basis. Nor is it necessary to justify those acceptance criteria that have been established in a methodology that is accepted or endorsed by the respective regulatory body, such as those that may be given in regulatory body approved or endorsed topical reports or endorsed codes and standards (see Sections 8.2.1 and 8.3).

4.3.1.7. Corrective actions

Corrective actions if an SSC fails to meet the acceptance criteria:

— Actions to be taken when the acceptance criteria are not met are described in appropriate detail or referenced in source documents. Corrective actions, including root cause determination and prevention of recurrence, are timely;
— If corrective actions permit analysis without repair or replacement, the analysis ensures that the SSC intended function(s) is maintained consistently with the design basis;
— For safety related SSCs, a QA programme confirms that the corrective actions are performed to address ageing effects and are in accordance with applicable code requirements or regulatory body approved standards. For example, in the United States of America, the implementation of a Corrective Action Programme approved under 10 CFR Part 72, Subpart G (Quality Assurance) [10] or 10 CFR Part 50, Appendix B (Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants) [11] ensures that corrective actions are performed to address ageing effects and maintain the design basis.

4.3.1.8. Operating experience feedback and feedback of research and development results

In the development of the AMP [12], consideration is given to site specific and industry operating experience, including research and development results (if applicable) from all MSs and international organizations relating to ageing management and performance of dry storage SSCs:

- Operating experience with existing programmes is discussed;
- The operating experience of existing AMPs or additional programmes, including past corrective actions resulting in programme enhancements, is considered;
- A past failure would not necessarily invalidate an AMP because the feedback from operating experience would have resulted in appropriate programme enhancements or new programmes. This information can show where an existing programme has succeeded and where (or if) it has failed in managing ageing degradation in a timely manner. This information would provide objective evidence to support the conclusion that the effects of ageing will be managed adequately so that the SSC intended function(s) will be maintained.

During the implementation of the AMP, the operator needs to review operating experience, as well as research and development results (if applicable), on a periodic basis to confirm the effectiveness of its AMPs or indicate a need to develop new AMPs. Section 9 provides additional information on how AMPs can evolve over time to respond to operating experience.

4.3.1.9. Quality management

Administrative controls document the implementation of the AMP and actions taken:

- The administrative controls of the programme are described. Administrative controls provide a formal review and approval process;
- For example, in the USA, an AMP summary is included in the licensee's FSAR. Any changes to AMPs are evaluated per the change control requirements in 10 CFR 72.48 [10].

Confirmation (verification) processes are followed to ensure that preventive and mitigating actions are adequate and appropriate, and that all corrective actions have been completed and are effective:

- When corrective actions are necessary, follow–up activities confirm that the corrective actions have been completed, a root cause was determined, and recurrence will be prevented;
- Record keeping practices to be followed are described.

4.4. EXAMPLES OF AGEING MANAGEMENT PROGRAMMES

Examples of AMPs are presented in references [13–15] and provide an acceptable generic approach to managing the credible ageing effects that were identified in the AMR.

The examples of AMPs may be used/credited by operators, but the regulatory body has to ensure that the operator demonstrates that the design features, environmental conditions, and operating experience for the subject facility or DSS are bounded by those evaluated in the given references. Otherwise, the regulatory body needs to ensure that the operator augments the AMPs as appropriate to address the impact of unique design or operating parameters.

The following existing examples of AMPs are applicable to storage facilities and DSSs and might be used to develop specific ones:

4.4.1. Monitoring of metallic surfaces

This programme manages the effects of ageing on the external surfaces of steel and stainless steel components that are directly exposed to outdoor air or concrete or are sheltered within overpacks. The scope of the programme includes metallic overpack exterior surfaces, dry storage canister support structures, access doors, vents, heat shields, embedments and anchorages, bolting, and other components important to safety. The AMP is a condition monitoring programme that consists of periodic visual inspections to monitor for general and localized corrosion, wear, cracking, and loss of pre–load (bolting). An additional AMP may be necessary to manage protective coatings that perform an important to safety function. Also, this AMP does not apply to stainless steel storage canisters, which are addressed in Section 4.4.4.

4.4.2. Bolted cask seal leakage monitoring

The integrity of the confinement boundary of bolted casks is monitored by a pressure switch or pressure monitoring system. This system helps to ensure that suitable corrective actions can be taken to maintain the safe storage conditions of the casks. The ageing effects include loss of material resulting from corrosion of the sealing surfaces, O rings, and bolts; loss of strength due to thermal ageing and change in dimension due to creep of the metallic O rings that results in loss of sealing forces; and loss of preload of the closure bolts. The specific components and systems that are considered to influence the pressure include the shield lid/primary lid, secondary lid, O ring assemblies, and associated bolts and welds. The programme relies on continuous pressure leakage monitoring systems to verify the integrity of the cask closure seals and the confinement boundary.

In addition, the programme may include visual inspections of normally inaccessible components under the cask protective cover to monitor for corrosion, coating degradation, loose bolts, and evidence of water intrusion.

4.4.3. Reinforced concrete structures

This programme consists of condition monitoring, performance monitoring, and mitigation and prevention activities. The programme includes recurring visual inspections by personnel trained to monitor reinforced concrete for applicable ageing effects, such as inspections described in standards endorsed by the MSs (e.g., ACI PRC-349.3-18 [16] in the USA). Identified ageing effects are evaluated using acceptance criteria derived from design bases or industry guides and standards. The programme also includes periodic sampling and testing of groundwater and the need to evaluate 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 (when applicable) and daily inspections to make sure the air convection vents are not blocked (when applicable, and per the requirements of the approved design bases). Also, online corrosion monitoring could be performed via corrosion probes embedded into the concrete [17]. These can measure several variables that are critical for reinforcement bar corrosion [18], such as oxygen content, temperature, cement resistivity, corrosion potential, and corrosion current, as in the example of Section 7.3.2.1. They can be embedded during construction or inserted afterwards and monitored in situ or remotely through wireless connections. More detailed information can be found in Section 7.4.

4.4.4. Localized corrosion and stress corrosion cracking of welded stainless steel dry storage canisters

This programme applies to welded stainless steel canisters used in canister based systems. The canisters are usually the primary confinement boundary during storage. This programme includes examination of external surfaces of welded stainless steel canister confinement boundaries for corrosive atmospheric deposits (e.g., chloride salts), localized corrosion, and stress corrosion cracking (SCC) [19]. Visual examinations are performed to identify corrosion products that may be indicators of SCC in the canister welds and weld heat affected zones (areas where there may be sufficient residual stresses to cause cracking). When localized corrosion is identified by visual inspection, follow-up volumetric examinations are used to characterize the severity of corrosion and determine if cracking is present. The dimensions of cracks are measured and used to inform corrective actions to ensure that the confinement function is maintained. Monitoring of confinement to detect helium leakage has been developed in Japan as described in Section 8.4.2.6.

4.4.5. Structural and functional integrity monitoring of canister/cask internals

The objective of the programme is to manage the ageing effects on the structural and functional integrity of the spent fuel assemblies and internals of storage canisters/casks to ensure spent fuel integrity, fuel retrievability even after extended long term storage, and subsequent transportation of the spent fuel for reprocessing or disposal [20].

The ageing effects managed include corrosion, creep, distortion, cracking, and peeling of coatings. The programme checks the ageing degradation of heat transfer, radiation shielding, criticality control, confinement boundary, and structural support functions of the storage canister/cask internals resulting from extended exposure to high temperature and radiation. The programme focuses specifically on high burnup fuel (HBF) assemblies (i.e. in the USA fuel assemblies with burnups exceeding 45 GWd/tU). The programme consists of: (a) a site specific assessment of the canister/cask designs to verify ongoing conformance with applicable standards and guidance documents; (b) comprehensive thermal and radiation analyses of the specific canister/cask designs used at the site to identify cask/canister locations that are likely to be most sensitive to changes in temperature and radiation because of postulated degradation of spent fuel cladding and storage basket/assembly or from helium leakage; and (c) development of a programme for monitoring radiation and temperature at such locations to determine data trends. Any significant deviation from the data trend is assessed for potential degradation of the functional and structural integrity of the canister/cask internals. Potential deviations in data trends because of helium leakage are managed by the programme activities under the AMPs for welded canisters and bolted casks.

4.5. TIME LIMITED AGEING ANALYSES

Depending on national requirements, some ageing effects may not require an AMP if the effects can be shown, by analysis, to be incapable of adversely affecting a safety function for the licensed operating period. Such analyses are usually a design basis safety analysis with a time dependent component, referred to as TLAAs. These TLAAs are calculations or analyses performed to demonstrate that SSCs within the scope of ageing management will perform their safety function(s) throughout the operating period.

TLAAs may be used to calculate fatigue life (number of cycles until predicted failure) or time limited life (operating time frame before expected loss of intended function, such as a corrosion rate calculation). TLAAs have to account for environmental effects that contribute to ageing effects. Examples of TLAAs include the following:

– Fatigue analyses of metallic structures and components to determine the number of loading cycles to failure;
– Boron depletion analysis of neutron poison plates to evaluate the potential loss of criticality control due to neutron irradiation;
– Corrosion rate calculations to determine if the loss of material from metallic surfaces could degrade structural and shielding functions;
– Analyses to evaluate whether the cumulative exposures to radiation and elevated temperatures during the operating period could change the mechanical properties of metallic or concrete materials and degrade their structural performance.

5. IDENTIFICATION AND CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS (SCOPING EVALUATION)

This Section provides information on identification and classification of the SSCs to be considered in the AMR for DSSs and Dry Storage Facilities (DSFs). This scoping evaluation is the first step in identifying relevant ageing mechanisms and effects that influence the safety of DSSs and DSFs.

5.1. CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

The main objective is to identify SSCs that are within the scope of the AMR and considered either ‘Important to Safety’ or ‘Not Important to Safety’. The in scope SSCs are reviewed in more detail to detect and describe the subcomponents that maintain the intended function or functions of the SSCs.

All SSCs that are ‘Important to Safety’, or those whose failure may prevent a function that is ‘Important to Safety’, have to be included in the AMR. References [21] and [22] can be used in determining the classification of DSS and DSF components according to their importance to safety.

5.1.1. Structures, systems and components classified as important to safety

According to Refs [1] and [23], the ‘Important to Safety’ SSCs (and associated subcomponents) are relied upon to perform one of the following functions:

– Maintain the conditions required by the regulations, specific license, or Certificates of Compliance (CoCs) to store spent fuel (and other high level radioactive waste) safely;
– Prevent damage to the spent fuel (and other high level radioactive waste) during handling and storage;
– Provide reasonable assurance that spent fuel (and other high level radioactive waste)

can be received, handled, packaged, stored and retrieved without undue risk to the health and safety of the public.

These ‘Important to Safety’ SSCs (and associated subcomponents) ensure that the following safety functions are fulfilled: (1) criticality, (2) shielding, (3) confinement, (4) heat transfer, (5) structural integrity, and (6) retrievability.

The important safety functions are abbreviated and rearranged with the following definitions, which are included in Ref. [5]:

– “Confinement Boundary (CB) − The components and supporting materials that are incorporated into the storage system design for the purpose of retaining the radioactive material under normal, off normal and accident conditions;
– Criticality Control (CC) – The components and supporting materials that are incorporated into the storage system design for the purpose of maintaining the contents in a subcritical configuration under normal, off normal and accident conditions;
– Radiation Shielding (RS) – The components and supporting materials that are incorporated into the storage system design for the purpose of reducing radiation emitted by the contents under normal, off normal and accident conditions;
– Heat Transfer (HT) – The components and supporting materials that are incorporated into the storage system design for the purpose of removing decay heat under normal conditions and protecting temperature sensitive components (e.g., lead shielding and seals) under off normal and accident conditions;
– Structural Support (SS) – The components and supporting materials that are incorporated into the storage system design for the purpose of maintaining the contents in a safe condition during normal, off normal and accident conditions;
– Fuel Retrievability (FR) – The components and supporting materials that are incorporated into the storage system design for the purpose of operations support (e.g., for loading, unloading, maintenance, monitoring, or transporting) and the failure of which could preclude any of the three options below:
    - Remove individual or canned SNF assemblies from wet or dry storage.
    - Remove a canister loaded with SNF assemblies from a storage cask or overpack.
    - Remove a cask loaded with SNF assemblies from the storage location” [5].

5.1.2. Structures, systems and components classified as not important to safety

As indicated in Ref. [1], although these SSCs (and associated subcomponents) might be initially classified as not important to safety (NIS) according to the design bases of the storage systems or storage facilities, their failure could prevent fulfilment of a function that is 'Important to Safety', or their failure as support to SSCs could prevent fulfilment of a function that is 'Important to Safety', which would make them also a SCC 'Important to Safety'. Some examples are the I&C quick connectors, the helicoil inserts in the cask flange, and — depending on the design bases — handling devices. In many MS, cranes to emplace storage casks are defined as important to safety.

5.2. STRUCTURES, SYSTEMS AND COMPONENTS WITHIN THE SCOPE OF THE AGEING MANAGEMENT REVIEW

According to Ref. [1], the scoping evaluation needs to differentiate SSCs (and associated subcomponents) according to the criteria specified in Sections 5.2.1 and 5.2.2 and define the intended function of each SSC (and associated subcomponents). In addition, the SSCs will be classified as either belonging to SNF assemblies or depending on individual design bases. In this latter group, it is also considered important to distinguish between DSSs and other SSCs that depend on the DSFs. Also, DSSs are separated depending on whether they are 'bare fuel' type or 'canister with overpack' type.

5.2.1. Spent nuclear fuel assemblies

Although the SNF assembly is not necessarily an SSC of the DSS or DSF, it is included in the scope because SNF has to remain in its approved configuration during the entire period of operation of the DSS or DSF, as stated in Refs [1] and [15]. The safety analyses for a DSF or DSS (e.g., criticality and shielding analyses) may depend on the fuel assembly maintaining a specific configuration (e.g., geometric form, or a defined number of fuel rods or filler rods in the fuel assembly lattice).

Therefore, the condition of the fuel assembly hardware and cladding are within the scope of the AMR and need to be reviewed for ageing mechanisms and effects that may lead to a change in the analyzed fuel configuration. Specific focus is given to the integrity of the cladding, spacer grids, lower and upper end fittings, guide tubes (PWR and WWER fuel) and for BWR fuel assemblies, on the channels. Details of ageing effects of the fuel are not given here, as these are given in more detail in Refs [24–26] that contain the findings of several IAEA CRPs.

5.2.2. Structures, systems and components depending on individual design bases

The information herein provided is not intended to be used as a checklist or as a scope setting document, but only for reference. Scoping is cask design and storage facility specific; results depend on the facility design and current licensing bases and how a [particular] SSC is credited in the [design basis] safety analyses or described in the FSAR. The inclusion of a certain SSC in this Section does not mean that this particular SSC has to be within the scope of the AMR for all facilities. Conversely, the omission of a certain SSC from this Section does not mean that this particular SSC is not within the scope of ageing management review for any facilities.

Reference [2] can be used to check all general cask components and their definitions. In addition, Ref. [4] provides selected definitions and usages of terms for describing and standardizing certain SSCs.

5.2.2.1. Dry storage casks

Each dry storage cask design has its own Safety Analysis Report (SAR), where the fulfilment of every safety function is demonstrated. The relationship between the SSCs (and associated subcomponents) and their intended safety function or functions is well defined for every case.

Tables 1–3 in this Section are first divided into major component areas (called items), which in turn are subdivided into their main subcomponents (SSCs). These Tables also provide information regarding the description of each component and its intended safety functions. In addition, Ref. [3] provides detailed information on the components of different standardized commercial casks.

The SSCs for 'canister with overpack' type cask designs include a fully welded metal cylinder, with a fuel basket inside, that is placed inside an overpack or a module for storage as shown in Fig. 9. Those modules can be oriented vertical or horizontal. Typically, this design requires a transfer cask to carry out transfer of the loaded canister to or from the overpack or transportation cask.

FIG. 9. Loading of a horizontal storage module (courtesy of South California Edison).

The SSCs for 'bare fuel' type cask designs that are used for SNF storage include various metallic and non-metallic subcomponents of the cask body, the closure system, the fuel basket, the lifting and handling devices, and other auxiliary subcomponents as listed in Table 1 and 2.

5.2.2.2. Dry storage facilities

In this Section, typical designs and technologies of DSFs are analyzed. Although a specific study needs to be carried out for each case, the most common SSCs that are present in DSFs are included in this Section. Depending on the national regulatory framework, infrastructure related to storage systems may be considered a part of the storage system.

Table 3 is provided as a basic guide to identifying SSCs included in the scope of the AMR. Again, this Table is divided into facilities (called items) and then into major component areas (SSCs). Owing to the complexity and diversity of the designs in these facilities, information is not included at the subcomponent level, since it would be part of the specific design of each DSF. Also, this Table provides information about the description of each SSC and its intended safety functions.

References [3] and [4] provide detailed information about a specific HSU system and could be used to develop a particular study. Reference [3] includes the analysis of a concrete pad.

TABLE 1. DRY STORAGE CASKS: ‘CANISTER WITH OVERPACK’ TYPE

Item	Description	SSC	Description	Safety function
Sealed metal canister	Metal cylinder that is sealed at both ends.	Basket	A honeycombed structural weldment with square openings that contains a storage position for each SNF assembly of the type for which it is designed.	SS, CC, FR, HT
		Canister	A cylindrical, welded metallic structure that contains the SNF assemblies or other high level waste (HLW) and performs the function of confinement.	CB, HT, RS, SS
		Drain and vent components	Used to remove moisture and air from the inner cask and backfill it with inert gas.	CB
Overpack	A concrete/metal structure into which a sealed metal canister is placed.	Concrete/ metal casks	Movable structures with one storage cavity.	SS, HT, RS
		Storage module	Ventilated (or not ventilated) modular reinforced concrete structure used to store sealed metal canister.	RS, SS, HT, FR
Transfer cask	Provides shielding during canister movements between workstations and the overpack.	Cask body	Cylindrical shells and bottom and top cover plate or lids. Some designs include neutron shielding and/or secondary outer shell. Some designs include o rings.	RS, SS, HT
		Cask lid	Closure system.	RS, CB
		Water jacket	Exterior protection for neutron shielding.	RS
		Cask attachments	Lifting and handling features.	SS
Ventilation system	Provides passive cooling in the dry storage cask.	Inlet and outlet vents	Air ducts to allow passive cooling of the canister by natural convection.	HT
		Annulus between the canister and the overpack	The flow through the annulus cools the hot canister surfaces and decay heat is transferred to the outside .	HT
Lifting and anchoring	Used to handle the main items of the dry storage cask or to attach it to pads or structures.	Trunnion	A vertical handle for the cask or a pivot point for the rotation of the cask.	SS, FR
		Anchor studs	Devices used to attach the dry storage cask to the ISFSI pad at a site.	SS, FR

Notes: Section 5.2.2.2 provides specific information for the SSCs associated with particular DSSs;

CB = Confinement Boundary; CC = Criticality Control; FR = Fuel Retrievability; HT = Heat Transfer; RS = Radiation Shielding; SS = Structural Support.

TABLE 2. DRY STORAGE CASKS: 'BARE FUEL' TYPE

Item	Description	SSC	Description	Safety function
Cask body	Metal cask that envelops the inner cavity. Includes drilled holes where the bolts from the closure system are attached. For cast casks the neutron shielding material is embedded in the thick wall which provides the intended safety functions of the inner and outer shell.	Inner shell	Metal forging, ductile cast iron or multi wall assembly that contains the SNF assemblies or other HLW.	CB, RS, SS, HT
		Neutron shielding material	Non metallic material that moderates neutrons and can absorb some neutrons.	RS
		Outer shell	The outermost radial surface of the cask.	RS, SS, HT
	Lids bolted or welded to the cask body. Ensures the sealing of the inner cavity to avoid the release of radioactive material to the environment and protect the inner cavity. Allows the retrieval of the SNF located in the inner cavity.	Inner lid, inner lid bolts, inner lid seal	Closure and sealing system of the inner cavity.	CB, RS, SS
		Inner lid drain and vent penetrations, bolts and seals	For pressurization with inert gas and removal of water and moisture (drainage and drying process).	FR
		Outer lid, outer lid bolts, outer lid seal	Closure and sealing system of the inter lid cavity.	RS, SS
		Outer lid control pressure penetration, bolts and seals	Allows inter lid drainage and drying process.	FR
		Pressure monitoring device	Monitors pressure (inter lid or inter seal space) to detect any potential leak.	CB
Basket	Locates the spent fuel contents inside the inner cavity.	Spent fuel support structure	Provides support and structural protection to the SNF or other HLW.	SS, HT
		Neutron absorber material	Maintains the radioactive content in a sub critical state.	CC, HT
Lifting and handling	Devices to move the cask from the spent fuel pool area to a storage facility.	Trunnions and bolts	-	FR
Ancillary shielding components	Provides additional protection against environmental conditions and reduces doses to personnel in the storage facility.	Protective cover/anti air crash cover Auxiliary shielding equipment	-	RS, SS

TABLE 3. DRY STORAGE FACILITIES

Item	SSC	Description	Safety function
Storage building	Canister	A metal cylinder that is sealed at both ends and is used to store SNF assemblies or other HLW.	CB, FR, CC, HT, SS, RS
	Well	Vertical metal element in the form of a cylinder that keeps canisters in a vertical position.	HT, SS, FR, CB
	Storage building	Reinforced concrete structure (walls, floor slabs, roof, foundation slab) that houses the storage wells.	RS, HT, SS, CC, FR
	Canister lifting and transferring system	Equipment used to grab, lift, transport and position the canister in the well for storage (and reverse the operation, if required).	SS, RS, HT, FR
	Ventilation system	Air inlets and outlets that provide passive cooling to the well storage in the vaults. Heat removal is by natural convection.	HT

TABLE 3. DRY STORAGE FACILITIES (cont.)

Item	SSC	Description	Safety function
Horizontal storage unit (HSU)	Horizontal storage module (HSM)	Low profile structure constructed from reinforced concrete and structural steel (walls, floor, roof), which contains an access door and a passive ventilation system.	RS, SS, HT, FR
	Basemat (pad)	Reinforced concrete slab serving as a stable foundation for the HSM.	SS, FR
	Approach pad	Approach pad is a concrete slab adjacent to the basemat used for loading and retrieval of the canisters.	SS, FR
	Shielded access door	A thick shielded access door that covers each HSM access opening or docking flange in the front wall. Allows transfers of canisters.	RS, FR
	Dry shielded canister (DSC) (see 'sealed metal canister' in Table 1)	Welded metal cylinder that is sealed at both ends and stores SNF.	CB, FR, CC, HT, SS, RS
	Transfer cask (as described in Table 1)	A cylindrical vessel with a bolted top cover plate and a welded bottom end assembly used for canister transfer operations.	RS, SS, HT, FR
	Ventilation system	Air inlets and outlets for passive cooling of the canisters.	HT
	Canister transfer facility (optional)	Underground ventilated reinforced concrete and steel structure used in some cases, transfer of canister from the transport cask to the transfer cask	SS, HT, RS, FR

TABLE 3. DRY STORAGE FACILITIES (cont.)

Item	SSC	Description	Safety function
Vertical storage unit (VSU) (Above Ground[a])	Dry shielded canister (DSC) (see “sealed metal canister” in Table 1)	Welded metal cylinder that is sealed at both ends and stores SNF	CB, FR, CC, HT, SS, RS
	Overpack (see ’overpack’ in Table 1)	A concrete structure into which a sealed metal canister is placed.	SS, HT, RS, FR
	Pad	Reinforced concrete outdoor foundation slab where VSU are placed	SS, FR
	Ventilation system	The inlet and outlet vents that are located at grade level.	HT
	Cavity enclosure container (CEC)	A cylindrical cavity consisted of a thick shell integrally welded to a bottom plate.	SS, RS, HT, FR
	Closure lid	A steel structure filled with plain concrete.	RS, HT, FR
	Subgrade	The lateral space between each CEC and the pad and is filled with a controlled low strength material or, alternatively, ‘lean concrete’.	SS, RS, HT
	Transfer cask (as described in Table 1)	A cylindrical vessel with a bolted top cover plate and a welded bottom end assembly used for canister transfer operations.	RS, SS, HT, FR

TABLE 3. DRY STORAGE FACILITIES (cont.)

Item	SSC	Description	Safety function
Vertical storage unit (VSU) (Below Ground)	Cavity enclosure container (CEC)	A subterranean cylindrical cavity consisted of a thick shell integrally welded to a bottom plate.	SS, RS, HT, FR
	Closure lid	A steel structure filled with plain concrete	RS, HT, FR
	Pad	A monolithic reinforced concrete structure providing a load bearing surface for the cask transporter and enables rainwater to be channeled away from the storage arrays.	SS, FR
	Support foundation pad (AFP)	Underground pad robustly supporting the storage module underground and minimizes long term settlement.	SS, FR
	Subgrade and under grade	The subgrade is the lateral space between each CEC, the SFP, and the pad, and is filled with a controlled low strength material. The space below the SFP is referred to as the under grade.	SS (subgrade and under grade), RS (subgrade)
	Dry shielded canister (DSC)	Welded metal cylinder that is sealed at both ends and stores SNF.	CB, FR, CC, HT, SS, RS
	Transfer cask	A cylindrical vessel with a bolted top cover plate and a welded bottom end assembly used for canister transfer operations.	RS, SS, HT, FR
	Ventilation system	The inlet and outlet vents that are located at grade level.	HT
	Engineered features	Engineered fill.	RS

TABLE 3. DRY STORAGE FACILITIES (cont.)

Item	SSC	Description	Safety function
Cask storage facility	Storage area (typical of indoor facilities)	Reinforced concrete structure (walls, floor slabs, roof, foundation slab) where the casks are stored in a vertical or horizontal position.	SS, HT, RS, FR
	Basemat (pad) (typical of outdoor facilities)	Reinforced concrete outdoor foundation slab where the dry storage casks are placed in a vertical or horizontal position.	SS, FR
	Ventilation system (typical of indoor facilities)	Air inlets and outlets for passive cooling of the casks located in the storage area. Heat removal is by natural convection.	HT
	Lifting and specific manipulation tools	Specific devices used to manipulate, lift and place casks in the storage position (and reverse the operation, if required).	SS, FR
	Canister transfer facility (optional)	Underground ventilated reinforced concrete and steel structure used to effectuate, in some cases, transfer of the canister from the transport cask to the transfer cask (and from the transfer cask to the storage cask before placing it in the storage area).	SS, HT, RS, FR
	Anchor studs (optional)	Devices used to attach the cask to the basemat.	SS
	Engineered features (optional)	Earthen berm, shield walls surrounding the basemat.	RS

[a] The VSU above ground could be single not ventilated or multiple according to Fig. 2.

5.3. OTHER CONSIDERATIONS

There are some components of DSFs that have not been mentioned in Section 5 but, in some cases, can be included in the scope of the AMR. For example, radiation monitoring instrumentation can be included in the scope if its measurements affect the fulfilment of a safety function. Other instrumentation or active components/systems subject to a change in configuration or replacement based on a qualified life or service time can also be included.

Examples of equipment normally excluded from the scope are as follows:

- Security equipment;
- Medical service equipment;
- All auxiliary or temporary equipment, including items needed for cask loading and unloading, e.g. vacuum drying equipment, portable radiation survey equipment, welding and sealing equipment, lifting rigs and slings, and other tools, fittings, hoses, measuring devices, and gauges.

6. TECHNICAL UNDERPINNING SPECIFIC AGEING MECHANISMNS

The development of an effective AMP requires an understanding of the degradation mechanisms that may be active for the storage system materials in their service environments. Several resources are available to aid this understanding, including inspection findings and other operating experience from the nuclear and non–nuclear industries, reports from research programmes, and technical documents that summarize and evaluate available ageing information (e.g., textbooks, engineering handbooks, industry standards and guidelines). For example, NUREG–2214 [15] and Electric Power Research Institute Report 1010639 [27] provide summary evaluations of materials, service environments, and ageing mechanisms for DSSs and NPP components, respectively.

Various ageing mechanisms in credible and noncredible environments are summarized in Table 4 for concrete overpacks and support pads, Table 5 for materials for spent fuel assemblies, and Table 6 for materials for various components in dry cask storage systems. The environment abbreviations used in the tables below are as follows: outdoor air (OD), demineralized water (DW), embedded in concrete (E–C), embedded in metal (E–M), embedded in neutron shielding (E–NS), fully encased or lined (FE), helium (HE), groundwater/soil (GW), and sheltered (SH).

6.1. FUNCTIONAL MATERIALS

The state of knowledge of ageing mechanisms relevant to different types of storage systems are explored below for three specific examples; stress corrosion cracking (canister overpack systems), creep of seals (bolted casks), and alkali—silica reactions (systems that include concrete structures). These examples are not intended to provide a complete description of all ageing mechanisms, rather to illustrate how storage system materials need to be systematically evaluated to understand how they may degrade under thermal and radiation exposures, corrosive environments, and mechanical stressors.

TABLE 4. AGEING MECHANISM EVALUATIONS FOR CONCRETE OVERPACKS AND SUPPORT PADS

Ageing Mechanism	Credible Environments	Noncredible Environments
Freeze and Thaw	OD, GW (above freeze line)	SH, FE, GW (below freeze line)
Creep	–	all
Reaction with Aggregates	all (where moisture available)	–
Differential Settlement	GW	OD, SH
Aggressive Chemical Attack	OD, GW	SH, FE
Corrosion of Reinforcing Steel	OD, GW	SH, FE
Shrinkage	–	OD, SH, GW, FE
Leaching of Calcium Hydroxide	OD, SH, GW	FE
Dehydration at High Temperature	–	OD, SH, GW, FE
Microbiological Degradation	GW	OD, SH, FE
Delayed Ettringite Formation	–	OD, SH, GW, FE
Salt Scaling	OD, GW (above freeze line)	SH, FE, GW (below freeze line)

TABLE 5. AGEING MECHANISM EVALUATIONS FOR SPENT FUEL ASSEMBLY MATERIALS

Material	Ageing Mechanism	Credible Environments	Noncredible Environments
Cladding Material (Zirconium based Alloys)	Hydride Reorientation[a,b]	HE	–
	Delayed Hydride Cracking[b]	–	HE
	Thermal Creep[b]	HE	–
	Low Temperature Creep[b]	–	HE
	Mechanical Overload[b]	–	HE
	Oxidation	–	HE
	Pitting Corrosion	–	HE
	Galvanic Corrosion	–	HE
	SCC	–	HE
	Radiation Embrittlement (Rad. Embrit.)	–	HE
	Fatigue	–	HE
Assembly Hardware Materials (Zirconium based, Inconel, and Stainless Steel Alloys)	Creep	–	HE
	Hydriding	–	HE
	General Corrosion	–	HE
	SCC	–	HE
	Rad. Embrit.	–	HE
	Fatigue	–	HE

[a] Only if hydrides are already present in the cladding,

[b] Applicable to high burnup fuel (>45 GWd/tU).

TABLE 6. DRY STORAGE SYSTEMS MATERIALS

Material	General Corrosion		Crevice Corrosion		Galvanic Corrosion[a]		MIC		SCC		Creep		Fatigue		Thermal Ageing		Rad. Embrit.		Stress Relaxation		Wear	
	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC
Steel	OD SH DW GW E–C	E–M E–NS HE	OD SH DW GW E–C	E–M E–NS HE	OD SH		GW E–C	OD SH DW E–M, E–NS HE		OD SH		OD SH DW GW E–M E–NS HE	TLAA[b]			OD SH DW GW E–M E–NS HE		OD SH DW GW E–M E–NS HE	SH	OD	OD	
Stainless Steel		OD SH DW E–M E–NS HE	OD SH	DW E–M E–NS HE	OD SH			OD SH DW E–M E–NS HE	OD SH[c]	DW E–M, E–NS HE		OD SH DW E–M E–NS HE	TLAA[b]		HE[d]	OD SH DW E–M E–NS		OD SH DW E–M E–NS HE		OD SH	OD	
Al Alloys	SH[e]	E–M E–NS HE	SH	E–M E–NS HE	SH	HE		SH E–M E–NS HE			Analyses Required[f]		TLAA[b]		Analyses Required[e]			SH E–M E–NS HE				

TABLE 6. DRY STORAGE SYSTEMS MATERIALS (cont.)

Material	General Corrosion		Pitting & Crevice Corrosion		Galvanic Corrosion[a]		MIC		SCC		Creep		Fatigue		Thermal Ageing		Radiation Embrittlement		Stress Relaxation		Wear	
	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC	C	NC
Nickel Alloys		OD		OD				OD		OD			TLAA[b]						OD		OD	
Copper Alloys	OD			OD				OD												OD		
Coatings																	Analyses Required					

Where: MIC = Microbial induced corrosion; SCC = Stress corrosion cracking.

[a] Where dissimilar material galvanic couples exist.

[b] TLAA if applicable.

[c] SCC credible at welds and other regions where sufficient stress exists; transfer cask components exposed to indoor/outdoor air are not considered to be susceptible to SCC due to periodic rinsing with demineralized water.

[d] Thermal ageing is credible only for precipitation hardened martensitic stainless steels.

[e] General corrosion is not considered to be credible for anodized aluminium.

[f] Creep and thermal ageing are relevant only for load bearing components.

6.1.1. Stress corrosion cracking of canister

The intended function of canisters for concrete shielded cask is confinement boundary as classified in Section 5.2.2.2. SCC is a credible ageing mechanism, depicted in Fig. 10, that could affect an important safety function.

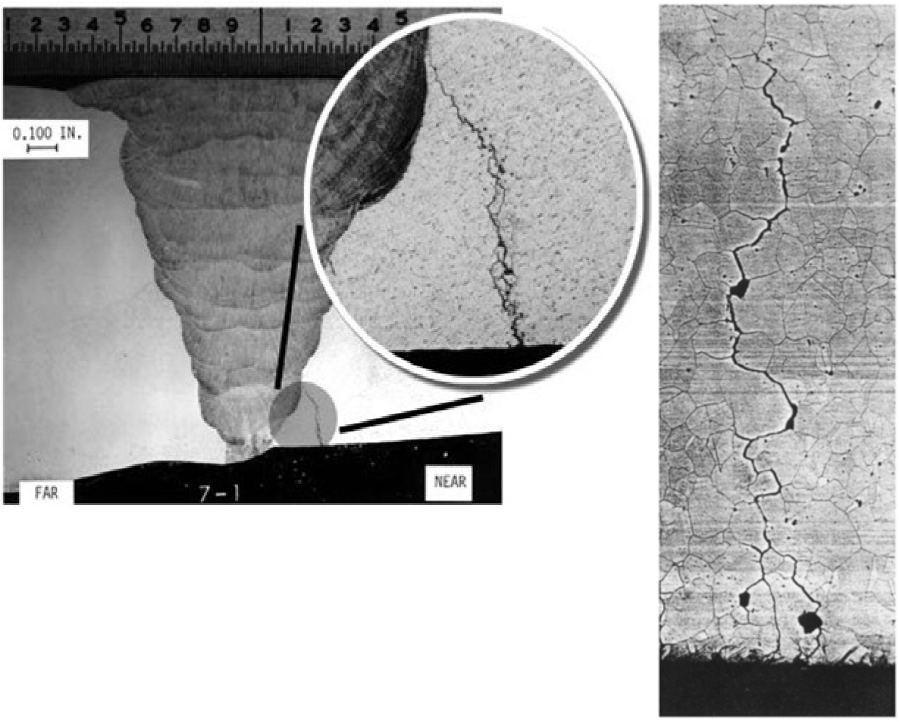


FIG. 10. Examples for SCC cracks (Courtesy of EPRI).

The mechanism of SCC, measures to prevent SCC, and monitoring of confinement are described in the following subsections.

6.1.1.1. Mechanism of stress corrosion cracking

In case of SNF stored in canister based DSSs, the surrounding ventilated concrete structures provide shielding, and the canisters act as the confinement barrier that retains the inert atmosphere surrounding the fuel assemblies. The decay heat from the SNF is transferred by the inert gas to the canister wall through convection. Heat at the canister surface is transferred to the atmospheric air that flows through the ventilated concrete shielding and over the outer surface of the canister. The atmospheric air will also bring particulate materials that may include chloride salts that can deposit on the canisters. A high chloride salt loading is found where the storage facilities are close to marine environments, but some level is also present at inland sites; Local use of road de-icing salts and close proximity to cooling towers may also be chloride salt sources.

Austenitic stainless steels are susceptible to chloride induced stress corrosion cracking (CISCC) in certain environments under tensile stress. As an example, CISCC induced by particles containing sea salt and chlorides has been observed on chemical plant structures built in coastal regions. In order for CISCC to occur, three conditions — sufficient tensile stress, aggressive chloride environment and a susceptible material — have to be satisfied. In case of the canister, those fabricated from austenitic stainless steel with welds containing residual stresses and having sea salt deposits on the surface are susceptible to CISCC. If one of those three conditions is not fulfilled, then CISCC will not occur. Avoiding at least two of the conditions is desirable for long term safe operation.

The process by which CISCC is expected to occur in a canister is represented in Fig. 11.

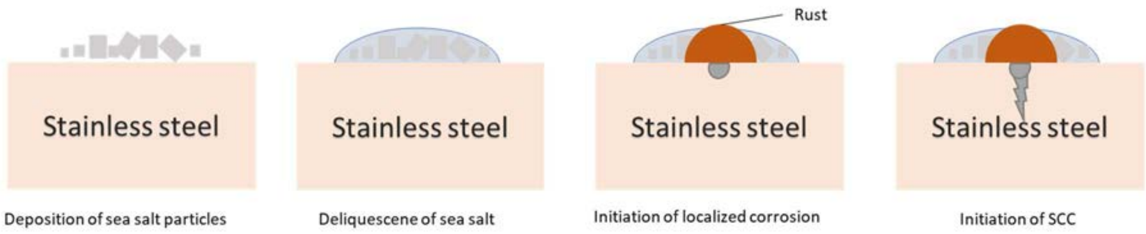


FIG. 11. SCC initiation process.

Firstly, sea salt particles deposit and build up on the canister surface. Absorption of water from humidity in the air results in the particles deliquescing into brine. The brine solution causes localized corrosion, such as pitting or crevice corrosion. Finally, cracking occurs if tensile stress exists around the corroded pit. As localized corrosion produces rust spots, cracking usually occurs in areas that also have rust formation. Each of the three critical factors required for CISCC are considered below in more detail.

(a) Stress

Canisters are fabricated from wrought plate material that is rolled to form a cylinder and welded. As most metals swell during heating, in constrained structures like a canister, residual stress around the weld line is created by the thermal cycles during heating from the welding process and followed by a shrinking process during cooling. The result is tensile stress within the material. The level of this residual stress caused by the welding process can even exceed the yield stress.

Surface finishing with a hand grinder may add additional tensile stress. Grinding typically introduces stress in two directions: with the working direction of the grinding tool and along the rotational direction of the grinder head. Surface finishing with a grinder also increases local hardness. It is reported that stress corrosion crack depth increases with surface hardness at operating temperatures typical of light water reactors (LWRs).

Compressive stresses cannot cause CISCC, as the stress counteracts crack opening. CISCC will be mitigated when the tensile stress is lowered to less than zero (i.e. the stress becomes compressive). Several stress mitigation methods, which are already applied on LWRs, are applicable to canisters. For example, shot peening, laser peening and water jet peening can be applied to introduce compression stress at and near the canister surface. Annealing is another option for mitigation of tensile stress, this time by stress relaxation. Because canister wall thicknesses are small, compared to the diameter, annealing of canisters would require a retainer to prevent deformation of the canister body during the annealing process. For this reason, annealing is not typically used for austenitic stainless steel SNF storage canisters.

(b) Environment

Sea salt, transported by cooling air, is a major environmental factor in CISCC because it is a source of aggressive ions that drive local corrosion. At sufficient relative humidity (RH), the deposited salts undergo deliquescence resulting in the formation of a saturated salt solution and the formation of the electrolytic cells necessary for corrosion. The environmental conditions for deliquescence are dependent on the composition of the deposited salt. Sea salt contains a mixture of ionic species, predominantly sodium chloride and magnesium chloride. At room temperature, sodium chloride deliquesces at 75 %RH, whilst magnesium chloride deliquesces at a much lower RH of around 30–35% [15].

In order to initiate the localized corrosion, a certain area of continuous salt is necessary, so it is anticipated that initiation and propagation of the SCC depends on the density of deposited sea salt.

(c) Material

Different alloys display different degrees of resistance to localized corrosion, which, as shown in Fig. 11, can be a precursor to the initiation of CISCC when sufficient tensile residual stresses are present. Austenitic stainless steel, such as types 304 and 316 are vulnerable to CISCC, however the addition of 2—3 weight % molybdenum to type 316 stainless steel increases the resistance to localized corrosion and CISCC compared to type 304 stainless steel. Alloys with higher chromium and molybdenum concentrations have increased resistance to localized corrosion and CISCC [15].

6.1.1.2. Stress corrosion cracking prevention

Several measures to prevent SCC are applicable from every aspect of stress, environment and material [26, 28, 29]. The best method to be applied will depend on the view of the environment of the facility and management policy of the facility operator. For example, a cost effective method is inducing compressive residual stress into the critical regions of the weldment of the canister. Surface treatment techniques such as shot peening, laser shot peeing (LSP) and low plasticity burnishing (LPB)[30] are currently being used in the industry to mitigate or impede SCC by inducing compressive residual stresses into the surface material.

Corrosion tests were carried out using a plate with a laser beam weld, half of which was surface treated by the LPB. As an accelerated condition, the salt concentration on the surface was adjusted to 10 g/m^3 as Cl, which is more than 10 times the threshold chloride concentration for SCC initiation of 304L stainless steel. Test conditions were a constant temperature of 80 ˚C with relative humidity of 35% over 1000 hours. Even in the severest environmental condition, SCC initiation of 304L stainless steel was not observed on the surface treated by the LPB, whereas the other half without the LPB initiated SCC. This indicates that surface treatment by the LPB could be effective to prevent SCC. Further demonstrative test to confirm the effectiveness of applying compressive stress on the welded area was conducted using a full scale canister in diameter [29].

6.1.1.3. Monitoring for stress corrosion cracking

An AMP is designed to monitor or inspect SSCs to detect and address degradation before there is a loss of an intended safety function (see Section 3.4). An example of an AMP for localized corrosion and SCC of welded stainless steel canisters is summarized in Section 4.3. This example includes examination of canister’s external surfaces to detect and address corrosion and SCC before a loss of the confinement function.

6.1.1.4. Additional monitoring of confinement to detect helium leakage

It is important to monitor/inspect canisters to detect and address SCC before a loss of confinement, as discussed above. However, an additional confinement monitoring method is being developed to detect helium gas leakage from a canister crack. Three different methods have been developed to detect helium gas leakage through measurements of temperatures at different points on the canister surface, to detect the change in temperatures expected to occur as the helium leak changes the heat transfer properties of the internal gas environment. The three methods are briefly described in Section 8 and radiological consequences and management for the monitoring method of helium gas leakage have been addressed in a recent literature [31].

6.1.2. Creep of lid seals for bolted metal casks

The intended function of metal seals for bolted metal casks is that of a confinement boundary as outlined in Section 4.3.2 Creep is a plausible ageing mechanism that could affect function of the seal [29]. The mechanism of leakage due to creep, longevity of the seal, and monitoring of confinement are described below.

There are various approaches describing the long term behaviour of metal seals. The use of a special approach, however, requires careful review of the boundary conditions for a specific application.

6.1.2.1. Mechanism of leakage due to creep

The containment system of a metal cask typically consists of primary and secondary lids. The containment function is ensured by inserting seals between the cask body and the lids and then bolting them together. Metal seals are used for long term durability at high temperature. Therefore, it is important to clarify the influence of the stress relaxation of the seals on the restoring force and containment performance over time.

In order to clarify the creep effect on the time dependent deformation of the seal, basic creep tests can be carried out using a scaled model of the closure system equipped with a metal seal [25]. Figure 12 illustrates a metal seal with aluminium coating.

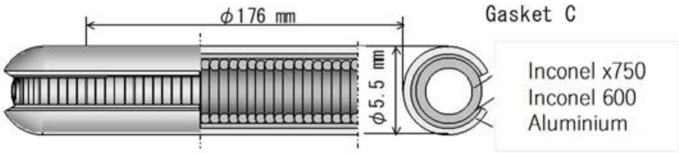


FIG. 12. Example of a metal seal with aluminium coating (reproduced from Ref. [25] permission courtesy of EPRI) .

The height recovery of an initially compressed seal can be described by the compression set (CS) Eq. (1) :

$$CS = \frac{d_0-d_2}{d_0-d_1} \times 100\ (\%) \qquad (1)$$

Figure 13 shows the definition of d_0, d_1, and d_2.

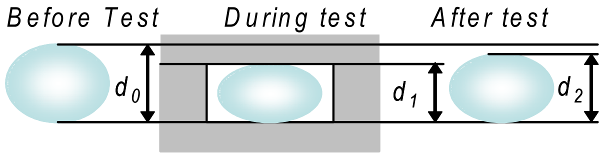


FIG. 13. Illustration of the compression set (CS, reproduced from Ref. [25] permission courtesy of EPRI).

The Larson–Miller parameter (LMP) describes thermally activated processes in materials, such as creep, using a time temperature equivalence approach of Arrhenius type Eq. (2) [32]:

$$LMP = T' \times (C + \log_{10}(t')) \qquad (2)$$

where T'=T / 1K with the absolute temperature T and t'= t / 1h with the time t. C is a constant value and quite sensitive to the LMP approach. As metal seals are a complex construction of various materials, the constant C may vary. Determination of the C value bases on a large database of test results from many seal tests. When LMP is used, the estimate for long term behaviour at lower temperature can be predicted using the result for short term behaviour at higher temperature [33, 34].

Acceleration of ageing by higher temperature has to take into account additional effects [35] which may impact the Arrhenius plot.

The relation between LMP and compression set of the seal has to be found experimentally and is exemplarily depicted in Fig. 14 [25].

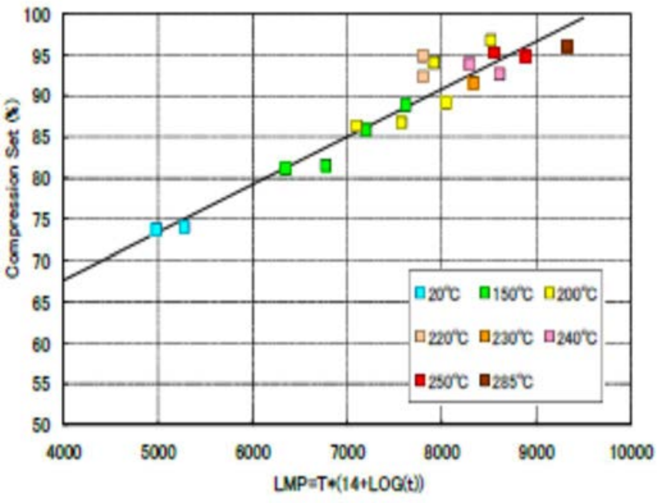


FIG. 14. Relation between LMP and compression set (reproduced from Ref. [25]).

The characteristic mechanical behaviour of metal seals can also be obtained by compression and decompression procedures. Figure 15 illustrates resulting characteristic load deformation curves with specific seal properties for some specimens [36]. Generally, the seal force is specified as a length related force Y. The markers in Fig. 15 define specific seal conditions, as stated in Fig. 15.

The usable resilience (r_u) is the distance between e_2 and e_1 and represents the ability of elastic recovery of the seal. Depending on temperature, the initial operation point (Y_2, e_2) changes over time to an operation point (Y_{2R}, e_2) with reduced seal force Y_{2R} (restoring seal force) while maintaining e_2. This also leads to a reduced usable resilience (r_u').

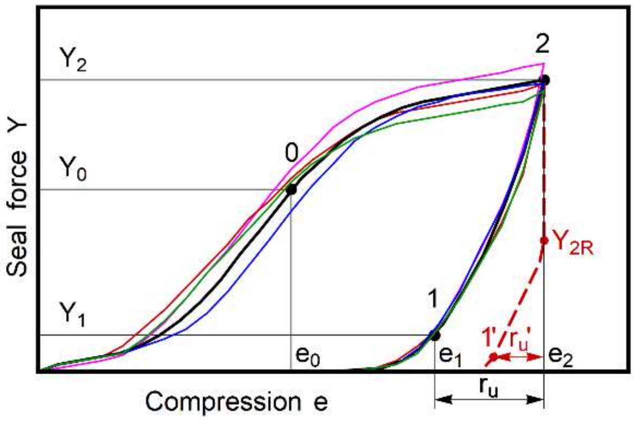


FIG. 15. Seal force and compression curves of various metal seal specimens (courtesy of BAM).

Where:
0 – load Y_0 at e_0 on the compression curve to achieve the required leak tightness.
1 – load Y_1 at deformation e_1 at which the required leak tightness is no longer met during load relieving.
2 – load Y_2 to reach optimum compression at e_2 (full contact of flanges).

6.1.2.2. Longevity of metal seals

The measurement parameter for leakage rate of a metal seal after assembly is typically the standard helium leakage rate as defined by Eq. (3) [25]:

$$Q_{He/St} = \Delta p \, V / \, t \tag{3}$$

where Δp is the pressure change (during measuring time), V is the volume, and t is the measuring time.

To demonstrate the leak tightness, the leakage rate $Q_{He/St}$ needs to be smaller than a required limit value defined for the specific seal type for each metal seal used in the cask. A loss of leak tightness may occur due to mechanical load scenarios of the cask or reduction of the restoring seal force because of time dependent creep processes of the seal and/or lid bolts.

The long term containment by metal seals has been demonstrated e.g. by using full scale cask lid structures. Methods to evaluate the containment performance have been developed. In long term containment tests [28], the leakage rates of two types of metal seals (aluminium and silver coatings) in two lid models were maintained at a constant level for about 19 years, demonstrating long term containment. The LMP values of these metal seals were calculated as 7942 and 7781 using the measured temperature. Namely, containment performance of the seal coated by aluminium has been assured until LMP = 7942 and that of the seal coated by silver has been assured until LMP = 7781. With those LMP values, a relationship between initial temperature of the seal and assured period of sealing performance has been obtained. Figure 16 shows exemplarily the relation between LMP and leakage rate.

When LMP exceeds a threshold value, leakage rate increases.

The creep tests shown in Figs 14 and 16 assumed that the creep mechanism in the temperature range from 20 to 285 °C is the same.

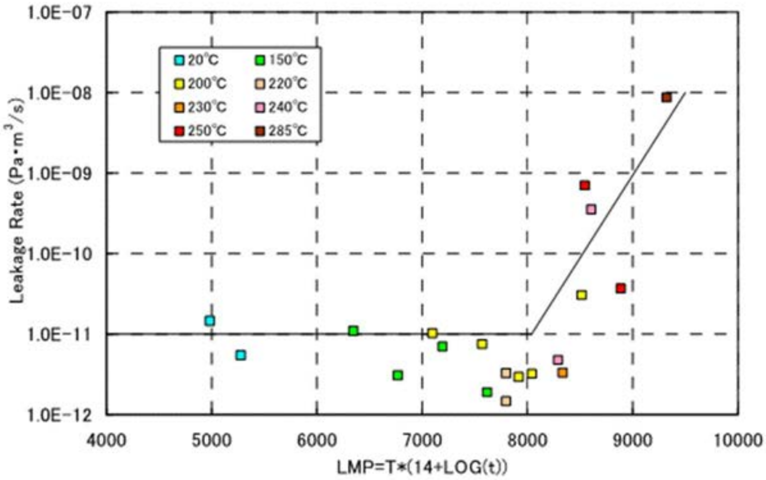


FIG. 16. Relationship between LMP and leakage rate of metal seal (courtesy of CRIEPI).

This assumption has been demonstrated in the long term containment tests of full scale cask lid models as shown in Fig. 17.

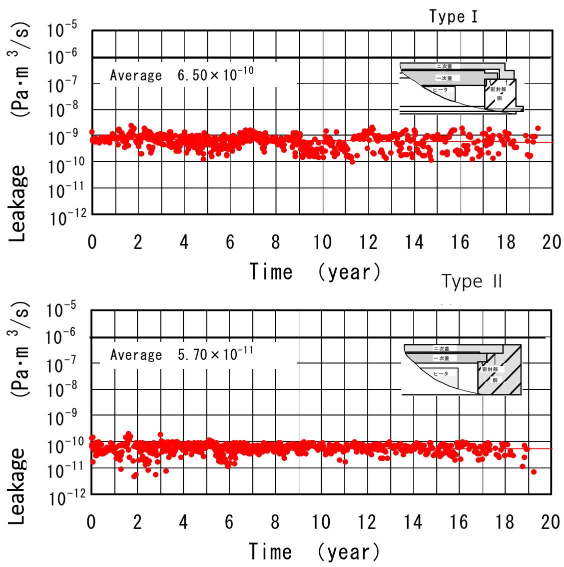


FIG. 17. Measurements of leakage rate of metal seals (Type I: aluminium coated seal, Type II: silver coated seal, courtesy of CRIEPI).

Using the data provided in Fig. 5 and assuming that the storage period is 60 years, sealing performance is assured for a seal coated with aluminium where the initial seal temperature is 134 °C or less. For specific seals and setups, the LMP needs to be investigated and verified to be able to use this reasoning.

Alternative approaches focusing on restoring seal force calculated from enhanced power law model and modified Arrhenius equation compared to test results for Helicoflex metal seals with outer aluminium and silver coatings have been reported in the literature [30].

The time and temperature dependent values of the restoring seal force Y_{2R} are schematically displayed for three different temperatures in Fig. 18. The purpose of such studies is to find a suitable function to describe the long term relaxation behaviour using experimental data in a temperature range between 20 and 150 °C based on short term experimental data [35, 36].

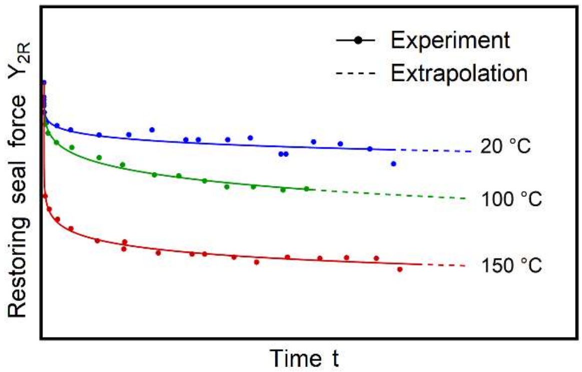


FIG. 18. Restoring seal force performance depending on time and temperature (courtesy of BAM).

6.1.3. Alkali–silica reaction of concrete

One of the degradation reactions affecting concrete is the alkali silica reaction (ASR). Some types of aggregate can react with Na^+, K^+ and OH^- ions in the pore solution, giving rise to detrimental expansion.

The principal reactions can take place with aggregate containing certain forms of amorphous or poorly crystalline silica and with dolomitic limestone aggregate (alkali carbonate reaction, ACR). ASRs and ACRs are significant causes of damage to concrete structures in many countries around the world [38].

The significance of ASR for concrete structures is that deleterious expansion causes cracking of the concrete and reduces the tensile strength, which may have consequences for the structural and shielding capacity [22].

ASR was recently identified as having affected reinforced concrete structures in the nuclear power industry [37]. An aggregate from local sources is normally used in concrete, meaning that the experiences of different countries are based on a specific range of aggregates and their specific reactivity. Operators may implement measures during initial construction to prevent ASR induced concrete degradation, such as selecting nonreactive aggregates. However, due to the uncertainties in the testing for aggregate reactivity and the operating experience with late or slow expanding aggregates [17], ASR is considered a credible ageing mechanism in concrete exposed to any environment with available moisture [18]. Operators may take further measures to prevent ASR by installing and preserving a waterproof membrane, hydrophobic treatment, or maintaining dewatering channels, to prevent further water ingress.

Regardless of these measures, visual inspections of concrete routinely identify ASR through the unique ‘map’ or ‘patterned’ cracking, and the presence of alkali—silica gel in areas where concrete is exposed to moisture and likely to experience ASR. The example AMP for reinforced concrete structures in Section 4.3.3 provides a programme for condition monitoring, performance monitoring, and prevention and mitigation activities for concrete ageing effects, including ASR. The programme includes periodic inspections of reinforced concrete as described in standards endorsed by the MSs, e.g. Ref. [16]. The programme also includes corrective actions consistent with applicable consensus rehabilitation guides or standards endorsed by MSs.

7. OPERATING EXPERIENCES (LESSONS LEARNED)

This Section summarizes operating experiences and lessons learned [38] associated with ageing management for different DSSs as provided by some MSs.

7.1. BOLTED CASK SYSTEMS

Bolted cask systems are a dry technology that have developed over the last 30–35 years. They can be regarded as an established industrial technology with a high level of reliability and robustness. To date, no such casks have been involved in serious incidents or accidents, and even during the Fukushima Daiichi accident, loaded casks at the Fukushima Daiichi NPP site performed well. However, several operational occurrences have been reported in countries using these systems [38].

Corrosion of trunnion bolts has been identified in some cask storage facilities after several years of cask storage. From the perspective of the long term storage of SNF, bolt corrosion can affect the safe handling of the cask. This has resulted in control and evaluation of the trunnion bolt status by the operator of the storage facility in close cooperation with the national regulatory body. Trunnion bolts have been inspected for corrosion and if needed, replaced and resealed. The design of new, modified casks has been changed so that all trunnion bolts are protected by a stainless steel ring with sealing. This design prevents the intake of borated water into the bolt shafts during the cask loading procedures in the SNF pool [39].

7.1.1. Dual purpose casks in Japan

A storage facility at Mutsu City using DPCs for fuel destined for reprocessing was licensed in 2010. Prior to the project, two electric power companies operated on-site storage systems using bolted casks (for storage only) [40]. This Section provides information related to the integrity of the dry metallic storage casks and their contents in NPPs, which was requested by the Nuclear Safety Commission in “Long term Integrity of the Dry Metallic Casks and Their Contents in the Spent fuel Interim Storage Facilities.” [7]

7.1.1.1. Integrity of spent fuel

Tokyo Electric Power Company Holdings (TEPCO) has been storing BWR fuel, using nine dry storage casks (four medium sized casks for 37 assemblies and five large sized casks for 52 assemblies), at the site of the Fukushima Daiichi NPP since September 1995. Similarly, Japan Atomic Power Co., Inc. has been storing BWR fuel using 15 casks for 61 assemblies at the Tokai Daini NPP since December 2001. Some of these casks were opened after several years to inspect the integrity of the spent fuel inside the casks: TEPCO opened casks in 2000 and 2005 (5 and 10 years after the start of storage, respectively), and Japan Atomic Power Co., Inc., opened casks in January 2009 (7 years into the storage period), and the following facts were reported [6].

(a) Items of inspection

Confirmation of presence/absence of krypton gas in the casks and visual check of spent fuel stored in the casks.

(b) Inspected fuel and casks

The following types of fuel and one dry storage cask for each fuel type were inspected according to Table 7:

- FA1, 8×8 type fuel (stored for approximately 10 years at the Fukushima Daiichi NPP);
- FA2, new 8×8 type fuel (stored for approximately 5 years at the Fukushima Daiichi NPP);
- FA3, new 8×8 zirconium liner type fuel (stored for approximately 7 years at the Tokai Daini NPP).

TABLE 7. CHARACTERISTICS OF THE ASSESSED FUEL THROUGH PRESENCE/ABSENCE OF KRYPTON GAS[a]

	Fukushima Daiichi NPP		Tokai Daini NPP
Fuel type	8×8 fuel	New 8×8 fuel Zirconium liner fuel	New 8×8 fuel Zirconium liner fuel
Storage period	~10 years	~5 years	~7 years
Burnup of the stored fuel	Up to ~28 GWd/t	Up to ~32 GWd/t	Up to ~33 GWd/t

[a] Estimates of the fuel cladding tube temperatures (at the time of vacuum drying/at the beginning of the storage/at the time of inspection) were approximately 90°C/90°C/80°C (8×8 fuel), approximately 145°C/140°C/115°C (new type of 8×8 fuel in Fukushima Daiichi NPP), and approximately 140°C/165°C/140°C (new type of 8×8 fuel in Tokai Daini NPP).

(c) Results of inspection

As a result of checking the gas inside the cask for each of those dry storage cask, no krypton gas was detected, and no breakage of the fuel cladding tube was observed after each storage period (approximately 10 years maximum).

No significant change was observed in the appearance of the fuel assemblies after each storage period (approximately 10 years maximum) in comparison to the start of the storage as shown in Fig. 19.

7.1.1.2. Integrity of metallic casks

For metallic casks the most attention to the confinement boundary has been given the metallic gasket in the bolted lid system.

(a) Results of leak tight tests on primary lid

During the inspections performed to date at the Fukushima Daiichi and Tokai Daini NPPs, the leak tightness of the primary lid of the dry storage cask used for storage was tested; the structure of the lid of the dry storage casks used in these plants is virtually the same as the lid structure of the metallic casks[1]. Test results are shown in Table 8.

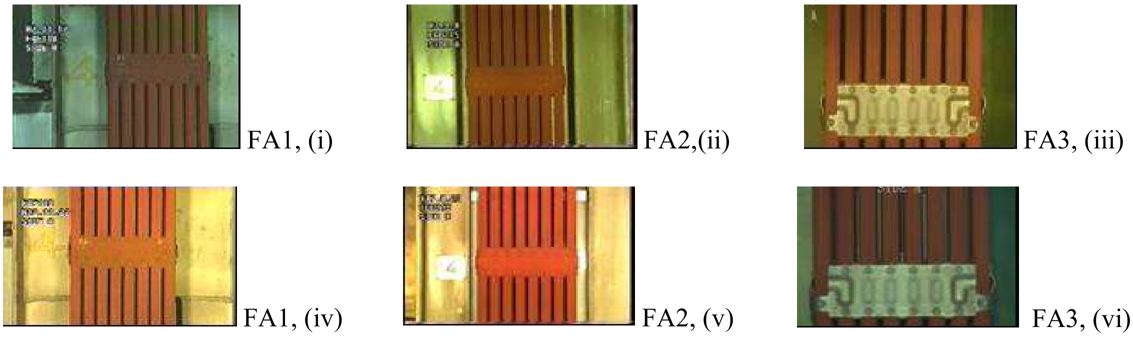


FIG. 19. Visual inspection of dry stored fuel at Fukushima Daiichi and Tokai Daini NPPs. At the start of storage (i ‐iii) and at time of inspection (iv ‐vi). The photos show the central area of the fuel (around the fourth spacer from the top). Differences in colour are due to differences in photographing conditions (courtesy of TEPCO).

TABLE 8. RESULTS OF LEAKTIGHTNESS TESTS OF PRIMARY LIDS

	Fukushima Daiichi NPP (PaM^3/s)		Tokai Daini NPP (PaM^3/s)
	Inspection in 2000	Inspection in 2005[a]	Inspection in 2009
At the time of fuel loading[b]	9.0×10^{-10}	4.9×10^{-10}	1.6×10^{-10}
At the time of inspection	5.3×10^{-8}	1.6×10^{-7}[b]	9.0×10^{-11}
Standard leak rate (criteria)[c]	Below 1×10^{-6}		Below 1.6×10^{-7}

[a]The metal gasket that was used in the Fukushima Daiichi NPP at the time of the 2005 inspection was replaced in 2001,
[b]Fukushima Daiichi NPP, 1995; Tokai Daini NPP, 2002,
[c]The leak rate required to keep the inner pressure of the cask negative during the designed storage period.

Although results of the leak tightness test at the Fukushima Daiichi NPP satisfied the leak tightness test criteria, leak rates at the time of inspection were approximately two orders of magnitude higher than the values at the time of fuel loading.

At Tokai Daini NPP, the cask loading procedure requires a plug to be installed in the leak check hole in the primary lid to close it and prevent pool water access. The inspected section had not been exposed to the helium gas environment; thus, results of the leak tightness test,

[1]The lid is designed to have a double sealing structure consisting of the primary lid and the secondary lid, and the pressure inside the space between the two lids can be monitored. Moreover, a metal gasket is used at the sealing section of the lid.

which was carried out in 2009, were almost equivalent to the leak rate at the time of fuel loading.

Considering these differences, the results of the leak tight test at the Fukushima Daiichi NPP are attributed to the fact that the inspected part had been exposed to the helium gas environment during storage; thus, the leak rate appeared to be higher because of the remaining helium gas at the location of the inspected part[2].

(b) Results of visual check of metal seal of primary lid

Although results of the leak tightness test at the Fukushima Daiichi NPP showed no abnormality, whitening was observed on the metal seal surface during the visual check. On the other hand, no abnormality was observed by visual check at the Tokai Daini NPP (note that the material and structure of the metal seal used in the dry storage casks and in the metallic casks are the same), as shown in Fig 20.

The whitening observed during the inspection at the Fukushima Daiichi NPP in 2000 was attributed to the insufficient removal of pool water from the flange of the primary lid during the preparation of the metallic cask for the inspection in 2000; the remaining moisture caused aluminium (contained in the metal seal cladding) to whiten during storage. Subsequently, TEPCO ensured thorough removal of water during preparation of metallic casks. At the Tokai Daini NPP, where a similar inspection was conducted in 2009, the same material was used in the metal seal of the dry storage cask. As whitening of the metal seal was not observed, the validity of the above improvement was confirmed. It is necessary to continue thorough water removal in the future when preparing metallic casks, to ensure the reliability of the sealing function.

During the inspection at the Fukushima Daiichi NPP in 2005, whitening was observed on the entire surface of the metal seal. TEPCO personnel noted that when they had transported the dry storage cask to the pool in 2005 to open the primary lid of the cask for a visual check of the spent fuel, it took several days to open the primary lid in the pool, and thus the many hours of contact with pool water were considered the cause of the surface whitening. Subsequently TEPCO avoided keeping the dry storage cask in the pool for many hours before opening the primary lid for the inspection[3]. At the Tokai Daini NPP, where a similar inspection was conducted in 2009 without lengthy contact with pool water, no whitening was observed on the metal seal, confirming the validity of the observation.

[2] The result of the leak tightness test at the Tokai Daini NPP is one order of magnitude below the value at the start of storage. This level of variability in the leak-tight test results is also seen in current tests. Therefore, it cannot be said that the leak-tightness of the lid increased during storage.

[3] Before the cask inspection at the Tokai Daini NPP in 2000, the cask was in the fuel pool for about 1 hr, and no whitening was observed on the metal gasket of the lid.

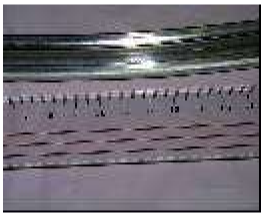
	Fukushima Daiichi NPP		Tokai Daini NPP
	Inspection in 2000	Inspection in 2005	Inspection in 2009
Example photos			
	Enlarged photo	Enlarged photo	Enlarged photo
Inspection results	· Whitening on some parts of the metal seal surface. · No abnormalities on the seal surface.	· Whitening on the metal seal surface · No abnormalities on the seal surface.	· No abnormalities on the metal seal. · No abnormalities on the seal surface[a].

FIG. 20. Visual inspection of primary lid metal seals (courtesy of TEPCO).

Notes: Estimated temperatures of the metal gasket (at the start of storage/at the time of inspection) were approximately 65°C/60°C at the Fukushima Daiichi NPP inspection in 2000, approximately 55°C/55°C at the Fukushima Daiichi NPP inspection in 2005, and approximately 95°C/85°C at the Tokai Daini NPP inspection.
[a] Reddish brown appearance of some areas is due to reflection of lighting.

7.1.1.3. Impact of Fukushima Daiichi accident

In 2011, the Great East Japan earthquake (magnitude 9.0–9.1) and subsequent tsunami impacted the Fukushima Daiichi NPP site. One of the affected buildings was the cask custody building that was storing nine dry metallic casks [41]. The cask custody building was located beside the sea, and a vast amount of seawater, sand, and debris entered the building when the tsunami occurred.

From the casks' external appearance, as well as results of temperature and dose measurements, it was concluded that the safety performance of the dry casks had been maintained [41]. As it was difficult to continue using the cask custody building, the casks were inspected in the common pool. The inspection included the following:

- Leak test of the primary and secondary lids (Containment Function);
- Visual inspection of sealing parts (Containment Function);
- Visual inspection of fuel cladding on 3 bundles (sub–criticality function and spent fuel integrity);
- Gas sampling to detect Kr–85, if any (Spent Fuel Integrity).

The visual inspection of the metal gaskets revealed whitening at the periphery due to the seawater exposure. All of the secondary lid metal gaskets were replaced. After all inspections, it was confirmed that the metal casks had maintained their safety functions despite the earthquake and tsunami. The metal casks were moved to a newly constructed temporary cask custody area built on a hill on the Fukushima Daiichi NPP site to reduce the tsunami risk.

7.1.1.4. Evaluation of accumulated knowledge

Summarizing the examinations and inspections described above the following conclusions were drawn with respect to the stored fuel and the casks.

(a) Integrity of spent BWR fuel

Inspections of stored 8 × 8 fuel, a new 8 × 8 fuel assembly, and a new 8 × 8 zirconium liner fuel assembly have been conducted over approximately 10 years in NPPs, and so far there has been no observation of breakage of fuel cladding tubing or changes in the appearance of the fuel assemblies. Burnup rates of the inspected fuel, shown in Table 7, ranged from approximately 28 to 33.5 GWd/t, which are typical burnup rates for each fuel type. Estimated fuel cladding tube temperatures (at the beginning of the storage) ranged from approximately 90 to 165 °C.

(b) Integrity of metallic casks

From the inspections of dry storage casks performed in NPPs, it was confirmed that the integrity of sealing performance was preserved for a certain storage period. Moreover, knowledge related to the integrity after Fukushima Daiichi accident can contribute to further enhancement of reliability of the dry storage casks. This knowledge can be applied to the metallic casks, whose basic structure is virtually the same as that of the dry storage casks.

7.1.2. Dual purpose casks in Germany

Dry storage of SNF and vitrified waste from reprocessing is undertaken in Germany using DPCs[26]. According to the recommendation of the German Nuclear Waste Management Commission (ESK), all DPCs are equipped with a monitored double lid system. Monitoring of the sealing function takes place in the pressurized inter lid space, in combination with a pressure switch, which triggers an alarm signal in the surveillance system of the spent fuel storage facility (SFSF) if the pressure drops below the specified minimum pressure difference. The switch is also self–monitoring, so that even internal defects lead to an alarm signal and the switch can be replaced if necessary.

Currently, there are about 1300 DPCs in 16 SFSFs, where some of these DPCs have already been stored for up to 28 years. The interim storage facilities can be divided into 3 central storage facilities (Ahaus, Gorleben, Lubmin) and 13 decentralized facilities, 12 located at the sites of NPPs and one at the site of a decommissioned research reactor. The majority of the DPCs were wet loaded in the pools of the NPPs and then vacuum dried. All cask loadings of vitrified waste and most SNF from research or prototype reactors were under dry conditions. For the DPCs' cask bodies, ductile cast iron (CASTOR cask types) and forged steel (TN cask types) are used as depicted in Fig. 21.

Because of the shielding capability of DPCs and their storage in an accessible hall, the casks can be checked at any time during normal operation. As a result, corrosion findings on the external surfaces would be detectable even without special inspections within the scope of the operations in the storage facility. However, experience to date shows that no safety relevant damage has occurred; the findings are limited to signs of wear and tear and locally limited corrosion due to damage to the paint caused by use. For handling, the metal casks are equipped with one or two pairs of trunnions, which are bolted to the cask body. Initially, the trunnions of the CASTOR THTR/AVR casks were made of construction steel, with surfaces that were chrome plated for corrosion protection. With the increasing number of crane handlings of the CASTOR THTR/AVR during routine loading, transport, and storage operations, it became evident that the loads on the trunnions resulted in wear marks that were associated with abrasion of the chromate layer. As a result of the abrasion of the protective layer, corrosion occurred that required repair measures. It was decided to dismantle the trunnions and either recoat them or replace them with new trunnions made of stainless steel.

FIG. 21. TN 85 and CASTOR HAW28M casks in the SFSF at Gorleben, Germany (courtesy of BGZ).

The application of this procedure was possible on loaded casks because the metal casks have very good shielding properties, and the necessary installation work could therefore be carried out without complex radiation protection measures. After the trunnions were replaced and bolted, overload testing on the loaded casks was successfully conducted. Based on this experience, all later generations of CASTOR casks were supplied exclusively with stainless steel trunnions. This experience also shows that corrective measures can be implemented even after long term storage.

Owing to the lengthy operation of the storage facilities (since the mid 1980s) and the large number of casks stored, extensive operational experience with this storage system is available in Germany. This experience also includes preparation for shipment after a long term storage period. In this context, it has been shown that the double lid system can still be tested after storage, with the only difference being an increase in the testing times compared to those at the time of cask loading. These findings are in accordance with the results obtained in Japan after testing of the leakage rate of long stored casks.

Another finding, resulting from the operation of the centralized storage facilities, is that the heat loads assumed for the design of the storage buildings are far from being achieved in reality. There are two explanations for this fact. First, most cask loadings, especially those of casks for SNF assemblies, were carried out in such a way that the design capacity of the cask was not fully utilized. Second, because of the change in the waste management strategy in Germany (i.e., the erecting of decentralized SFSFs at the NPP sites), the originally assumed occupancy of the centralized storage facilities in Ahaus and Gorleben was never reached.

In Ahaus, this low heat load resulted in an imbalance between the heat output of the stored casks and the heat removal from the storage building. As a result, the surface temperatures of some particularly cold casks fell below the dew point and condensation of water from the ambient air occurred, resulting in localized corrosion and paint flaking. To avoid further condensation, the ventilation of the storage facility was adapted to the actual operating conditions by closing the ventilation flaps on the roof. This measure successfully prevented further condensate related damage to the cask surface.

7.1.3. Dual purpose casks in the United States of America

The AMPs described in the following Sections are being implemented in accordance with the guidance described in the national abstract in Appendix I.7.

7.1.3.1. Experience with bolted casks in the United States of America.

Because over 90 % of the dry stored inventory in the USA is stored in welded canister systems, there is relatively little ageing management experience involving bolted cask systems; the extensive USA experience with welded canister systems is discussed in Section 7.2. With respect to bolted cask systems, one notable example has been observed and is documented in U.S. Nuclear Regulatory Commission (U.S. NRC) Information Notice 2013–7 [42]. In this instance, a TN–68 cask that had been in service at the Peach Bottom site for 10 years was found to be experiencing a leak rate in excess of technical specification limits. Upon return of the cask to the SFP, it was determined that the outer portion of the cask lid seal had corroded. As the inner portion of the lid seal was still intact, the cask's primary confinement was not compromised. The seal leakage was determined to have been caused by corrosion affecting the outer portion of seal caused by water ingress through the access plate in the protected cover. Corrective actions, including improvements to the access plate design, were taken and additional instances of this occurrence have not been observed.

Two significant research and development (R&D) projects to ensure that ageing effects do not result in the degradation of spent fuel in storage have been undertaken in the USA [24, 26]. The primary concern in this regard is embrittlement of the cladding due to hydride reorientation as the fuel cools. The first of these projects was completed in 2001 with joint sponsorship of the U.S. NRC, Electric Power Research Institute (EPRI), and DOE, and involved the unloading and examination of fuel having burnup less than 45 GWD/tU that had been in storage for 14 years. The second, which was initiated in 2017, involved higher burnup fuel. In the latter project, a DSS was outfitted with monitoring equipment and placed into storage at the North Anna site in Virginia. Considerable data have already been collected from this DSS and sister rods extracted from the same assemblies that were loaded into it. The plan is for this DSS to be shipped to a Department of Energy (DOE) site where fuel can be unloaded for examination similar to that conducted in the first project.

The first project used a DSS initially loaded at the Surry site in Virginia that was transported to the Idaho Nuclear Engineering and Environmental Laboratory (INEEL, now simply known as the Idaho National Laboratory), where it was unloaded and detailed examinations of the fuel were conducted. The project goal was to provide confirmatory data for allowing licensees to continue dry storage beyond 20 years, as well as for U.S. NRC staff to use in their technical reviews.

At INEEL, assemblies from the Surry DSS were easily removed from the inner storage basket for inspection, despite their snug fit. The assemblies were in good condition, with no unusual features. Oxide build up on the cladding of the limited number of rods (12) that were examined was similar in depth and appearance to that observed on wet stored fuel at the same burnup. The rods exhibited typical creep down of the cladding and evidence of some rod spacer grid mechanical interaction. The fission gas releases of 0.4–1.1 % in four rods that were analyzed were well within the range measured on as irradiated rods at this burnup. That is, there was no increase in fission gas release due to storage. Finally, microscopy revealed that there had been no significant radial reorientation of hydride precipitates in the Zry–4 cladding despite this high temperature operation [43].

The INEEL project was successful in building a high degree of confidence that spent fuel would retain its integrity in dry storage. The results derived from this work were a key aspect of U.S. NRC's technical basis for extending license terms from 20 years to 40 years in 2011.

More recently, EPRI, DOE, Dominion Power, and Orano have partnered on a second project, known as the 'High Burnup Dry Storage Cask Research and Development Project', to investigate the integrity of higher burnup fuel (greater than 45 GWD/tU). This time, improvements in technology have enabled online data gathering capability to be included, and rods extracted from the stored assemblies pre–trial are being examined to provide data for comparison to the data that will be gathered when the DSS is unloaded. As stated in the project test plan, the objective of this project is "to provide confirmatory data for model validation and potential improvement, provide input to future SNF dry storage cask design, support license renewals and new licenses for Independent Spent Fuel Storage Installations (ISFSIs), and support transportation licensing for higher burnup SNF" [44].

The HBRDP utilizes a TN–32 bolted lid cask that was loaded in November 2017 with intact, high burnup SNF with four different kinds of cladding taken from the spent fuel pool at Dominion Virginia Power's North Anna Power Station (North Anna): standard Zry–4, low tin Zry–4, Zirlo, and M5 [25]. The lid of the TN–32 was modified to be able to insert temperature probes inside the cask and into the guide tubes of the spent fuel assemblies at various axial and radial locations. Cask cavity gas samples were obtained during and after the drying process. Prior to loading, fifteen individual rods, referred to as sister rods, were extracted from the assemblies to be loaded. These rods have undergone detailed examination at Oak Ridge National Laboratory to provide a basis for comparison between pre– and post–storage conditions as well as to gather additional data on the overall condition of higher burnup fuel.

Currently, the HBDRP cask remains in storage at the North Anna site. It has already yielded a wealth of data relevant to the integrity of higher burnup fuel in storage. Gas samples indicated that there was no damage to the fuel during the drying process (which is the time of greatest risk), and thermocouple data indicate that existing thermal models are highly conservative. Testing of the sister rods has also provided valuable information on high burnup fuel cladding, which can be used to inform modelling of its performance during storage. This project is continuing to build upon the solid technical basis for long term storage.

7.2. WELDED CANISTER SYSTEMS

There is a huge number of welded canister systems in service, in the USA alone more than 4000 systems are in service [45]. As shown in Fig. 4 welded canister systems are currently used to store about 52 % of the global inventory in dry storage systems [6].

7.2.1. Welded canister systems in the United States of America

A major share of the above mentioned 52 % of the global inventory stored in welded canister systems are in operation in the USA [6].

7.2.1.1. Programmes intended to address chloride induced stress corrosion cracking

In 2011, the Nuclear Energy Institute (NEI), EPRI, and U.S. NRC initiated a programme, known as a Regulatory Issue Resolution Protocol (RIRP), to gather data to determine the minimum conditions (environmental and cask), and the associated timescales, necessary for potential initiation of SCC in stainless steel dry SNF storage canisters deployed at ISFSIs located in chloride containing atmospheres [46, 47]. Concerns had been raised regarding the susceptibility of austenitic stainless steels (304, 304L and 316L), used for the confinement boundary in SNF storage, to SCC when exposed to a chloride containing atmosphere. It was believed that it was theoretically possible for fog and spray aerosols from saltwater bodies to contain airborne chlorides that might deposit on the canister surfaces; this had the potential to lead to CISCC. Degradation caused by this phenomenon may then impact the ability of the confinement boundary to maintain its safety function over an extended operating period. ClSCC, if present, may also have an impact on the future transportability of the canister (if the system or component is dual purpose certified, and the canister is used as a second watertight barrier for moderator exclusion during transport).

Over 5 years, several research and development activities were conducted under the collaborative framework of the RIRP. This effort established the foundation for an inspection, monitoring, mitigation, and repair regime that has now been extensively deployed to ensure long term DSS integrity. One of the most notable activities completed during this time was EPRI's Susceptibility Assessment [48]. This assessment greatly improved understanding of the timescales under which SCC could occur. It also enabled industry to understand which of the varied environments in which DSSs were located were more or less likely to experience SCC.

In total, the work products generated under RIRP N–10–01 resulted in a significant advancement in the understanding of the metallurgical and environmental factors necessary for CISCC. The advancement in understanding the technical issue was instrumental in developing susceptibility assessment criteria and ageing management recommendations. Key work products developed to address technical information needs included the following:

- Testing to determine the effects of metallurgical conditions resulting from typical canister fabrication processes, and the effects of environmental conditions including temperature, RH, salt composition, and surface chloride concentration;
- Evaluation of the potential for CISCC of welded stainless steel canisters using operational experience, CISCC test data, and environmental data;
- Development of canister CISCC susceptibility assessment criteria based on system design and materials, canister loading parameters, site specific environmental conditions, and time in service;

— Ageing management guidance and an example AMP for localized corrosion and CISCC of welded stainless steel canisters.

Following closure of the RIRP, work to address SCC transitioned from a research and development phase to a technology deployment phase. Section 8.2 describes experience in USA in applying the technologies that were developed through the R&D effort.

7.2.1.2. Programmes intended to address concrete degradation

AMPs, as described in NEI 14–03 [49], also include provisions for ensuring the integrity of concrete structures and ISFSI pads. These inspections are discussed in Section 8, and examples of findings are discussed in U.S. NRC Information Notice 2013–07 [42], the Maine Yankee inspection experience discussed in Section 7.2.1.3. (a), and the Ageing Management INPO Database (AMID) (see Section 7.2.2.).

7.2.1.3. Examples from inspections conducted at different sites

As illustrated with Fig. 22 dozens of inspections of DSS exterior surfaces have been undertaken using the technologies described in 8.2, and along with the projects described in 8.5, give a high degree of confidence in the integrity of the fuel inside these systems.

FIG. 22. Canister exterior surface inspection (courtesy of 3Y).

This Section focuses on two sites that have been leaders in deploying the latest inspection technologies and a third that was among the first to apply the results of the research projects to ensure the long term integrity of spent fuel inside the DSSs.

(a) Maine Yankee cask pre–application inspections

The Maine Yankee NPP completed decommissioning in October 2005. Most of the site has been restored to greenfield conditions, with the only remaining nuclear being the standalone ISFSI, in which 60 DSSs of the NAC–UMS 24 design are stored. Initial licenses for these DSSs were due to expire in either April or November of 2020. Prior to applying for license renewal, Maine Yankee conducted pre–application inspections and submitted the results along with its renewal application. These inspections were completed in 2018 [50].

The Maine Yankee inspection plan was guided by the results of the EPRI Susceptibility Assessment [50], which informed the selection of lead casks to be the first inspected. The inspections were performed by the DSS supplier, NAC International (NAC), and conducted according to Ageing Management Plans written to address the following ageing issues:

- Localized corrosion and stress corrosion cracking of welded stainless steel;
- Internal vertical concrete cask metallic components;
- External vertical concrete cask metallic components;
- Reinforced concrete structures.

To facilitate the inspection, NAC removed the shield plug lid from the top of the vertical concrete cask using a large crane. Once this lid was removed, personnel were able to inspect the closure weld at the top of the cask. To inspect the remaining canister surfaces, a remotely operated vehicle (ROV) was introduced into the annulus between the concrete cask and the canister. The ROV was equipped with a borescope camera to conduct the inspections. The ROV was capable of 90–degree turns to navigate its way through the inlet and outlet vents and was equipped with magnetic wheels to make vertical runs on carbon steel surfaces inside the annulus. Temperature and radiation probes were also sent into the annulus using the ROV.

Overall, inspection coverage of the welded stainless steel canisters ranged between 85% and 90%. No areas of concern or evidence of SCC were found. One area of paint deterioration and localized corrosion on the vertical concrete cask liner was found and is being addressed in accordance with the site's corrective action programme. Interestingly, the canisters appeared to be dry even though they were inspected on very humid summer days. Salt concentrations ranged from 2.9 to 11.4 µg/cm. The inspection results confirmed that the DSSs were in very good condition, with no significant degradation to SSCs that are important to safety.

(b) San Onofre nuclear generating station inspection and maintenance programme [50]

The San Onofre nuclear generating station (SONGS) permanently ceased operation in 2013. The plant is in the process of being actively decommissioned. All spent fuel was transferred from the spent fuel pool to DSSs by August of 2020. Three types of DSS designs are in use at SONGS: the NUHOMS 24PT1, NUHOMS 24PT4, and HI–STORM FW/MPC–37. There are 50 canisters stored in DSSs of the NUHOMS design, along with 73 of the HI–STORM design. The two NUHOMS designs are licensed under a common Certificate of Compliance (CoC), for which a renewal application was submitted to U.S. NRC in May of 2019. The HI–STORM design is a relatively new system that was first loaded at SONGS after the plant ceased operations and is not due for license renewal until 2035.

U.S. NRC requirements for lead canister inspections at SONGS are the same as those that were followed at Maine Yankee. However, SONGS is also regulated by the California Coastal Commission (CCC), a State of California entity. The CCC required additional assurance of long term DSS safety before it would approve a permit for SONGS to begin actively decommissioning the plant, owing to concerns about a spent fuel pool no longer being available if something were to go wrong with the DSSs over time. The process that was developed is the most significant demonstration of DSS ageing management technology in USA history.

The SONGS inspection and maintenance programme (IMP) includes a full suite of inspection, test canister, and repair demonstration technology deployment. In addition to inspections on DSSs loaded with spent fuel, the IMP includes a test canister programme in which canisters identical to those storing spent fuel are outfitted with an electric heat source to simulate the heat load associated with spent fuel. With no radiological constraints, personnel can more easily inspect the test canister, use tailored temperature profiles to serve as a leading indicator of loaded canister performance, and use the canister as a test bed for refinements in inspection and repair tooling. The chronology of the SONGS IMP inspection programmes is summarized in Table 9.

TABLE 9. TIMELINE OF CONDUCTED/PLANNED SONGS CANISTER INSPECTIONS

Date	Items Inspected	Description
2019	Eight MPCs	Eight MPCs chosen in response to a 2018 canister downloading issue were inspected robotically and verified to be acceptable for continued use.
By 30/04/22	Test Canister	The test canister was inspected robotically and verified to be acceptable for continued use.
By 30/04/24	Test Canister, Two MPCs	Will be inspected remotely using robotic technology.
By 30/04/27	Test Canister	Will be inspected remotely using robotic technology.
By 30/04/29	Test Canister, Two MPCs	Will be inspected remotely using robotic technology.
By 30/04/32	Test Canister	Will be inspected remotely using robotic technology.
By 30/04/34	Test Canister, Two MPCs	Will be inspected remotely using robotic technology.

The SONGS IMP goes further than an inspection programme. The CCC also required SONGS to demonstrate that it had the capability to repair any degradation of a multipurpose canister (MPC) that may be detected in inspections. In August 2019, SONGS completed the final qualification test of a process utilising a metallic overlay (also known as ‘cold spray’) as a method to repair a potential defect on a spent fuel canister in storage. To demonstrate this capability, a special robot applied a nickel coating to a replica of a 16 mm (5/8 inch) thick canister wall. This replica was part of a full-scale demonstration of a canister in its storage position.

After reviewing the SONGS IMP and receiving substantial public input, the CCC approved the permit for plant decommissioning. As a result, the decommissioning of SONGS is now well underway.

(c) Approach to ensuring the integrity of fuel in long term storage

NEI 14–03 [49] introduced the concept of tollgates to be included as part of an operations based AMP. The tollgate concept provides a structured way for licensees to pause and formally assess aggregated feedback at specific points in time during the license renewal period. Tollgates are established in the renewed specific ISFSI license or CoC and are implemented by licensees to formally evaluate the aggregate feedback at future points in time and perform a safety assessment that confirms the safe storage of SNF and other contents. The impact of the aggregate feedback can then be assessed by the licensee as it pertains to components at that site's ISFSI, and corrective actions taken when warranted.

The Prairie Island NPP in Minnesota was the first to have applied this concept to capture the results of the HBDRP project, described in Section 7.1.3.2, in its renewed license to provide long term assurance of safety. Appendix A to the plant's license renewal application commits to three specific future tollgates, as follows:

(i) 2028 — Prairie Island will evaluate data obtained from the loading and initial period of operation at the HBDRP cask;
(ii) 2038 —Prairie Island will evaluate data obtained from the destructive and non–destructive examination of fuel unloaded from the HBDRP cask;
(iii) 2048 —Prairie Island will evaluate any additional available new data.

The information obtained at each tollgate will be shared with U.S. NRC and relevant stakeholders near the plant to confirm the ability of high burnup assemblies to continue to perform as expected throughout the extended storage period. If, by 2033, it becomes apparent that a plan to unload the HBDRP cask is not in place, Prairie Island will be required to submit a license amendment application describing alternate means of obtaining this confirmation.

This application of the tollgate concept amplifies the existing practice of continual evaluation of industry wide dry cask storage operational experience for impacts on a given licensee's dry cask storage AMP. This concept is also being adopted by other USA NPPs in accordance with NEI 14–03, particularly with respect to the integrity of high burnup fuel in storage.

7.2.2. The ageing management Institute of Nuclear Power Operators database

Nuclear Energy Insitute 14–03 [49] tasked the Institute of Nuclear Power Operations (INPO) with establishing a clearinghouse of ageing related information relevant to dry cask storage SSCs. Its development was facilitated through agreements between INPO and the 10 CFR 72 CoC holders. AMID has since been implemented through separate agreements between the CoC holders and ISFSI licensees (including ISFSI licensees for shut down plants) to provide a centralized location where the ISFSI licensees and CoC holders can store, access, search and create reports. Periodic reports are provided to the U.S. NRC upon request. Unlike other INPO operating experience sharing tools, AMID contains positive information about inspection results that simply confirmed DSS integrity along with lessons learned that might require industry wide corrective action. A Users' Guide, for this database is provided by NEI 16–10 [51].

Nuclear Energy Insitute 16–10 [51] provides guidance for personnel entering, approving, and revising records in AMID and describes how users of this information can search and extract information from AMID that is relevant for use in performing tollgate assessments as defined in NEI 14–03 [49]. The ISFSI AMID is not intended to include records unrelated to ageing management, such as operating experience involving canister fabrication, fuel loading, preparation for storage, or deployment to the ISFSI.

Thus far, thirty three inspection reports have been filed in the AMID database. These can be categorized by topical area as follows (number of inspections):

– AMP–required inspections (Stainless Steel Canister, nine reports);
– Proactive inspections for technology development and demonstration (eight reports);
– Investigation of potential or observed concrete degradation (seven reports);
– Routine inspections required by licensee procedure or to provide state/local confidence (three reports);
– Investigation of potential or observed coating degradation (bare fuel DSS) (three reports);
– Investigation of observed corrosion or discoloration (DPC DSS, three reports).

All of these inspections confirmed the integrity of the DSSs. No significant repair actions have been taken to date. Two examples, first an investigation of an observed condition and second an AMP inspection, are discussed below.

7.2.2.1. Observed condition

In September of 2016, evidence of corrosion was discovered on the carbon steel outer shell of a Holtec HI–STORM 100 overpack at Plant Hatch. The corrosion had occurred behind a radiation placard that had fallen off. The corrosion area was measured and characterized. It was determined to be crevice corrosion associated with damage to the coating, which compromised corrosion protection. The corrosion depth was within tolerance. This condition was remediated by sandblasting away rust and repainting the cask. Every other cask at the site was inspected and no similar indications were found.

7.2.2.2. Ageing management programme inspection

In October 2020, AMP inspections of the standardized NUHOMS HSMs and stainless steel canisters were performed at the Susquehanna NPP. The HSMs were examined using direct visual inspection and the canisters were examined by remote visual inspection using Suction/Deployment Crawlers and a GE camera. Chloride deposits on the canister were sampled using the suction crawler. It was determined that observed ageing effects did not challenge the intended safety function of the HSMs. Chloride load on the canister was determined to be low and the risk of CISCC was therefore also determined to be low. No evidence of CISCC was observed.

7.3. VERTICAL STORAGE SYSTEMS

The operating experience related to NPP spent fuel dry storage in Argentina is described. There are two dry storage facilities in Argentina. The first one, ASECQ–CNE is located at the site of the Embalse NPP and has operated since the early nineties. The other, ASECQ–CNA–I, is currently under construction at the site of the Atucha–I NPP.

7.3.1. Dry storage at ASECQ–CNE Argentina

This installation consists of an array of vertical storage units made of concrete and lined with carbon steel. The 0.5 m long CANDU spent fuel bundles are placed into cylindrical metallic containers which are placed one on top of the other up to the top of the silo, which can hold up to nine of these canisters. Fig. 23 shows a view of the site (a) and construction details (b). Fig. 24 depicts a metallic canister that holds spent fuel bundles.

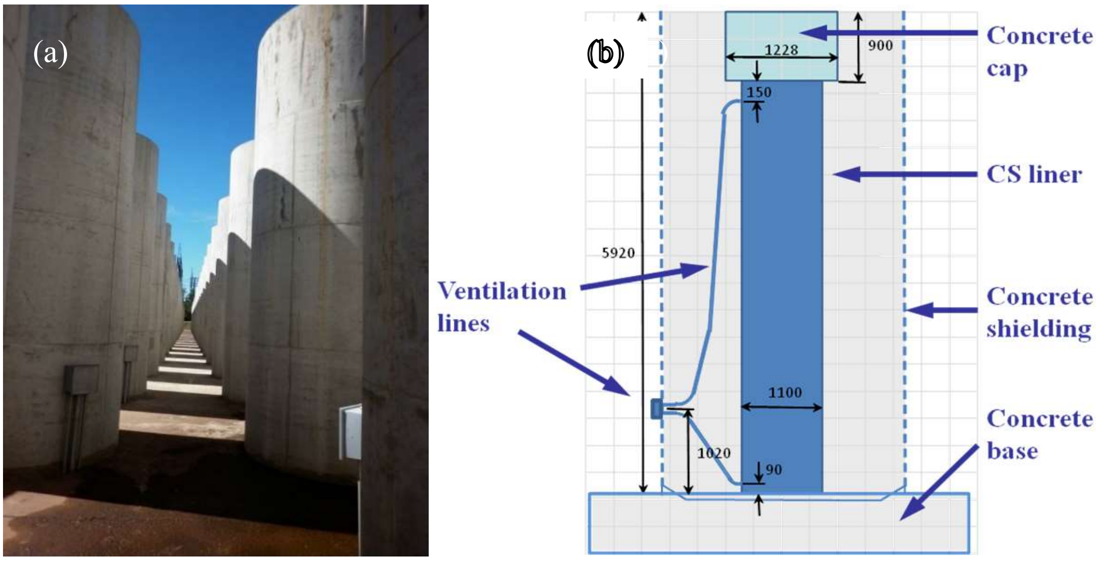


FIG. 23. Dry storage site at Embalse NPP. (a) View of the array. (b) Diagram of a vertical storage unit (units in mm). (courtesy of CNEA).

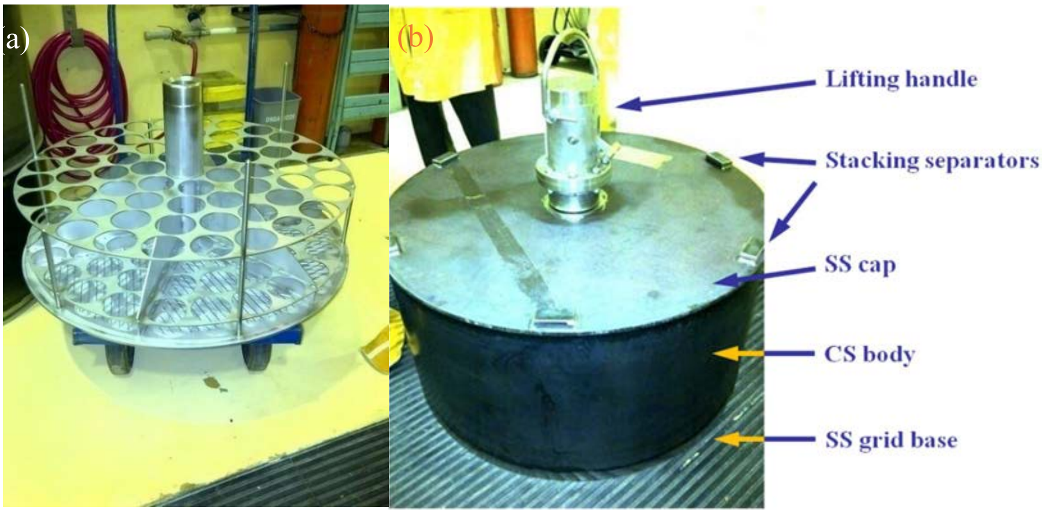


FIG. 24. (a) Stainless steel grid to hold spent fuel bundles from CNE. (b) Closed canister (courtesy of CNEA).

ASECQ–CNE was intended to hold SNF for a period not longer than a few decades; but since no decision has been made to date regarding their final disposition, an extended storage time will be needed. Considering that CNE has undergone a Life Extension process, this facility is expected to be active for a period close to a century. Hence, a study of possible materials degradation processes that could take place in this installation was necessary to evaluate its capability to withstand long term operation provide continued safe SNF storage.

7.3.1.1. *Programmes intended to address canister corrosion*

To correctly assess the degradation state of the canisters, it is crucial to know the history of water ingress events and their duration. Water can be collected in the silos during construction; for this reason, they are drained and the interiors are fully dried by hand before SNF loading. The loading procedure takes approximately a week and it is possible that water ingress can occur during this period if proper precautions are not taken. Initially, the mouth of the vertical unit was covered with a hat like protection if needed, but this device failed in some cases to prevent rainwater ingress. For this reason, the procedure was changed, and the silo concrete cork is placed into position, with a rubber ring inserted in between the cap and the silo, hermetically closing it. When the loading process is finished, the rubber seal is taken off and the steel cork tab is welded to the inner silo liner tab, making the unit air tight.

Occasionally, the presence of water has been verified in various units; the procedure applied in these cases is to pump it out to the lowest possible level, leaving the rest to dry by evaporation. Later determinations indicate that the inner air in these particular silos stays within the desirable humidity range.

Another issue that arose in the past was that the inner silos' air, which is periodically replaced with dry air, was found to have become humid after some time. This problem was solved by increasing the purging time from 24 hours to 15 days.

The following activities have been carried out in relation to the study of carbon steel corrosion in the ASECQ–CNE Dry Storage Facility:

(a) Calculation of general corrosion

Using uniform corrosion rates of carbon steel in various atmospheric environments [52], a 50 year corrosion value was calculated by extrapolating a 4 year slope, representing a conservative approach. Considering the worst case (humid, saline environment), the average penetration would reach a value of 2.1 mm, which constitutes a safe situation for a 10 mm (3/8”) thick wall.

(b) Corrosion tests in canister material

If for any reason water ingress to the interior of the silos would occur the conditions would be different. In order to allow a kind of benchmark, corrosion tests were performed on samples of the same material as the canisters, including welds (designated WS). Carbon steel samples (designated CS) were also tested for use as references. All details pertaining to the procedure followed in these tests can be found in Appendix III. The average corrosion rates obtained for CS and WS samples were 120±15 µm/yr and 154±33 µm/yr, respectively. This difference would indicate a galvanic acceleration, although it is computed as not relevant for an analysis of variance calculation with a significance level of 0.5. However, the results of cross section metallographic analysis indicated massive, localized attack, with penetration as high as 850 µm (see Fig. 25). This value is equivalent to a corrosion rate of 3.2 mm/yr, a high enough rate to result in canister perforation in three years.

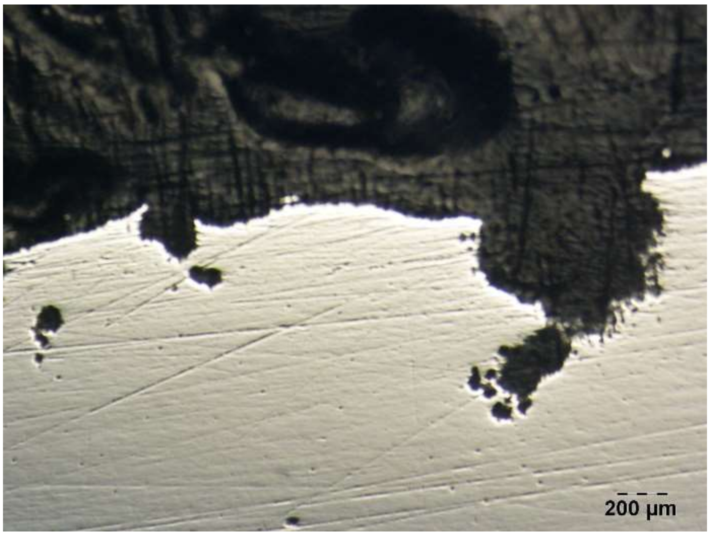


FIG. 25. Penetrating pit in corrosion tests on WS. Scale length: 200 µm (courtesy of CNEA).

(c) Mock–up canisters

The mock–up canisters, see Fig. 26, are canisters identical to those used for storage and have been left exposed to the atmosphere since the beginning of the facility operation. Although they look quite oxidized after 30 years, they bear no signs of penetrating corrosion.

FIG. 26. Mock–up canisters at the ASECQ–CNE facility (courtesy of CNEA).

(d) Mock–up silo

An endoscope was introduced into a mock–up silo through the ventilation lines to get a close view of the canisters. This silo belongs to the first batch put into operation; it is filled with canisters but is not loaded with SNF. The first step was to introduce a testing cable to check for irregularities and obstructions (Fig. 27a). Then the endoscope was inserted up to the point where an image of the canister could be obtained from the ending point of the inferior ventilation line (Fig. 27b).

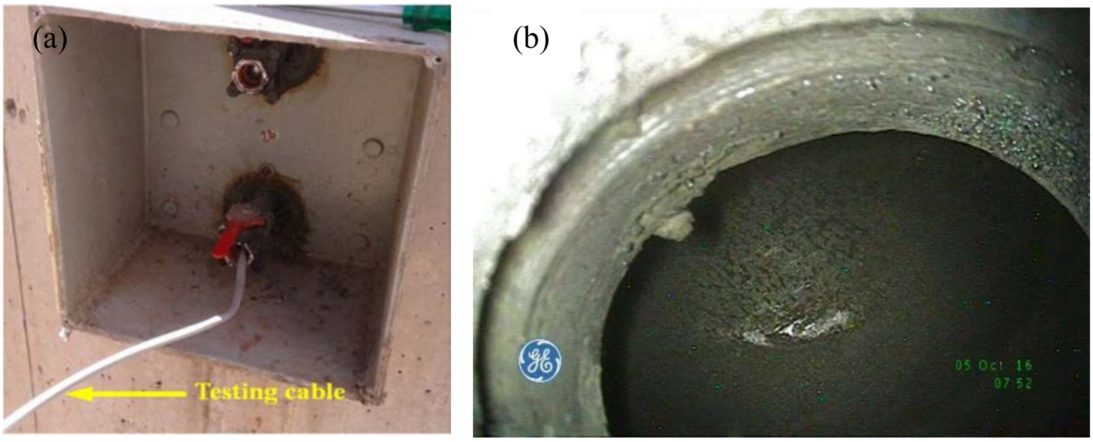


FIG. 27. (a): Introducing a test cable into the silo. (b): View of the canister surface (courtesy of CNEA).

As the photographs show, there are no relevant corrosion associated features, only the expected normal oxidation. However, the welded region could not be reached because of the endoscope geometry. Alternatives for a more comprehensive visual inspection are being studied. The mock–up silo is equipped with samples of the various materials used in the facility, like pieces of silo liner steel and canister steel including welded coupons, which are periodically extracted to perform visual inspection and weight loss determinations.

7.3.1.2. Ongoing operational measures

As part of facility operations, the following measures are performed in order to ensure facility safety and compliance with regulatory requirements.

(a) Regular air maintenance

The inner silo air is periodically renovated by vacuum pumping it through filters and replacing it with dry air. The air taken from inside is analyzed biannually to measure humidity content and radioactivity. All units were treated at the beginning of operation, however, because of the aforementioned extension in the purging period (from 24 hours to 15 days), only 20 units are renovated per campaign, which means that 5 years are required to complete the operation across the entire installation.

Any time that the RH content is found to be high after the air maintenance process, the purging operation is repeated until the values decrease to acceptable levels.

The results of these determinations indicate that RH, measured at 25 °C in the lab, is kept below 30%. The value in the interior is even lower than that, given that the internal temperature in the storage unit is above 50 °C. As for the radioactivity, it is always found to be in the range of 10^5 to 10^8 Bq; the sum of all the silos' activity amounts to 0.43 GBq, which is only 0.2% of the permitted release amount for the whole site, including the nuclear reactor. No ^{137}Cs activity has been detected.

(b) Surveillance of cracks in the concrete structures

The external surface of the units is surveyed to detect eventual cracking in accordance with national requirements, as specified in the ageing management manual for CNE NPP. Some cracks were found to be about 2.54 cm long, getting close to the reinforcement bars (rebars). They were filled with cement to avoid further deterioration.

7.3.1.3. Conclusions relative to ASECQ–CNE

Results of the corrosion calculation show that in the event of water ingress, the canisters will be penetrated by corrosion in a matter of a few years. The fact that both humidity and radioactivity levels were found to be within safe margins indicates that any periods of water ingress into the interior of the silos have been insufficient to produce such damage. The operational experience has led to developing methods to ensure safe operation, including hand drying the silo interior before spent fuel storage, improving the protection during canister loading, and drying the air much more effectively. These conclusions are supported by the satisfactory result of the inspections performed on mock–up canisters and on coupons kept in the mock–up silo.

7.3.2. Dry storage at ASECQ–CNA–I

The ASECQ–CNA–I dry storage facility in Argentina is under construction. The storage facility consists of a dry concrete pool inside which the 6 m long pressurized heavy water reactor SFEs can be accommodated in 112 vertically hanging silo units containing 18 SFEs each as depicted in Fig. 28. This facility is designed for a storage period of at least 50 years.

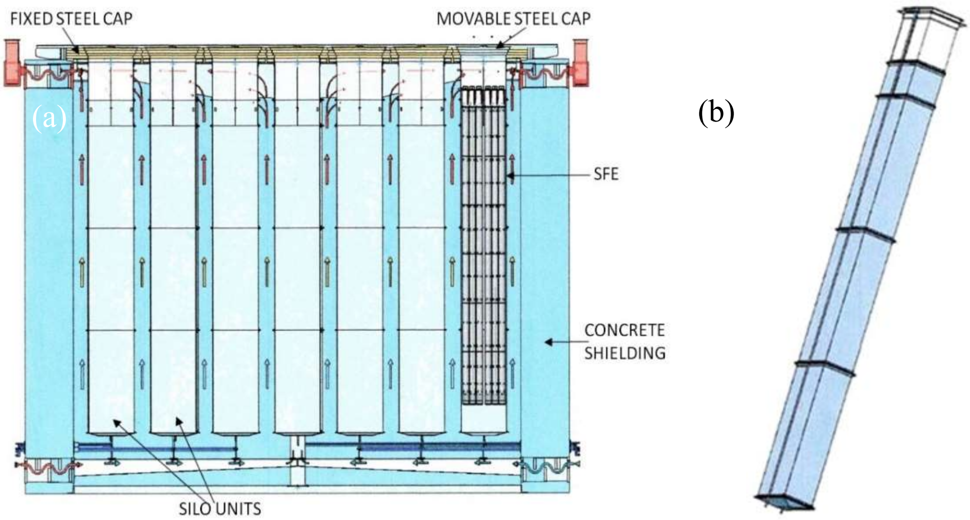


FIG. 28. Cross section of the silo unit array showing hanging SFEs (a), Single silo unit (b). (Courtesy of CNEA).

7.3.2.1. Programmes intended to address rebar corrosion

To ensure that the facility's structural components will not suffer any material degradation, monitoring of the concrete structures and supervision of silo unit fabrication procedures is already ongoing.

Rebar corrosion is the primary issue of concern with respect to the durability of concrete structures and can be assessed using corrosion probes. Some of these probes have been developed at the CNEA Corrosion Laboratory [17] and have been used for various purposes, such as monitoring of new and pre–existing structures and selection of the best cement candidates. More technical details related to these devices can be found in Section 8.4, of the present report. Two activities have been carried out at the ASECQ–CNA–I, monitoring of a silo unit mock–up and monitoring of the ASECQ–CNA–I main structure.

Each sensor contains several electrodes that allow measurement of various parameters that have to remain within their respective safety ranges to ensure proper bar behaviour, as stated by specific standards and other recommendations [53-55].

(a) Monitoring of a silo unit mock–up

Figure 29 depicts the reading operation on the silo unit mock–up concrete shielding, using probes that are embedded into the concrete.

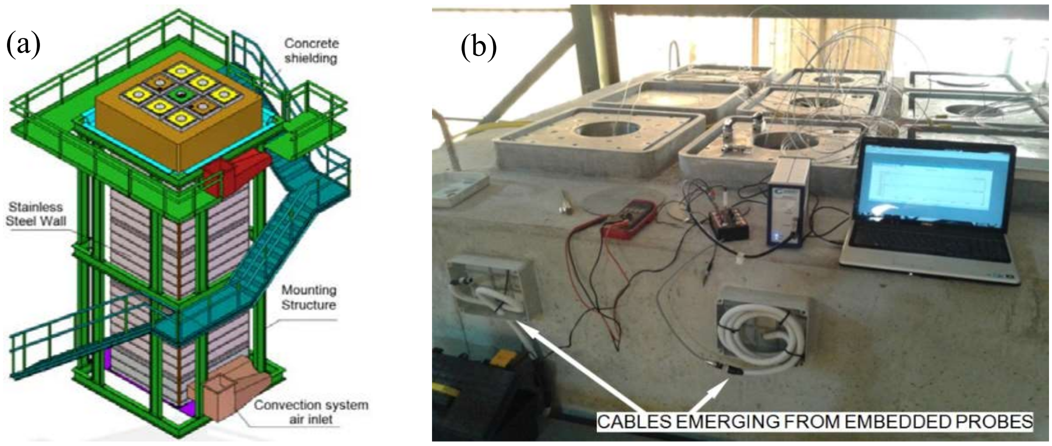


FIG. 29. Diagram of the silo unit mock–up (a), Monitoring operation on the concrete shielding (b). (Courtesy of CNEA).

According to the data collected for 200 days, all measured parameters have been evolving towards values in which the rebars will not suffer corrosion, thus ensuring the structural safety of the concrete structure. All the details pertaining to the procedure followed in these tests can be found in Appendix III.

(b) Monitoring of the ASECQ–I–I main structure

The locations of the eight inserted probes can be seen in Fig. 30. Four probes are located on the internal side of the shielding wall and four on the external side.

They are distributed at two different heights in order to monitor the effect of conditions at different distances from the ground. The measured parameters are corrosion potential (E_{corr}), corrosion rate, concrete resistivity, and oxygen availability (as Oxygen Reduction Current i_{O2}). Also, corrosion potential values measured directly on the rebars through an electrical connection are measured; all the details pertaining to the procedure followed in these tests are given in the reports in Appendix III.

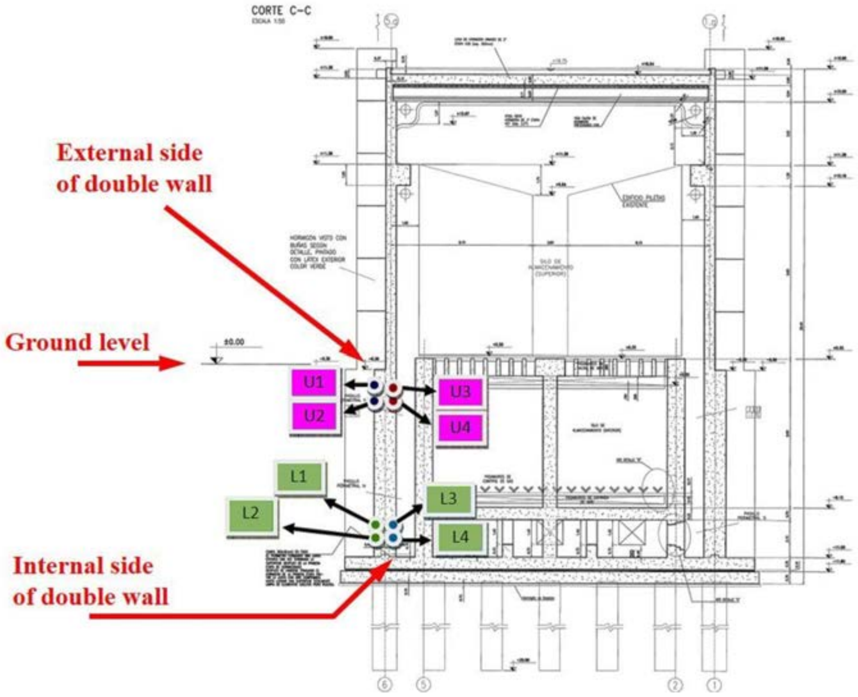


FIG. 30. Diagram of the ASECQ–CNA–I facility, showing locations of inserted probes. (Courtesy of CNEA).

The results for the main ASECQ–CNA–I structure can be seen in Fig. 31.

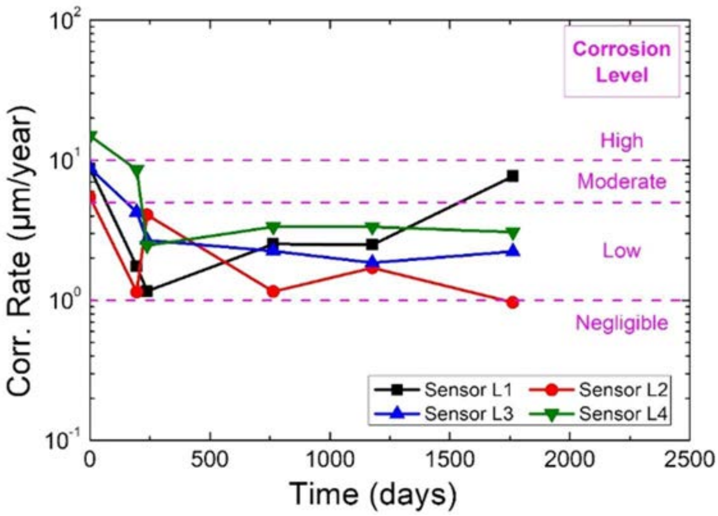


FIG. 31. Evolution of corrosion rate in the main ASEC–CNA–I structure (courtesy of CNEA).

7.3.2.2. *Conclusions relative to ASECQ–CNA–I*

The general trends point to acceptable behaviour, indicating acceptable performance of the structure from a corrosion perspective. One interesting aspect is that the corrosion potentials measured directly in the rebar are essentially coincident with those measured with the probes, thus validating use of the probes for monitoring the rebar. It is expected that, once construction is finished, these determinations will stabilize and the trends will be in the direction of reducing the rebar corrosion rate, with increasing values of corrosion potential and resistivity, whereas the levels of corrosion current and oxygen flux will tend to decrease. The monitoring system is expected to assess the corrosion of the rebars throughout the life of the facility.

8. MONITORING, MITIGATION, AND REPAIR TECHNOLOGIES FOR AGEING MANAGEMENT PROGRAMMES

A key component of an AMP is the ability to evaluate (or inspect) the condition of the safety related SSCs using NDE methods. For the purposes of this report, 'inspection' and 'NDE' may be used interchangeably. The results from an inspection are typically used in one of three ways:

(i) To identify active degradation that exceeds the AMP acceptance criteria and requires corrective action. Inspections may be conducted more frequently and on a larger sample size;
(ii) To identify potential degradation that is below the AMP acceptance criteria and is monitored and trended per the AMP and assessed for mitigating actions. Inspection frequencies and sample sizes may be adjusted as necessary;
(iii) To verify that part or all of the inspected component is free from degradation and will follow the planned inspection frequency.

DSSs are manufactured by numerous international commercial entities and are licensed and available for deployment in many countries worldwide. A single vertical dry storage canister can be more than 6 m (20 ft) tall and 3.3 m (11 ft) in diameter and may weigh more than 163 t (360 000 lb). In the USA alone, there are over 50 different types of DSSs approved by the U.S. NRC for general use. Because of the wide range of existing canister sizes and concrete overpack designs, there are a wide range of potential accessibility issues that could be encountered in designing inspection systems for currently available licensed systems. These accessibility issues include different annulus gaps and widths, entry pathways, welding processes, materials, weld locations, etc.

According to Fig. 4, the DSS used to store about 95 % of the global inventory (excluding storage buildings) fit into one of two categories; bolted cask systems that contain bare fuel, and concrete storage overpacks that contain welded canisters. While both systems can be found in countries worldwide, bolted cask systems are more common outside of the USA and welded canister systems are more common in the USA. A short overview about specific ageing related effects, inspection challenges, mitigation and repair technology developments and operational experience of those two main system classes is given below.

8.1. BOLTED CASK SYSTEMS

Once the cask is delivered to the final user, a set of acceptance tests are performed. These are visual tests focusing on the status of all cask components, especially metallic and elastomeric seals, lids, and the cask body. An example of possible surface damage is shown in Fig 32. (a) and (b).

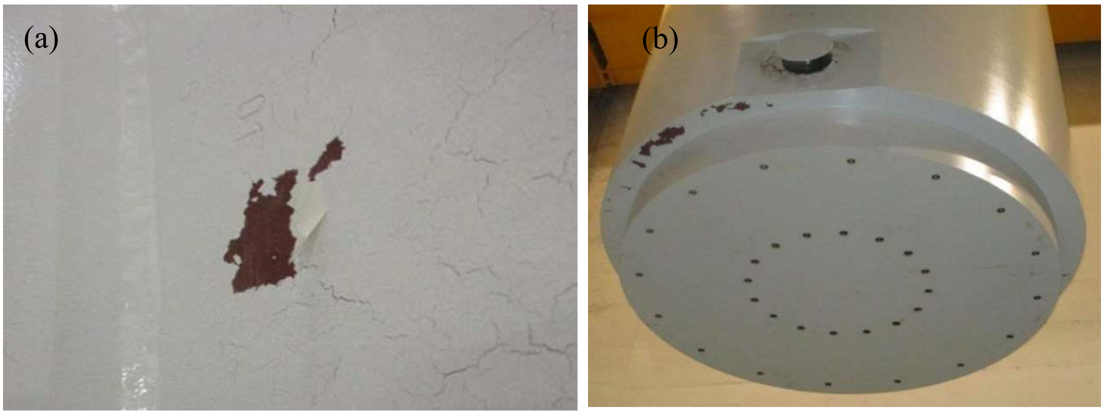


FIG. 32. Examples of defects on the outer protective coating of metal casks (courtesy of ENSA).

During use (transport/storage), DPCs are periodically inspected after a specified number of transports or after several years. These tests range from visual inspections of the cask components, checking tightening torques, and calibration of pressure sensors or switches, to periodically repeated leak tightness tests. Quantitative visual examinations, to allow for trending analyses over time, are typically done for those systems where access to the cask surface is not limited. Casks are also continuously monitored, e.g., for their surface temperature, internal pressure, leak tightness, and γ and n dose rates.

Inspections performed by the regulatory body typically assess for compliance of the transported or stored casks against operational limits and conditions (OLCs) approved by the regulatory body and/or with the conditions of the cask certificate. Storage inspections, therefore, focus on the control of cask types and maximum number of casks in the storage hall, the geometry/positioning of the casks, maximal cask surface temperatures, leak tightness of casks, maximum allowable dose rate on the cask surface at a 2 m distance, average surface contamination, thermal power of stored spent fuel assemblies, operation of the cooling system in the storage hall, condition of the fire protection system, etc.

The principal inspection and maintenance activities to be performed on bolted cask systems located in a dry storage facility are:

- Before carrying out any movement of the cask, a visual inspection is performed on the lifting and handling devices (i.e., trunnions and their bolts). If any crack or considerable damage is observed, the component is replaced. Furthermore, the tightening torque of the bolts are checked if required. In the case of any loss of torque relative to the specified values, appropriate measures are taken (e.g. the bolts are tightened);
- The pressure level of the inter lid space is constantly monitored by the pressure sensor or pressure switch. Pressure sensors measure the pressure in the inter lid space whilst the pressure switches monitor the pressure level of the inter lid space.

Pressure sensors measure the actual value of the pressure between the lids, and an alarm is triggered in the monitoring system if the measured value drops below the lower pressure margin. The pressure sensors need to be calibrated regularly, in accordance with national metrological rules.

Pressure switches, in the event the pressure drops below the switching point due to leakage, open a contact and a corresponding alarm signal is triggered in the monitoring system of the storage facility. The pressure switches are self–monitoring so that internal errors also lead to a corresponding alarm signal. For the pressure switches used in the Czech Republic and Germany [56], a large amount of operating experience exists. With over 1400 pressure switches of this kind installed worldwide, some of which have been in operation for more than 30 years and a total operating time of over 20 000 years, only a very limited number of failures have occurred. The probability of a pressure switch failing is in the region of $<10^{-5}$/year. None of the reported alarm signals to date in the Czech Republic and German have been related to a leakage of the double lid closure system. All alarms have been assocuated with errors/failures in the self–monitoring system..

Depending on national regulations, in the event of a small loss of pressure with respect to the reference value, it is possible to inject additional inert gas. In Germany's case, each triggered pressure switch alarm (even due to a small loss of pressure) requires the pressure switch to be replaced and the inter lid space to be refilled with the required amount of inert gas (helium). In the event of a gross loss inin pressure, the performance of the monitoring system is first checked for possible malfunction. If this possibility is excluded, corrective measures/procedures for managing a leaking closure system are activated.

Periodic inspection of the pressure sensors are often specified in maintenance schedules for at least every three years and always when a malfunction is suspected. In Germany, no periodic inspections are necessary for the pressure switch due to itself–monitoring design.

In addition to the cask properties, transport inspections evaluate compliance with transport regulations such as SSR–6, ADR, RID/COTIF, IMDG Code, and others.

8.1.1. Ageing related effects

Metallic cask systems are typically constructed using cast iron or carbon steel materials that are not susceptible to CISCC but that are susceptible to general corrosion. Concrete cask based systems, however, differ by using a sandwich cask body composed of an outer and inner steel liner with a concrete layer in between. The remaining components of this DSS are identical to metallic bolted cask systems.

Metallic and concrete cask systems are typically coated with a corrosion resistant material, and the inner cavity of the cask needs to be properly dried after loading with spent fuel. The cask manufacturers define the maximum value of residual moisture after cask drying to minimize the degradation of the spent fuel assemblies due to oxidation. Furthermore, the inner space of the cask is filled with nitrogen or inert gas, such as helium.

For cask systems, some contaminants can cause degradation to bolt and seal materials, or to areas with coating damage or delamination's. Cask systems employ monitoring techniques for seals (inner pressure monitors), and the coating can generally be evaluated from a general visual examination of the external surface, which is readily accessible.

As such, there are fewer needs for NDE development for cask systems compared to welded canister systems. However, some of the techniques described in Table 10 in Section 8.2.2 could be applicable in some form to both systems. For instance, ultrasonic testing is used to inspect the trunnions of casks and in case of any needs for surface inspection technologies like eddy current testing could be used.

8.1.2. Inspection challenges

Despite the general inspection challenges for bolted cask and welded canister systems being identical (AMP, evidence of DSS inspection), the regulatory inspections of bolted cask systems are relatively straightforward and less technically challenging. This is because all external components of bolted casks are accessible to visual inspections (see Section 7.1.2). Opportunistic inspections of the bottom may be informative if the cask is lifted for operational reasons and could then be performed by a remote visual examination of the underside of the cask. The inner components, such as the spent fuel basket, seals, and neutron absorbing components, and the spent fuel itself, are protected by a nitrogen or helium atmosphere. Pressure monitoring systems allow for detection of possible leakages (see Section 8.4.1).

Inspection challenges such as elevated temperatures and radiation levels, limited space, or the need to handle heavy loads (casks) are not typical for bolted cask systems. Routine inspections are performed in situ, usually without the need for lifting and handling systems. Depending on national regulatory requirements, it might be necessary to move the cask to a dedicated maintenance area for a more detailed visual inspection every ten years. Some bolted cask systems may also contain welds (e.g., repair lids, welds at cask body), which are easily accessible from outside. These welds can be controlled by commercially available ultrasonic weld control techniques.

Owing to the shielding capability of DPCs and their storage in a hall with personnel access, the casks are accessible at any time during normal operation. As a result, corrosion on the external surfaces would be detectable even without special inspections and as part of the storage facility's operational scope. However, experience to date shows no occurrence of safety relevant damage and findings limited to signs of wear and tear and locally limited corrosion due to paint damage incurred during handling operations.

8.1.3. Operational experience

Operational experience is substantial for the inspection of cask DSSs, as they are one of the oldest technologies employed globally. A typical example of bolted cask ageing management inspections recently occurred in the USA, at the North Anna NPP. The Orano TN–32 dry storage casks were painted white to better facilitate visual inspections and to provide protection from general corrosion. North Anna staff routinely address general corrosion events with periodic visual inspections followed by grinding and repair of corrosion sites using additional paint to restore surface protection. For its TN–32 renewal application, North Anna also conducted a lifting operation and inspected, photographed, and measured corrosion underneath one cask after coating delamination. The North Anna staff reviewed the inspection results and corrosion analysis that showed no safety issues, with plans to trend corrosion of the underside of the cask during future opportunistic inspections.

The ten-year inspections carried out so far reveal a positive picture. The outer surfaces of selected DPCs were subjected to visual inspection, with special attention paid to preservation measures. Again, no safety relevant effects were found. Therefore, damage to the paint layer and to silicone sealings has not been repaired so far as it has no bearing on maintenance of the intended safety functions.

8.2. WELDED CANISTER SYSTEMS

While the previously described casks are one closed system, the welded canister systems consist of a concrete overpack and a canister inside which acts as confinement barrier.

8.2.1. Ageing related effects

Dry storage canisters are subject to different age related effects, depending on the materials involved, the environment to which they are exposed, temperatures of the canisters, and so forth [47]; many of these age related effects have been covered earlier in this publication. Some of these effects can be either beneficial or detrimental to long term storage of DSSs:

- Materials, typically, canisters are constructed from 304L or 316L stainless steels. These materials are very robust against general corrosion but are susceptible to CISCC in marine environments. The concrete overpack faces the same challenges as described in Section 8.3;
- Environment, canister and cask based systems are exposed to a range of environments, as many systems are stored outdoors. Regulations in some countries require any over buildings to be seismically qualified, with the added time, cost, and maintenance involved possibly limiting their use in some countries. DSSs are operated in a range of climates, from colder climates to deserts and marine (coastal) environments; the local environmental conditions influence the potential degradation mechanisms [47]. For example, DSSs in various climates may experience salt deposition on canisters, but from different sources. DSSs in colder climates may be affected by proximity to salted roads in the winter, while DSSs in marine environments may be affected by proximity to breaking ocean waves;
- Temperature, degradation may be prevented, slowed, or accelerated by temperature conditions. Most degradation mechanisms occur more rapidly at higher temperatures; however, CISCC requires temperatures to be below a threshold for water to deliquesce on a surface to deposit chlorides. DSSs will cool over time, lowering the overall corrosion rates, but may start operation above deliquescent thresholds. Some have proposed intentional incremental blockage of vents to maintain higher canister surface temperatures as canisters age to avoid deliquescence as canisters cool over time, but to date, no such approach has been licensed or implemented.

EPRI have performed a number of studies related to DSSs that included:

- A literature review of environmental conditions related to CISCC [47];
- A report on failure modes and effects analysis [57] for a number of potential degradation mechanisms that could impact DSSs;
- A study focused on determining critical flaw sizes for DSSs under a range of conditions [58].

The conclusions of the flaw growth and flaw tolerance report were that canisters are both robust and flaw tolerant, indicating that canisters can withstand comparatively large flaws before there is a threat to canister structural integrity.

Although a structural failure is highly implausible, there is a potential for a flaw (e.g., a crack) to eventually grow through the wall of a canister, creating a pathway for gaseous materials (and potentially very fine particles, if damaged fuel or spalled crud particles are present) inside the canister to slowly escape. After the release of the helium fill gas, oxygen could enter the canister, initiating minor corrosion reactions. This situation could cause oxidation of the fuel cladding (and fuel, if exposed) if temperatures are sufficiently high. Recent studies related to thermal modelling [59] indicate that most fuel is already at or below this level from the time of loading.

Ageing related degradation could eventually result in a through wall flaw in a DSS, resulting in a breach of confinement. Various studies have evaluated the dose consequences from a range of potential scenarios [60–63]. Even when doses are within regulatory limits, such breaches are to be avoided. Previous studies conducted by EPRI have focused on welded canister integrity to evaluate existing margins to degradation [58]. Those studies showed significant margins between the flaw sizes that standard NDE techniques can detect and critical flaw sizes in canisters, meaning that it is highly unlikely that degradation will extend to a point where structural integrity is compromised. NDE inspection techniques and systems have been developed, deployed, and used for ageing management assessments of existing ISFSIs for regulatory decisions.

8.2.2. Inspection challenges

Various inspection and delivery systems are either available or currently under development. Several such systems have been deployed, not only for inspection of dry storage casks and canisters in and around welds and heat affected zones, but also for inspection of the base metal and, if used, surrounding support structures and overpacks (i.e., for the evaluation of concrete/metal interactions). These inspections can detect general/localized corrosion, pitting, and cracking. The results of these inspections have been utilized as a part of AMPs. Results from NDE inspections are typically tracked by a plant's corrective action programme (CAP) or equivalent. Additionally, in the USA, these inspections are collected and tracked nationally in AMID.

Inspection and robotic delivery system development continues, and these systems are being used in support of license renewal efforts to verify the continued safe operation of DSSs. These systems provide the capability to perform various forms of NDE inspections on dry storage canisters, and experience with performing inspections on loaded canisters is continuing to grow.

NDE techniques offer the potential for identifying early stage degradation years before a canister breach may occur. Many different NDE technologies exist and continue to be developed. For the inspection of DSSs, the preference is to employ mature, well characterized NDE technologies that provide a high level of confidence in the results, so that inspection results can be used for making decisions related to DSS ageing management.

A dry storage canister presents many challenges related to performing high quality examinations, including the following:

— Lifts: Performing inspections in a spent fuel pool would provide several options for inspecting DSSs. However, because of the weight and contents of the system, lifting canisters for movement to another location — such as a spent fuel pool — involves nuclear lifting practices and is difficult and expensive to perform, in addition to

increasing risks due to movement of a canister filled with spent fuel. As such, in situ inspection techniques are typically preferred to avoid the need for lifting and handling canisters more than necessary;
- Elevated temperatures: Inspection and delivery systems need to both withstand and operate in an environment with elevated temperatures. High temperatures may impact the ability of some materials and NDE techniques to operate effectively. However, temperatures vary with seasonal changes and slowly decrease over time;
- Elevated radiation levels: The DSS is a high dose environment, so inspection and delivery systems need to withstand highly elevated radiation levels, precluding the use of some materials and NDE techniques. While radiation levels decrease over time, they will remain highly elevated during the period of performance of a DSS;
- Confined space entry and narrow annulus openings: Openings to allow ingress of cooling air are restricted in size for various reasons. Many commercially available inspection and delivery systems are simply too large to enter these spaces, and adaptations or miniaturization is needed to perform inspections;
- Tortuous pathways and internal obstacles: DSSs are designed to protect people and the environment. This means that designs necessarily avoid direct pathways from canisters to the external environment, so inspection tooling and/or delivery systems, including cords and cables, will need to be able to not only enter and exit the DSS, but also maneuver inside the overpack.

Inspection systems have been developed to be able to perform high quality in situ inspections of canisters by entering the space between the canister and its surrounding vertical overpack to perform required inspections, and to be safely retrieved without the need to move or handle the canister or any heavy loads. Many different NDE technologies are available that can potentially help to address inspection needs for DSSs. Many of the current efforts are being coordinated within the EPRI ‘Extended Storage Collaboration Program’ (ESCP) under the NDE Subcommittee [64], which is a volunteer working group. Many organizations are collaborating to help the industry overcome these challenges. Significant progress has been achieved by several organizations related to the inspection of dry storage canisters via development of inspection techniques, robotic delivery systems, and initiation of mobile examination (inspection ring) and repair facilities.

Several NDE techniques were proposed within this group to address the challenges related to inspecting DSSs. Table 10 presents a snapshot of technologies and their capabilities as of 2020.

TABLE 10. MATRIX OF NDE TECHNOLOGIES AVAILABLE AND A QUALITATIVE ASSESSMENT OF THEIR SUITABILITY FOR DSS INSPECTION

NDE Technique	Temperature resistant	Radiation resistant	Small form factor	Sensitive to ODSCC	Suitability	Time to deliver
Visual (VT)	3	3	3	2–3	Yes	Now
Eddy current testing (ECT)	3	3	3	3	Yes	Now (if needed)
Ultrasonic testing (UT)	3	3	3	3	Yes	<1
EMAT / Guided waves (GW)	4	4	3–4	4	Yes	<1
Acoustic emission (AE)	–	–	–	2	Maybe	<3
X ray (RT)	–	3	–	2	Maybe	≥4
Penetrant testing (PT)	2	3	2	3	Unsuited	N/A
Thermography	3	1	3	2	Maybe	<3
Muon imaging	–	–	–	1	Unsuited	≥

Where: ODSCC – Outside diameter stress corrosion cracking;
Technical capability: Not applicable (–); Poor performance (1); Fair perfomance (2); Good performance (3).

While significant progress has been made in relation to inspections, a number of other factors are relevant to the successful deployment of inspection techniques. These include development of mock–ups for use in testing NDE techniques, development of delivery systems (robotics, etc.), and integration of these systems into a single package.

8.2.2.1. Mock–up development for canister based systems

Mock–ups using representative materials, geometries, anticipated flaws, etc., are needed to be able to demonstrate that NDE techniques are capable of detecting the flaws of interest in a representative environment [4]. Various suitable mock–ups have been designed and fabricated by a number of organizations that are available for R&D purposes [2, 15]. These mock–ups have a number of important characteristics:

- Thickness variation to represent the range of canister designs;
- Different weld geometries, including welds that are ground flush with the base metal, partially ground, or underground;
- Flaw variations to represent a range of sizes, shapes, orientations, flaw manufacturing techniques (EDM notches, thermal fatigue flaws, chemically grown SCC in sensitized material, etc.), and locations (in welds, welds into heat affected zones, in heat affected zones, and in base metal);
- Different mock–up geometries, including flat and curved plates as well as large shells.

Utilizing a wide sample set helps to prepare for a range of conditions that could occur in the field and ensure that NDE techniques can detect the anticipated flaws under realistic conditions. Fig. 33 shows details of a crack (a) in a weld (overview in (b)), similar to potential defects that could be encountered during an NDE inspection.

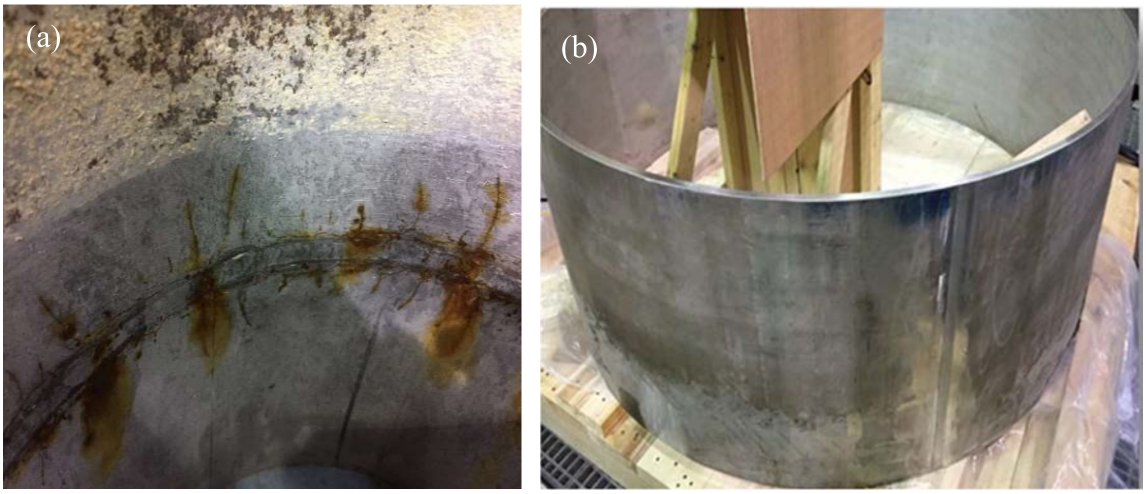


FIG. 33. DSS NDE mock–up containing a crack (a) transverse to the weld direction, overview (b) (courtesy of EPRI).

Because of the unique nature of DSSs, the need to develop high quality inspection tools, the range of defects desired to be detected (including size, shape, orientation, etc.), and the anticipated high temperature inspection conditions, it was decided to develop mock–ups with a QA pedigree. Fig. 34 shows some of the mock–ups, developed by EPRI with a QA pedigree, that are available for industry testing and development efforts [4, 64].

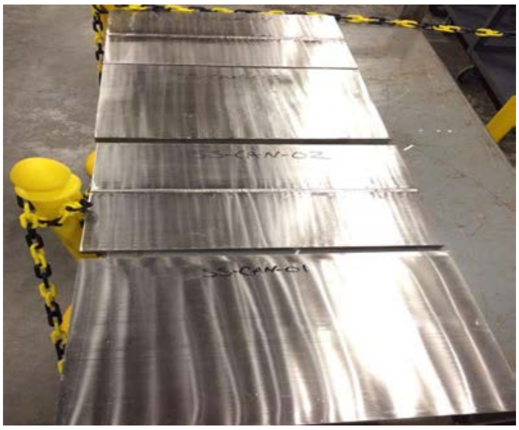


FIG. 34. DSS QA certified NDE mock–ups available for industry use (courtesy of EPRI).

Figure 35 shows a mock–up sample that contains multiple cracks as well as surface pitting. The largest pits are identified with red arrows; the simulated cracks are indicated with blue circles.

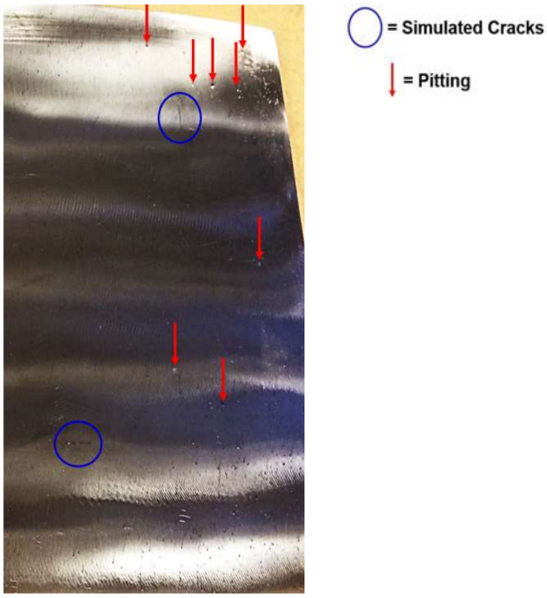


FIG. 35. DSS NDE mock–up containing simulated cracks (EDM notches) and surface pitting (courtesy of EPRI).

8.2.2.2. Development of robotics

For a dry storage canister environment, it does not matter how good an NDE technique is if it cannot be successfully deployed into the DSS. Therefore, robotics development is of key importance for inspecting DSSs. Various robotics platforms have been under development and are at different stages of technology readiness level [64]. Several of the robotic systems either developed specifically for vertical DSS inspection or proposed for modification from other applications are shown in Figs 36 to 39.

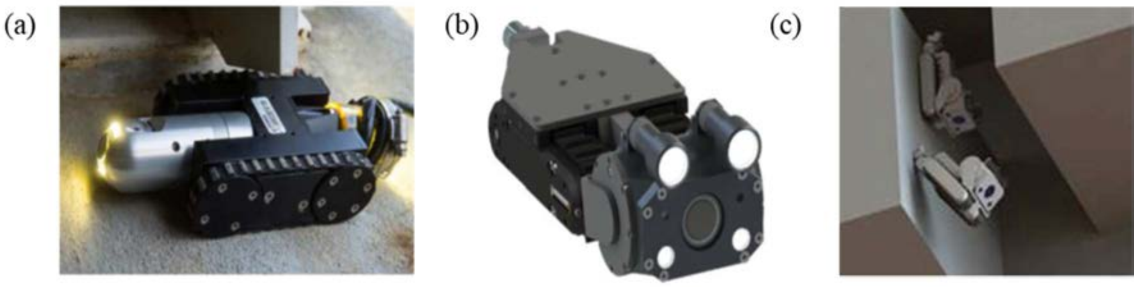


FIG. 36. DSS robotic platforms proposed for the inspection of canisters. (a) crawler protoype, (b) and (c) CAD models (courtesy of EPRI).

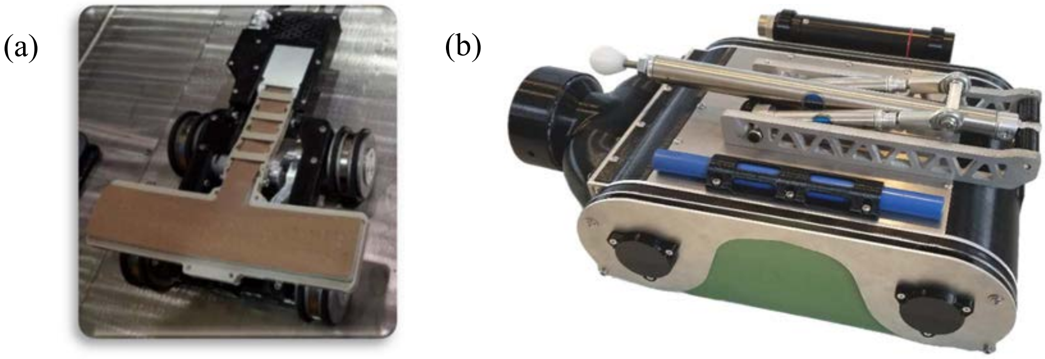


FIG. 37. DSS robotic platforms developed for the inspection of canisters. (a) Vacuum suction robot carrying a radiation probe (top). (b) Magnetic wheel robot carrying an eddy current array payload (courtesy of EPRI).

FIG. 38. Snake robotic platform proposed for the inspection of dry storage canisters (courtesy of EPRI).

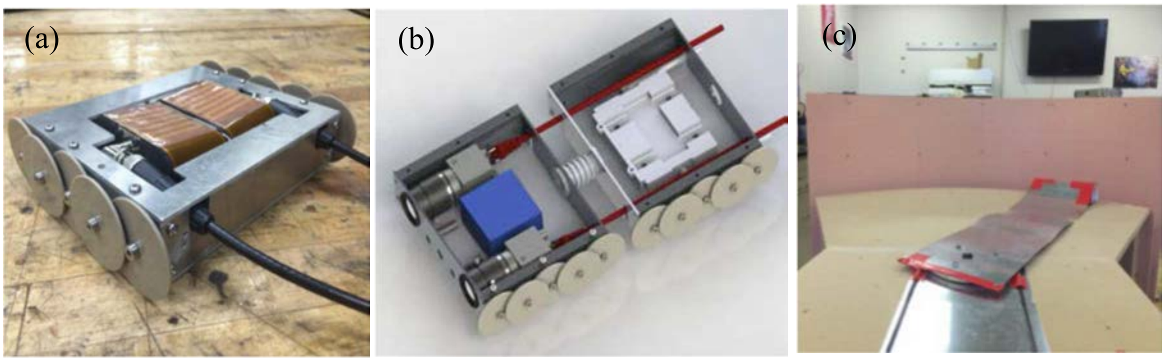


FIG. 39. Robotic train platform and delivery system developed for the inspection of canisters. (a) Robot carriage containing a guided wave platform. (b) Internal cutaway view of the robot. (c) Rotating delivery platform. (courtesy of EPRI).

Various laboratory and field trials have been completed by collaborations of utilities, vendors, technology providers, and so forth. These field trials have provided a means for accelerated development not only of in situ robotics, but also of inspection ring concepts for detailed inspection.

Figure 40 shows the site of one such field trial, in which various robotic platforms containing NDE systems were evaluated for fit, form, and function.

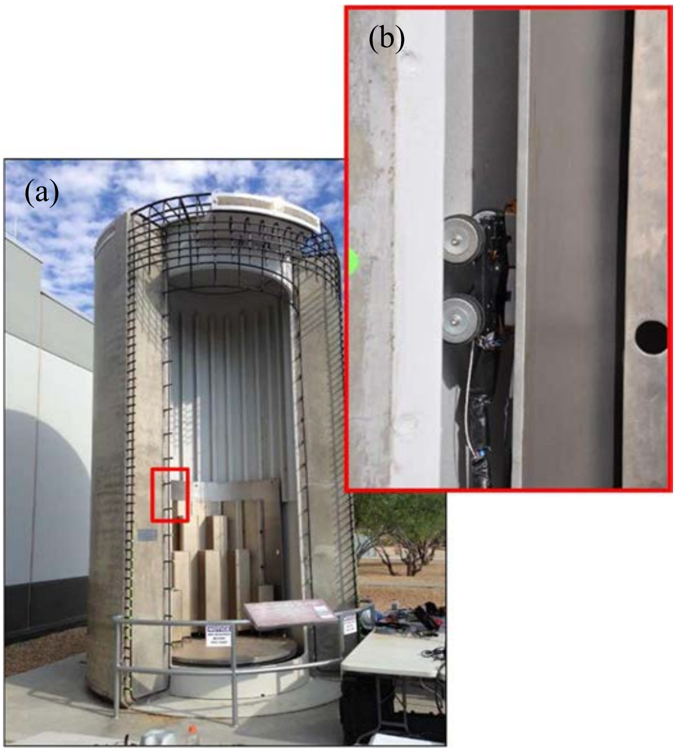


FIG. 40. (a) DSS mock up at the Palo Verde Energy Education Center. (b) Robotic inspection platform deployed to evaluate inspection capability and ability to access the canister within the narrow annulus space (courtesy of EPRI).

8.2.2.3. Development of NDE systems

This Section will provide a brief overview of several NDE technologies identified in Table 10, including strengths, weaknesses, experience with inspecting nuclear components and/or dry storage canisters, specific developments underway (or needed) for a DSS environment, and other important factors related to each technique.

(a) Visual testing

Visual testing (VT) is a method that is, in many respects, the simplest and easiest method to understand. VT, however, requires significant operator skill and expertise to ensure a high quality examination.

(i) Strengths

VT data are simple to acquire and readily interpreted. VT can cover a wide area quickly, can inspect virtually any material, and is easy for the non–expert examiner to evaluate. Components with complex geometries can be readily inspected. Typically, it is easy to determine during the inspection whether the VT data being acquired are of sufficient quality.

(ii) Weaknesses

VT is limited to surface inspection for opaque components (such as metals) as well as line of sight access. VT cameras are sometimes bulky and may not operate effectively in a highly radioactive environment (though miniature cameras and borescopes can be used). Defects may be masked at the surface, making detection challenging in some cases.

(iii) Dry storage inspection experience

VT has been utilized more widely than any other NDE technique, both in general and for DSS inspections. DSS NDE inspections typically start with VT, and if indications of degradation are observed, then alternative and more sensitive NDE technologies can be deployed for follow–up inspections.

(iv) Specific developments for a DSS environment

Recently, borescope inspection cameras have been developed that use highly radiation tolerant quartz fibres for transmitting visual data via fibre optics. Quartz fibres can withstand very high radiation doses without the browning effects seen in other optical fibres.

(v) Other factors

VT can be performed to meet different criteria based on resolution, inspection angle, lighting, distance from the inspection surface, etc., and can be classified into several categories (below) that are used for different applications:

- General visual, visual examinations that do not meet the requirements for the following four categories;
- VT–3, visual examinations utilized to evaluate the general mechanical and structural conditions of SSCs. Also known as a general examination (can also be used for identifying loose/missing parts or to verify clearances);
- VT–2, examinations utilized to detect leakage from pressurized components (pipes, tanks, etc.);
- VT–1, examinations utilized to detect discontinuities (i.e., cracks and other defects) on the surface of SSCs. VT–1 examinations have specific criteria: angle requirements, distance from the inspected surface, lighting, character cards resolution, etc;
- Enhanced VT–1, similar to a VT–1 examination, but with enhanced resolution.

(b) Eddy current testing

Eddy current testing (ECT) involves the use of induced electrical eddy currents within conductive materials that detect perturbations in the material that interact with the induced eddy current. Eddy currents are often used for crack detection and material sorting. They can also be used to evaluate the thickness of thin layers of paints or coatings on the surface of conductive materials.

(i) Strengths

ECT is very sensitive to surface and near surface defects and can find defects that may be occluded from a VT or other surface examination. Sensitivity is increased via the use of differential coils (which detect the signal difference between two coils instead of the absolute signal from a single coil), but at the expense of becoming directionally dependent, as described below.

(ii) Weaknesses

Detection can be directionally dependent (flaws parallel to the coil winding can be masked; similarly, for differential coils, flaws that interact equally with both coils can be missed.) For this reason, eddy current arrays that use multiple channel pairings in different (orthogonal) orientations are increasingly being used to reduce or eliminate this issue. Eddy current detection of defects is dependent on factors such as material conductivity, magnetic permeability, and frequency, and eddy currents will generally not penetrate deeply into materials. Surface finish can cause issues for some tests, but modern flexible arrays help to mitigate some of these effects.

(iii) Dry storage inspection experience

Eddy current arrays have been utilized on mock-up samples in laboratory testing as well as on unloaded systems in the field and could be deployed on loaded systems if the need arose. Inspection rings incorporating ECT are also being developed for field deployment.

(iv) Specific developments for a DSS environment

Curved eddy current array probes that utilize flexible printed circuit boards have been developed for use with DSS inspections [4, 65].

(v) Other factors

Flexible printed circuit boards present a number of unique advantages for ECT on surfaces (low form factor, ability to adapt/conform to most surfaces, etc.), but also pose some challenges. Early inspection testing identified the potential to sever electrical connections as a result of contact with rough/sharp surfaces that could be encountered, either around the weld or from surrounding overpack surfaces.

(c) Ultrasonic testing

Ultrasonic testing (UT) involves the detection of defects based on the interaction of acoustic waves in the material with the structure and/or defect. UT can be used to detect cracking, porosity, and wall thinning, and can also be used to characterize material properties (sound velocity and attenuation).

(i) Strengths

UT methods are capable of full volumetric examination of SSCs, meaning that UT is useful for both length and depth sizing of flaws. If a crack is observed in a DSS, UT methods will be critical for identifying the surface length and depth of the flaw. UT examination could also be useful for detecting cracking or corrosion in bolting for cask based systems. Additionally, some UT methods (such as laser UT) do not require contact with the inspected surface or couplants to be effective.

(ii) Weaknesses

While UT has many advantages for defect detection, it also has some drawbacks. For example, ultrasonic inspections typically require a couplant, such as water or ultrasonic gel, which can cause issues in a DSS environment. Concerns include delivering the couplant, removing it from the surface at the completion of the examination, and interactions/long term effects of the couplant with the surface; the prime concern being potential negative long term effects of salt redistribution on a canister surface due to the addition of water. Additionally, more operator skill and experience/training are required than with other techniques. The material geometry, surface conditions, etc., may all affect the quality of the examination, and temperature may affect the quality of the UT signal. UT inspections may also be directionally dependent, meaning that flaws that are parallel to the direction of the acoustic wave may go undetected.

(iii) Dry storage inspection experience

Ultrasonics have been utilized on mock–up samples in laboratory testing, and non–contact laser UT methods are being developed for field implementation. Inspection rings incorporating ultrasonic examination tools are also being developed for field deployment.

(iv) Specific developments for a DSS environment

Owing to the desire to avoid the use of couplants, miniaturized laser UT systems have been under development to provide non–contact volumetric examination for DSSs.

(v) Other factors

DSSs may also have occluded areas that present challenges for getting the UT probe close enough to the inspected area to get accurate data. Multiple beam skips, or alternative methods such as guided waves, may be utilized to overcome this challenge.

(d) Guided wave testing

Guided waves are a form of long range ultrasound in which the material itself is used as a waveguide for the acoustic signal. Many of the advantages and limitations are similar to those of UT and will not be discussed in great detail. Guided waves can be utilized for inspections via the use of electromagnetic acoustic transducers (EMATs).

(i) Strengths

Guided waves can be used to detect defects located at great distances from the probe (as much as 500 metres away). This characteristic is useful for the inspection of occluded areas; however, the added range of inspection is at the expense of resolution.

(ii) Weaknesses

Guided waves typically have a 'dead zone' near the probe where the guided waves are forming, and in which no useful data can be obtained. Guided waves lose resolution compared to traditional ultrasonic techniques and, while they are useful for defect detection, are not currently capable of accurate length and depth sizing.

(iii) Dry storage inspection experience

Guided waves have been utilized on mock–up samples in laboratory testing as well as on unloaded mock–ups in the field.

(iv) Specific developments for a DSS environment

Low profile guided wave probes have been developed for the inspection of DSSs.

(v) Other factors

Robotic deployment of these systems needs to take into account potential interactions of the strong magnets in the probe with the carbon steel materials in the DSS overpack liner, causing the probe to become stuck in the overpack. Similarly, the magnets could negatively interact with other onboard electronics, and these interactions need to be understood and mitigated.

Acoustic emission testing was included in Table 10; however, since it is primarily used as a monitoring technique, it will be covered in Section 8.4.

(e) Inspection ring examination system

For horizontal storage systems, a different strategy has been developed, in which the canister is moved to the NDE system rather than the other way around. The inspection ring examination system (see Fig. 41) has been specifically designed for ageing management inspections during the period of extended operations.

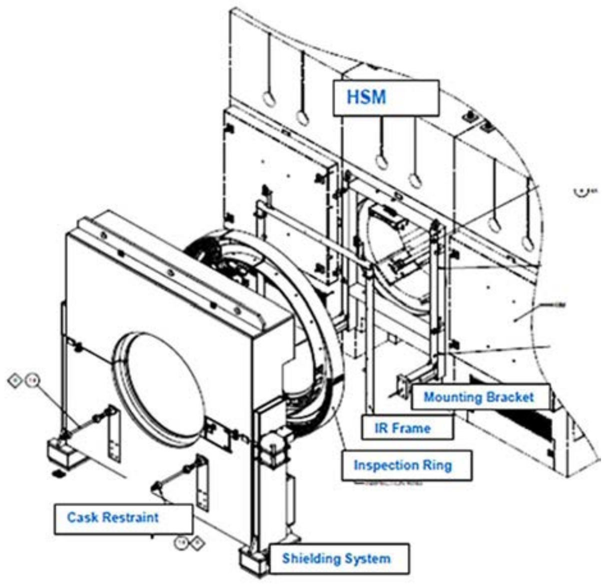


FIG. 41. Inspection ring subcomponents (courtesy of Karan Mauskar Orano TN).

The inspection ring examination system consists of four main tool functions: surface cleaning and sampling, VT, ECT, and UT. It also includes a brush to remove any deposits on the dry storage canister. The canister is inspected by withdrawing it from the HSM into the transfer cask. The entire cylindrical surface of the canister can be inspected as it is gradually withdrawn into the transfer cask. The mounting bracket attaches to the horizontal storage unit door embedment's and supports the inspection ring. The inspection ring frame is mounted on the HSM access door and slides into the inspection ring mounting bracket. It is designed to hold the inspection ring and its associated components. The shielding system is designed to enclose the inspection ring, the mounting bracket, and the inspection ring frame to provide radiation shielding during the inspection operation.

8.2.3. Non-destructive examination operational experience

Numerous inspections have been done to date for welded dry storage canisters deployed into concrete overpacks, including both horizontal and vertical configurations. Owing to the differences between the geometries, the space between the canister and storage overpack can be ring shaped. with gap dimensions ranging widely from less than 50 mm to over 100 mm. These specific design characteristics of the DSS require specialized robotic delivery systems to deploy the requisite NDE technologies into the gap between the canister and overpack to perform the inspection. Both vacuum suction and magnetic wheel robots have been developed. These systems have been designed for operation in high temperature, high radiation environments; they have been tested in over thirteen field trials, deployed over a dozen times in the field, and used to inspect over 20 different loaded DSSs of various DSS vendor designs.

8.3. STRUCTURAL CONCRETE

Concrete is used for the construction of DSSs, but it is also used as the material for the pad upon which the DSSs rest and for the storage facility building (if used). DSSs employ concrete and other materials to fulfil specific design functions. Concrete is one of the most commonly used materials for DSSs and is used in welded canister systems as a shielding material around the canister. Concrete, along with the dense rebar inside, is an effective shielding material and reduces doses by several orders of magnitude from the high radiation emitting from the canister to much lower levels outside the surface of the concrete overpack. Concrete provides an additional barrier for mechanical impact protection. The concrete overpack may also support seismic functions. Concrete is also used as the main construction material of concrete bolted casks (e.g., CONSTOR type).

Concrete is one of the most commonly used construction materials in the world because of its relatively low cost, favourable material properties, long term viability, and ease of construction. Concrete is a composite material containing a number of different materials that are combined, and, as such, concrete has a number of potential degradation mechanisms [15]. These include alkali—silica reaction, freeze—thaw degradation, rebar corrosion, etc. For this reason, it is important to conduct NDE examinations of concrete materials to verify the current state of the material. However, since concrete contains many different materials with highly variable material properties, it can be quite complex to inspect and monitor.

Various other materials are utilized for individual DSS designs. Some utilize support structures for DSSs, while others use channels or standoffs to ensure proper positioning of the canister in an overpack for ensuring cooling and other design functions. Each individual design may employ different materials for different design functions, and these are handled on a design specific basis.

Operational experience related to inspection and repair of concrete structures for horizontal and vertical DSSs exists in the USA. at several ISFSIs; the primary relevant code governing concrete inspections and repairs is the American Concrete Institute (ACI) Standard 349.3R–02 [16]. Other portions of the ACI code are also relevant to addressing several ageing mechanisms identified in the U.S. NRC MAPS report [15]. The ACI 349.3R–02 code includes a quantitative three–tier acceptance criterion for visual inspections of concrete surfaces, namely, (1) acceptance without further evaluation, (2) acceptance after review, and (3) acceptance requiring further evaluation. According to the U.S. NRC definitions Acceptance without further evaluation" signifies that a component is free of significant deficiencies or degradation that could lead to the loss of structural integrity. "Acceptance after review" signifies that a component contains deficiencies or degradation but will remain able to perform its design basis function until the next inspection or repair. "Acceptance requiring further evaluation" signifies that a component contains deficiencies or degradation that could prevent (or could prevent prior to the next inspection) the ability to perform their intended important to safety design basis function" [14].

Case studies of inspections at the Three Mile Island ISFSI have shown instances of concrete degradation, including freeze—thaw cracking Fig. 42 (a) and spallation, which were remediated using ACI approved repair methods as depicted in Fig. 42 (b).

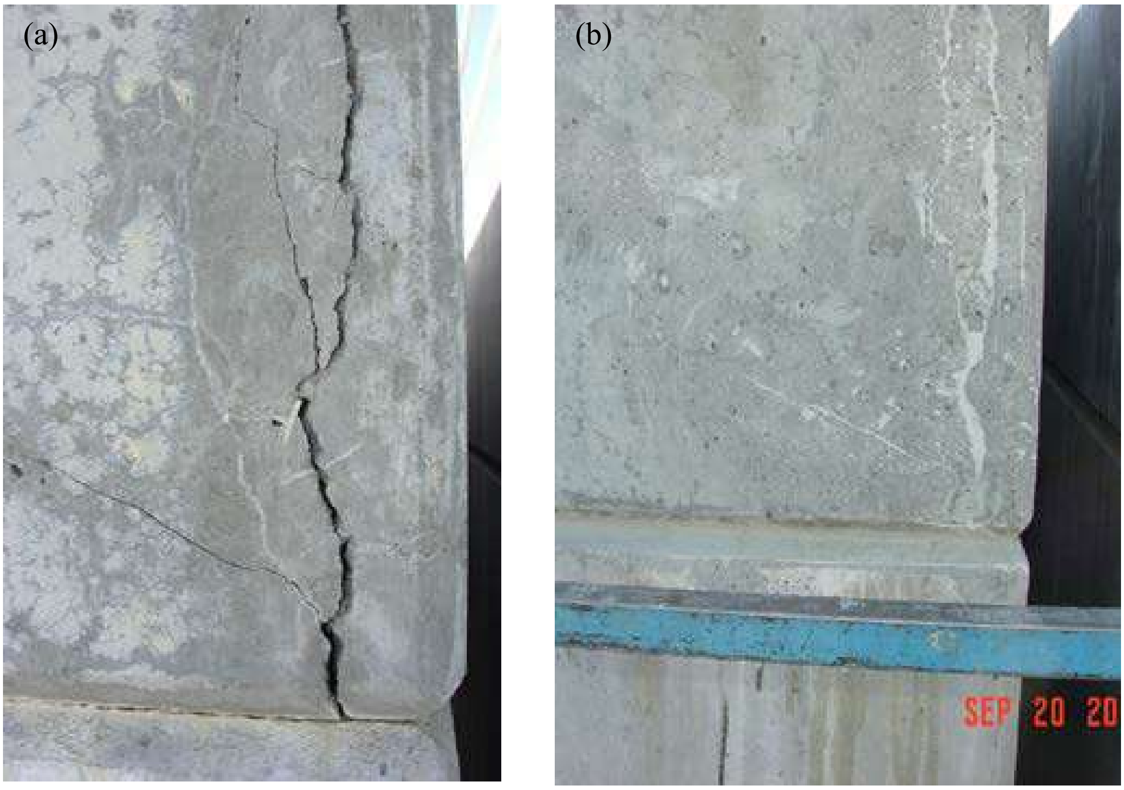


FIG. 42. Concrete horizontal storage module crack repair (courtesy of U.S. NRC).

8.4. DEVELOPMENT OF MONITORING SYSTEMS

Monitoring systems are, or could be, employed to help augment inspections or potentially reduce the frequency of (or eliminate) inspections. Continuous and periodic condition and performance monitoring would enable timely detection of ageing effects, such as pressure drop and/or leakage from the confinement boundary of bolted casks or welded canisters, thus allowing mitigation and repair to continue effective ageing management of DSSs and preserve public safety and the environment. Various monitoring technologies are at different stages of development and implementation. The primary monitoring systems that are in common usage today are pressure monitoring systems for bolted cask systems and temperature monitoring systems for both bolted cask and welded canister systems.

The following subsections describe a number of potentially useful monitoring systems that could be employed for DSS monitoring, but is not limiting and other monitoring systems could potentially be deployed to provide useful and valuable monitoring results [54].

8.4.1. Pressure monitoring systems

Pressure monitoring systems are employed on bolted cask systems. These systems have a double seal design with an internal seal between the cask internals and an intermediate cavity, and an external seal between the cavity and the environment. The double seal design allows for monitoring the pressure difference in the intermediate cavity such that it is possible to not only detect leaks, but also to determine whether the leak occurred in the external or internal seal.

8.4.1.1. Strengths

Casks are monitored online such that operators are notified quickly in the event of a leak occurring in a system.

8.4.1.2. Weaknesses

The systems may not be able to immediately detect a small leak when the change in pressure difference is slow.

8.4.1.3. Dry storage monitoring experience

Significant operating experience has been obtained over many years of operating these systems.

8.4.1.4. Specific developments for a DSS environment

Research on long term behaviour of seals is continuing to develop models for life prediction (see Section 6.1.2.).

8.4.1.5. Other factors

No other factors are applicable.

8.4.2. Temperature monitoring systems

Temperature monitoring systems are employed on many bolted cask and welded canister systems to monitor for blocked vents. Additional systems are proposed to monitor the temperature distribution on canister systems (or selected discrete points) in a manner that could detect the presence of a leak, since the loss of helium over time would result in a redistribution of the temperature profile as well as differences in peak temperatures [31].

8.4.2.1. Strengths

Temperature monitoring systems are suitable for monitoring for vent blockages in welded canister systems and can provide online feedback to operators. The monitoring methodology for leakage based on changes in temperature, three examples of which are described below, has been demonstrated, though practical temperature monitoring systems for canister helium leakage detection have not been implemented to date.

8.4.2.2. Weaknesses

Current operating systems for temperature monitoring can identify blocked vents for welded canister systems, but not the active degradation of canisters or leakages. While future systems may have this capability, challenges remain in implementation for existing DSSs, including limited accessibility to the canister surface, the placement of the temperature sensors, and autonomous long term remote monitoring operation.

8.4.2.3. Dry storage monitoring experience

Vent monitoring systems have been implemented and operational at a number of sites for many years and have a robust history of performance. Canister surface temperature monitoring systems have not yet been deployed to the field.

8.4.2.4. Specific developments for a DSS environment

Existing and proposed systems have been specifically designed for use with DSSs. Any proposed monitoring system will need to utilize temperature measurement devices that are sufficiently small to fit into DSSs, as well as evaluate communication methods. Wireless communication is very difficult because of the large amounts of shielding and conductive materials utilized in DSS designs, while wired communication has to address accessibility and placement of temperature sensors or thermocouples) near or in contact with the canister surface at desired locations. Recent advances in robotics development as described in Section 8.2.2. for inspection and repair, and demonstrations in several in service DSSs, indicate that accessibility and placement of temperature sensors near or on the canister surface are feasible and need to be demonstrated in a full scale test.

8.4.2.5. Other factors

No other factors are applicable.

8.4.2.6. Methods for Temperature Modelling

As mentioned above, monitoring of temperature distribution can be used to detect leakage of helium gas from a canister. Three methods are briefly described below; Fig. 43 shows the locations of the various temperature measurement points on the cask lid and canister.

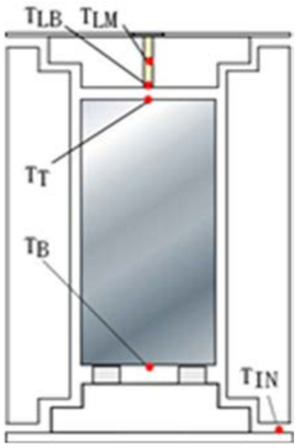


FIG. 43. Temperature measurement points in a ANL and CRIEPI project (courtesy of ANL).

Where:
T_{IN} – Inlet Temperature
T_B – Canister bottom temperature
T_T – Canister top temperature
T_{LB} – Overpack lid bottom temperature
T_{LM} – Overpack lid middle temperature.

The most prominent signal (i.e., the largest temperature difference) due to a leak is obtained with the ΔTBT method. However, the most suitable method for a particular application will depend on the cask design.

(a) ΔTBT method

The ΔTBT method is used for leak detection [66]. Figure 44 (a) shows a time profile of the temperatures at T_T and T_B (top and bottom of the canister, respectively) before and after the leak. Figure 44 (b) shows the relation between ΔTBT (obtained by subtracting T_T from T_B) and helium pressure.

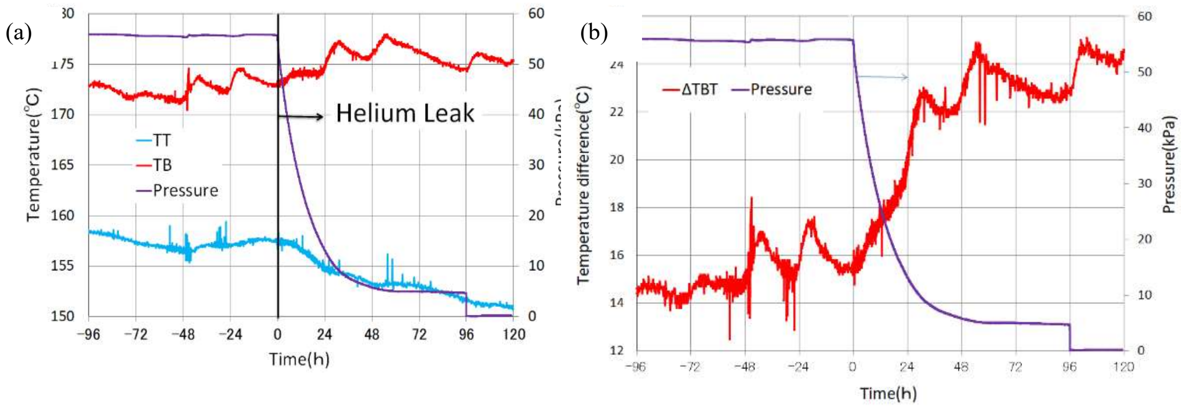


FIG. 44. (a) Temperatures at TT and TB, and helium gas pressure. (b) The relationship between ΔTBT and helium gas pressure. Collaborative project ANL and CRIEPI (courtesy of ANL).

(b) TT method

Figure 45 (a) shows a time profile of the temperatures at the middle (T_{LM}) and the bottom (T_{LB}) of the cask lid. The difference between the two temperatures increases after the leak begins. The TT method uses a time profile of the temperature difference obtained by subtracting T_{LB} from T_{LM} [67-70]. Figure 45 (b) shows the relation between δ ($T_{LM} - T_{LB}$), the increase in the difference between T_{LM} and T_{LB}, and helium gas pressure. A gas leak can be detected by using temperature information from only the top of the canister.

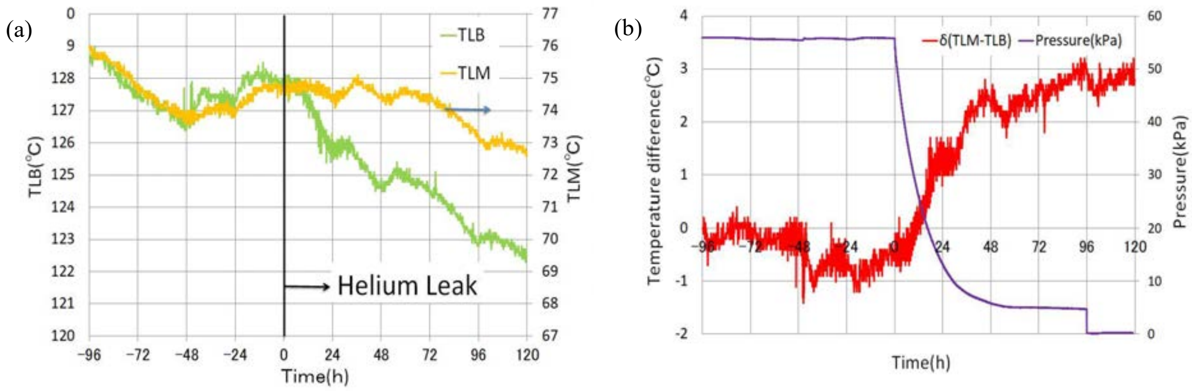


FIG. 45. (a) Temperatures at middle and bottom of cask lid. (b) The relationship between δ ($T_{LM} - T_{LB}$) and helium gas pressure. Collaborative project ANL and CRIEPI (courtesy of ANL).

(c) TB method

The relation between T_B and air temperature at the inlet (T_{IN}) is shown in Fig. 46 (a), and the relation between heat flux difference and the internal pressure of the canister is illustrated in Fig. 46 (b). There the qB curve indicates the value of the heat flux difference relative to the heat flux at the reference time of 36 hours before the leak. It is seen that qB increases with the leak induced decrease of the internal pressure. A gas leak can be detected by using temperature information from only the bottom of the canister [7].

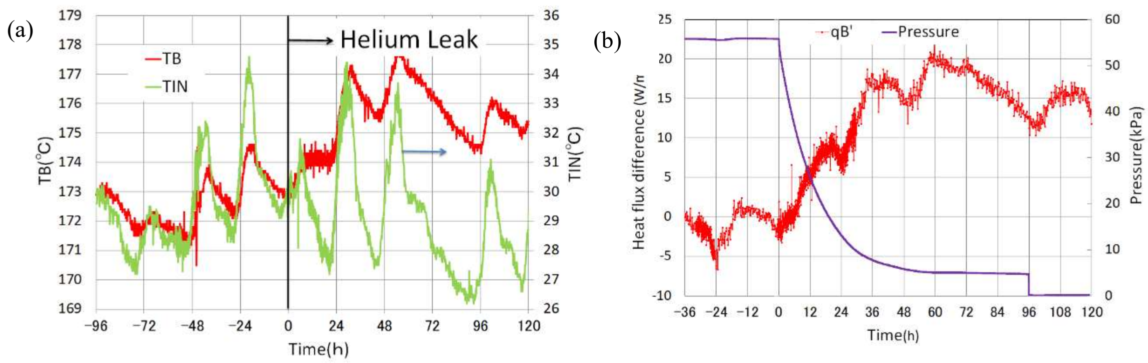


FIG. 46. (a) Temperatures at TB and TIN. (b) Relation between heat flux difference and helium gas pressure. Collaborative project ANL and CRIEPI (courtesy of ANL).

8.4.2.7. Remote area modular monitoring for temperature measurement

Remote area modular monitoring for temperature measurement (RAMM–TM) is “an innovative remote monitoring device that can detect gas leakage from canisters containing SNF by measuring changes in surface temperature differences between the top and bottom of the canister” [71]. The performance of RAMM–TM [72] in detecting canister gas leakage from a small, simulated chloride induced stress corrosion crack has been demonstrated experimentally. The device was able to detect a leakage of helium and air gas within hours of the leakage commencing. It uses the differences in surface temperatures measured at the top and bottom centre of the canister (ΔTBT) and their changes during gas leakage (depressurization) as an indicator. Advanced data analytics and high performance computing STAR–CCM + CFD simulations are being developed for demonstration and application of RAMM–TM [73] in actual dry cask storage systems.

8.4.3. Acoustic emission monitoring systems

Acoustic emission (AE) is an NDE method that detects and tracks the progression of cracks and other defects via ‘listening’ to the minute acoustic pulses that occur as cracks grow. These pulses travel through the material and can be detected using acoustic emission sensors. This method can be used to monitor DSSs for the propagation of defects.

8.4.3.1. Strengths

Acoustic emission offers the capability of monitoring a DSS such that future inspections could be eliminated (or their frequency could be reduced). Acoustic emission can be used to monitor and track the progression of defects and, if utilized with multiple probes, can also provide location information for detected signals.

8.4.3.2. Weaknesses

While AE offers a powerful monitoring solution, it is not without drawbacks. First and foremost, it cannot provide information retroactively, meaning that if a crack were already present in a material and had stopped growing, then AE would not detect the crack. So, unless deployed from day one, an AE test/inspection cannot function as a stand alone inspection because if a full through wall crack were present and had stopped growing, the AE system would not be able to detect the breach of the canister wall. For this reason, AE monitoring either needs to be supported by another inspection method to provide a baseline, or it needs to be deployed from the time that the canister is loaded with fuel.

8.4.3.3. Dry storage inspection experience

Acoustic emission testing has been evaluated on laboratory samples as depicted in Fig. 47 as well as on unloaded systems at vendor facilities and in the field [4].

FIG. 47. Acoustic emission testing of a growing SCC flaw (courtesy of EPRI).

8.4.3.4. Specific developments for a DSS environment

Previous field testing [4] demonstrated that AE monitoring could be performed using connections to the overpack instead of the welded stainless steel canister.

8.4.3.5. Other factors

Acoustic emission could also be affected by external noise events, such as inadvertent contact with the canister and weather events such as hailstorms; however, no issues have been observed with foot and vehicle traffic in the vicinity of the canister.

8.4.4. Helium monitoring systems

Helium monitoring systems employ helium detectors to detect the presence of a leak in a DSS by attempting to evaluate the loss of internal helium from the canister to the environment.

8.4.4.1. Strengths

Helium leak detection is a mature science that has been successfully used in numerous industries to detect leaks of various sizes. As canisters are filled with inert helium, no further helium needs to be injected, so such systems could potentially be deployed with relatively low costs.

8.4.4.2. Weaknesses

Helium is a common material and is present in the atmosphere in small amounts. This has caused issues in the past, with helium leak detection picking up background levels of helium in the environment due to the sensitivity of the detectors used. Similarly, at the instant of a crack going through wall, the dimensions of the crack will be very small (opening width of microns and length of tens to hundreds of microns); as such, leak rates will be very low at that instant. Similarly, as the helium leaks, internal pressures will slowly decrease, creating less driving force for a leak. At the same time, while the crack continues to grow, leak paths will open up to become wider and longer. It is currently unclear how to deal with these competing factors.

8.4.4.3. Dry storage monitoring experience

No such systems have been implemented for DSS monitoring.

8.4.4.4. *Specific developments for a DSS environment*

Helium leak detection systems are not designed for high radiation dose environments. However, since these systems could be deployed at vent locations, radiation doses are anticipated to be low. It may also be beneficial to install helium detection sensors at inlet and outlet vents to measure helium differentials to improve sensitivity and reduce false call rates.

8.4.4.5. *Other factors*

No other factors are applicable.

8.4.5. Concrete monitoring systems

Concrete is susceptible to a large number of potential degradation mechanisms based on the local environment and materials of construction [15]. To assist with monitoring of these different degradation mechanisms, various corrosion sensors have been developed to monitor concrete, many of which are commercially available [17]. Integrated sensor packages have also been developed that measure temperature, rebar corrosion, resistivity, chloride ion concentration, and other relevant parameters [19].

8.4.5.1. *Strengths*

Many concrete sensors are commercially available at relatively low cost. Such sensors could be utilized in a low radiation environment on the external surface of the DSS.

8.4.5.2. *Weaknesses*

Since concrete is subject to a number of degradation mechanisms, it could be cumbersome to install and monitor a myriad of sensors to monitor for many different potential degradation mechanisms on a large number of DSSs

8.4.5.3. *Dry storage monitoring experience*

Such sensors are already in use in some countries; however, they are not in widespread use.

8.4.5.4. *Specific developments for a DSS environment*

Since sensors would likely be used on the external surfaces of DSSs, they would not be expected to be subject to significantly elevated dose rates. However, some DSSs have concrete encased within steel. In these cases, there is no direct access to the concrete, so it may not be possible with currently available technologies to monitor concrete in these specific situations.

8.4.5.5. *Other factors*

As sensors are continuing to be developed, they will need to be evaluated for use in a dry storage environment.

8.4.5.6. *Concrete Corrosion Sensors at Atucha–1*

In Argentina the CNEA has developed specific corrosion sensing devices to monitor concrete structures. Sensors have been installed at a DSF and a silo mock–up at the Atucha–1 NPP site (see Section 7.3.2.1). Each sensor contains several electrodes that allow measurement of various parameters, namely, corrosion potential, corrosion current, oxygen flow, chloride content, concrete resistivity, and temperature (Fig. 48). All these variables need to stay within their respective safety ranges in order to ensure proper rebar behaviour, as stated in specific standards and other recommendations [54, 55] .

The corrosion sensors contain one piece of the same black steel as the rebars as a working electrode (E in Fig. 48), two electrodes of inert material (A and B), a reference electrode (D) and a thermometer (C), all of them embedded in epoxy resin. The electrodes proposed as reference electrodes consist of metal oxide 3 mm diameter activated titanium rod stock, which is commercially produced for permanent impressed current anodes used for cathodic protection of reinforcing steel in concrete.

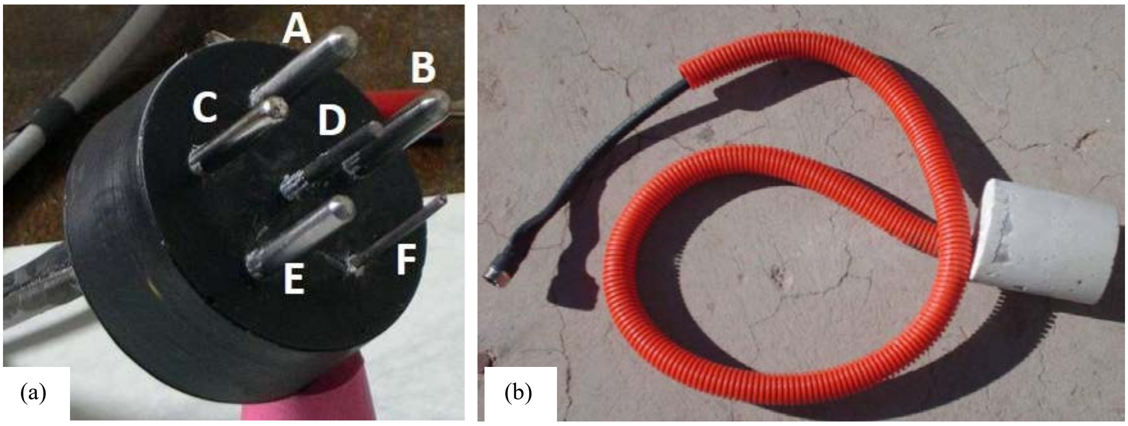


FIG. 48. Concrete corrosion probe. (a) Showing the electrodes. (b) Sensor covered with a porous mortar, ready to be inserted into the concrete structure. (Courtesy of CNEA).

The corrosion rate is measured using a variation of the 'classic' linear polarization resistance technique, which consists of polarizing the working electrode at a preselected current density using the two inert electrodes as counter electrode, measuring the potential of the rod against the metal—metal oxide electrode after a certain period, and applying the known Stern–Geary relationship [74].

The concrete electrical resistivity, the oxygen flow and temperature are measured as described in Ref. [17].

The analysis and interpretation of the data obtained are performed as described in Ref. [75]. These devices can be installed during construction of the facilities by immersing them into the concrete during cement casting, or they can be placed in existing structures by inserting them into holes properly drilled in the vicinity of the rebars.

8.4.6. Radiation monitoring systems

Many nuclear sites (either at or away from reactor) utilize radiation monitoring systems to ensure that public health and safety is maintained and compliance with regulatory limits is achieved. Such a technique would only be useful for detecting leaks that release radioactive materials. It could be argued that a leak with no radiological consequence would be of minimal concern; however, a leak (via a through wall crack or a seal breach) may violate the design basis.

8.4.6.1. Strengths

If a leak of radiological consequence occurred (caused by release of radioactive material through a crack or a seal breach), radiation monitors would be able to detect it.

8.4.6.2. Weaknesses

Such techniques may not detect a leak if no radioactive materials are released. Since fuel (UO_2) is more than ten times as dense as water, and fuel failures are very rare [76], and considering many other factors (how much material remains suspended in air versus falling to the bottom or depositing on walls after many years of storage, how much material can fit through a crack or other tight breach, etc.), detected radiation doses may be very near or at background radiation levels even if breaches do occur.

8.4.6.3. Dry storage monitoring experience

Such systems have been employed for many years at many sites and a large amount of data has been accumulated related to dose rates in the storage facilities and at the location boundaries.

8.4.6.4. Specific developments for a DSS environment

Since these systems have been utilized widely at many sites, data collection and trending would be the primary needs. There would also need to be a method for avoiding false calls built into such monitoring systems to account for increases in measured radiation doses as new DSSs are loaded with fuel and put into service in the vicinity of these monitors. There can also be consideration of decreasing radiation doses as the radioactive material decays, for trending purposes.

8.4.6.5. Other factors

No other factors are applicable.

8.5. MITIGATION AND REPAIR OF DRY STORAGE SYSTEMS

If degradation is observed on a DSS, mitigation or repair actions may be required to remediate the degradation. Mitigation and repair actions could potentially undertaken in situ or may include relocating the DSS into a dedicated on site facility or SFP to perform mitigating or repair activities.

Bolted cask systems may experience degradation of bolts, seals, or trunnions, or general corrosion under coatings and elastomeric sealing material. If seal degradation occurs at the primary lid, these DSSs are typically repaired in a SFP by relocating the cask into the SFP. There is some operating experience related to this process, owing to past degradation of these seals [3], though the implementation of this process requires the cask to be reopened in the SFP and the fuel redried. Because of the desire to minimize exposure of fuel to rewetting and drying cycles [77], some systems employ dual outer seals to minimize the potential for environmental corrosion of the outer seal. If no SFP is available at the facility, a hot cell can be used to transfer the fuel from one cask into another. This procedure requires opening the DSS, rewetting the fuel, repairing the seal, and redrying the cask. Bolt replacement or coating repairs are simpler and more straightforward to perform.

EPRI and the international nuclear power industry have developed several potential mitigation and/or repair strategies to address potential degradation phenomena. Mitigation and repair methods for welded DSSs are currently being researched, with commercial deployment of well established technologies already underway. Many of the current worldwide efforts are being coordinated within the EPRI ESCP, under the auspices of the Mitigation and Repair Subcommittee. Preliminary efforts in this area have been compiled into a summary report [78] that includes descriptions of different mitigation and repair options for welded canister systems as well as comparisons between the different technologies under development.

The intention of mitigation technology development is to prevent the potential for loss of canister confinement. These mitigation processes (e.g., cold spray, peening, low plasticity burnishing) can be applied either during fabrication or in targeted locations whilst the canister is in service. In the event of a potential loss of confinement function, several repair techniques are also under development and/or deployed on mock–ups or actual canisters; these techniques include welding and/or coating applications (welding, cold spray, organic/inorganic coatings, engineered paint, etc.). A successful case study included the development and deployment of laser peening mitigation technology to Holtec's Multi Purpose Canisters at SONGS [79]. Laser peening is a process that employs an intense laser pulse to create a plasma pressure wave that produces compressive residual stresses that eliminate one of the essential elements for the initiation of SCC. The first laser peened canisters for CISCC prevention were loaded at the San Onofre ISFSI on 23 August 2017, after extensive development and prototype testing, validation, and qualification [80].

Many of the repair technologies have been implemented by utilizing the robotic platforms developed for deploying the inspection tools. One mitigation/repair technology that has shown significant promise related to both canister surface mitigation and repair is known as cold spray [78, 81]. This technique deposits layers of particles that can be engineered so that the coated metal has similar or better properties than the original base metal. EPRI has worked with commercial dry storage and robotics vendors to develop and deploy cold spray repair technologies on robotic delivery platforms. After successful laboratory tests and simulated field trials, a USA utility worked with the dry storage vendor and deployed the cold spray technology to an on site test canister, demonstrating the feasibility of using this technology in the field. The utility plans to deploy it on loaded canisters as a contingency if SCC is observed.

9. UPDATE OF AGEING MANAGEMENT PROGRAMMES

During extended storage of spent fuel, the state of the storage system SSCs may change, related technical knowledge may change, and related regulations may change. With those changes, AMPs needs to be updated. Operating experience regarding age related degradation and ageing management is also captured in the update of AMPs. Operators are advised to review operating experience to confirm that the AMPs remain effective at managing age related degradation or make any necessary changes to AMPs to ensure their continued effectiveness. In this way, the AMPs are considered learning programmes that evolve over time to respond to operating experience. This approach is similar to the 'Plan/Do/Check/Act' (PDCA) cycle, introduced for ageing management of NPPs [82].

AMPs needs to reference a specific procedure to be used to obtain, review, and report operating experience. The assessment of operating experience can occur through:

– The operator's continual assessment of operating experience through its existing programmes, such as QA programmes;
– The operator's periodic assessment of the accumulated/aggregate operating experience and broader technical information at specific points in the operating period.

Both approaches are discussed in more detail in Sections 9.2 and 9.3.

9.1. OPERATING EXPERIENCE

In the evolution of an AMP, operating experience will be aggregated, trended, assessed, and (as appropriate) shared. 'Operating experience' is meant in a broad sense and may include the following information:

– Site specific operational data from inspections, monitoring, and AMP implementation, including information from the site CAP;
– Industrywide operating experience relevant to the design, materials, and/or operating/service environment;
– Vendor issued safety bulletins;
– Technological developments/advances in inspection, mitigation, and repair techniques;
– Research and development findings/results;
– Consensus codes, standards, or guides;
– Guidance documents and standards applicable to ageing management (e.g. IGALL [9], GALL–SLR[4], MAPS[15]);
– Non–nuclear operating experience (e.g., experience on materials degradation from other industries);
– Industry operating experience from all MSs and international organizations. Some information may be publicly available in certain countries [7]. For example, in the USA, the U.S. NRC develops publicly available generic communications that summarize operating experience for industry awareness or action;
– Results of operators' periodic assessments of operating experience, if such assessments are documented (see Section 8.3).

Information has to be relevant to the materials of construction, service environment, and the credible ageing mechanisms, and be published by reliable sources. In addition, the CRP participants generally recommend that the regulatory authority inspect operators' implementation of AMPs, including operators' periodic assessments of AMP effectiveness and any adjustments made to AMPs to respond to operating experience to ensure AMPs are effective for addressing ageing effects.

9.2. CONTINUAL/ONGOING ASSESSMENT OF OPERATING EXPERIENCE

AMPs need to be updated, when necessary, based on the ongoing review of operating experience. Operators are to review site specific and industry operating experience on an

[4] See Guidance for license renewal and subsequent license renewal on the U.S. NRC website www.nrc.gov.

ongoing basis through the operating period to confirm the effectiveness of the AMPs or identify a need to modify an AMP. This continuous review of operating experience may be accomplished through the operator's existing programmes, such as the QA programme.

9.3. PERIODIC ASSESSMENT OF ACCUMULATED OPERATING EXPERIENCE AND BROADER TECHNICAL INFORMATION

In addition to the ongoing assessment of operating experience that is part of an operator's QA programme, operators will conduct periodic assessments of accumulated/aggregate operating experience and technical information from sources beyond those used in normal programmatic activities. Consideration needs to be given to research programmes at all MSs and international organizations and operating experience from worldwide dry storage installations, nuclear power reactors, and non–nuclear industries.

In the periodic assessment, the operator performs and documents its assessment of the aggregate state of knowledge of ageing related information and operating experience to ascertain the ability of the AMP to adequately manage age related degradation throughout the licensed operating period. The periodic assessment ensures that, as knowledge and data become available from new analyses, experiments, and operating experience, the operators will revise the existing AMPs (or pertinent procedures for AMP implementation) as necessary to address any lessons learned or identified during the review of operating experience.

The AMP itself may include a structured approach for this periodic assessment including the following attributes:

– Timing and frequency. The timing and frequency of the assessment considers the expected time for ageing mechanism initiation and the rate of ageing degradation/progression. It also considers the expected availability of data from research programmes and the findings from prior periodic assessments;
– Performance criteria. Performance indicators to evaluate the effectiveness of each of the AMP elements;
– Proposed actions based on the assessment findings. The assessment may result in changes to preventive actions (e.g., adding a corrosion control measure), monitoring and inspection activities (e.g., scope, frequency), performance of mitigation activities/corrective actions (e.g., repairs or replacements), or changes to TLAAs.

The operator has to document the periodic assessment in a report to inform future periodic assessments and (as appropriate) share the operating experience and technical insights with other operators, MSs, and international organizations. The content of a documented periodic assessment may include:

– Summary of site specific monitoring data and inspection results, research findings, and operating experience;
– Aggregate impact of findings (including trends);
– Consistency with previously assumed degradation mechanisms and their progression (including a confirmation that any TLAAs remain valid);
– Effectiveness of AMPs;
– Corrective actions, including changes to AMPs;
– Summary and conclusions.

An example of a periodic assessment for storage AMPs in the USA is called a tollgate assessment (see Section 7.2.1). Guidance on tollgates may be found in NEI 14–03 [49], which defines tollgates as specific points in time when operators would be required to evaluate aggregate operating experience and other technical information and perform and document a safety assessment that confirms the continued safe storage of SNF. The AMID may be used as a resource to conduct the tollgate assessment (see Section 7.2.2). According to NEI 14–03, tollgate assessments will generally result in one of three conclusions:

– The information reviewed confirms the adequacy of current TLAAs and AMPs. Continued safe storage is expected to the next tollgate;
– Information is currently unavailable for a particular potential ageing related degradation mechanism. Plans to address the information gap have to be developed and implemented;
– The information reviewed introduces issues not currently managed adequately by current AMPs or assessed adequately in TLAAs. Corrective actions are required. This could involve changes to the TLAAs or AMPs, as appropriate, or could involve additional inspections, mitigation, repairs, or replacements of DSS SSCs.

10. CONCLUSIONS

A considerable number of storage systems have already reached the initially intended operational lifetime. Continued operation has been licensed for those facilities on the basis of a survey of the storage system condition. This knowledge is key for safe and reliable continuation of storage. AMPs have been introduced in many countries to gather and provide exactly this type of information.

Ageing management experience from NPPs has been a good starting point, and many storage facility operators have used staff with experience from NPPs to develop their dry storage facility AMPs. To make this knowledge, and the approaches chosen by different countries' regulators, accessible, this CRP collected experience from all participating countries. On the basis of the broad participation and the national information provided, a characterization of the main storage types and the identification of the key SSCs was completed.

During the CRP duration, significant development in the field of inspections of storage systems took place. The technology has evolved from lab scale to routine operation. Now it is part of the renewal process for storage licenses in the U.S.

Significant progress was made in the development of a monitoring method utilising temperature data from different locations of a canister.

Discussions among CRP participants have proven to be very fruitful and stimulated activities on ageing management in many countries.

As the storage durations are continuing to be extended, AMPs will remain an important safety pillow for DSSs. As the development of sensors has gained considerable momentum lately, those technologies might offer completely new monitoring possibilities.

APPENDIX I

NATIONAL AGEING MANAGEMENT GUIDANCE

I.1. GUIDANCE ON AGEING MANAGEMENT: ARGENTINA

In Argentina, maintenance of the spent fuel from nuclear power reactors is the responsibility of the operating company until plant shutdown. After that, the responsibility will be transferred to the government, through the CNEA. There is one dry storage facility at the site of the Embalse NPP called ASECQ–CNE, which has been operating since the early 1990s; another is under construction at Atucha–I NPP (ASECQ–CNA), and a third one is being considered for Atucha–II.

The Nuclear Regulatory Authority (ARN) implements the nuclear regulatory guidance in Argentina. Initially, it had not set any explicit requirements related to ageing management for storage facilities; all conditions the facilities had to comply with were set in their Construction, Operation and Life Extension Licenses. By 2020 ARN was in the process of elaborating a Specific Standard for the Dry Storage Facilities in Argentina, which was based on the IAEA Specific Safety GuideSSG–15, Storage of Spent Nuclear Fuel [83] . However, in the 7th National Report to the Joint Convention (2020), it is stated that "Upon request of ARN, the ASECQ has been included in the Ageing Management Programme for Power Plant Components and Systems Related to Nuclear Safety" [84]. As a consequence, a surveillance plan for baskets, interior lining and concrete structure of all the ASECQ silos system was incorporated. In addition to this surveillance action, a periodic measurement of aerosol and noble gases content inside the silos is conducted. During the period included in this Report, the 'Ageing Management Manual' was issued as part of the mandatory documents required by ARN for the Operating License of the second cycle of operation" [85]. This AMP also comprises surveillance operations on concrete structures and inner steel liners. Among the actions related to the former, a follow–up of cracks in the concrete is performed through visual inspection; as for the latter, moisture in the air inside the silos is periodically assessed. Other surveillance operations are considered only when a need could arise at some point, such as integrity determinations on samples of liner material and concrete, which have been exposed in a mock–up silo since the beginning of operation of the dry storage facility.

Construction, Operating, and Life Extension Licenses contain explicit restrictions pertaining to public safety. For example, they establish the maximum acceptable dose for people staying in the free transit zone. There are additional regulatory requirements that are not intended for dry storage installations; these establish standards which have to be fulfilled by the fuel in general, including when in storage. Other regulations, which add to the spent fuel storage regulatory framework, are related to safety in nuclear power reactors, and are specifically dedicated to design and accidents [86]. There is also a standard that refers to research reactors [87]. It provides criteria to restrict occupational exposures, what in turn impose restrictions to the spent fuel storage strategies.

I.2. GUIDANCE ON AGEING MANAGEMENT: CZECH REPUBLIC

In the Czech Republic, SNF from both NPPs and all research reactors is kept in dry storage facilities in dual purpose casks (DPC), until the receiving facility at the future disposal site is ready to accept the casks for re–packing into disposal casks and disposal (around 2065) [88].

All storage facilities are of indoor type and there are no plans to construct outdoor ones [89]. One reason for this decision is to provide a controlled environment for stored casks for better control of all ageing processes. All storage facilities provide the following basic storage functions:

- Storage of transport and storage casks containing SF;
- Handling of casks with a crane;
- Reduction of radiation exposure outside of the building to the minimum, which is well below the permitted values;
- Cooling of the stored casks and decay heat sink to the environment using natural aeration;
- Assurance of working conditions for the personnel of storage facility;
- Possibility to perform inspections and minor repairs of casks;
- Protection against weather effects;
- Prevention of unauthorized access, in conjunction with the physical protection system;
- Shielding from solar radiation.

The fundamental laws governing the licensing and approval process for any nuclear installation, incl. SNF storage facilities are the Building Act [90] and the Atomic Act [91]. Requirements for management of radioactive waste (RAW from nuclear installations and institutional radioactive waste) are defined in the Atomic Act and in the Decree on requirements for safe management of radioactive waste and decommissioning of nuclear installations or workplaces of category III or IV [92]. SNF is not considered to be RAW, but the same requirements as those applicable to RAW has to apply to SNF until such time that it is declared RAW by the producer or by the decision of the regulatory body.

There are other implementing regulations to the Atomic Act, such as the Decree on transport and type approval [93]. This decree provides details of type approval of DPCs. DPC safety case has to be performed as additional safety evidence for dry cask storage facilities, because the cask bears all of the following safety functions:

- Criticality control;
- Limit radiological exposure of workers and the population;
- Removal of waste heat;
- Prevent radiation leaks;
- Retrievability of SF in DPC and the cask itself.

The license issued by the Czech Republic regulatory body, SÚJB, is required for operation of every nuclear installation, including SNF storage facilities. Pursuant to Annex No. 1 to the Atomic Act (paragraph 1, letter f), the documentation for operation of a nuclear installation (safety case) includes, among others the AMP. Details regarding an AMP are listed in SÚJB Decree on nuclear safety [94]. The AMP has to include:

- A list of SSCs to be subject to ageing management process;
- A list of degradation mechanisms and impacts of ageing;
- A list of monitoring parameters and condition indicators that are used for the monitoring and determination of the development of the impacts of ageing;
- Rules for the monitoring and determination of the development of the impacts of ageing;
- A list of the acceptance criteria for monitoring parameters;

- Rules for assessing the monitoring parameters, assessing the current state of SSCs selected for ageing management and predicting the future state of these SSCs;
- A list of corrective measures in respect of a failure to meet the acceptance criteria for monitoring parameters;
- Rules for the monitoring of the efficiency of measures in operation and maintenance of SSCs selected for ageing management to mitigate or eliminate the impacts of ageing and the effects of degradation mechanisms on these SSCs;
- Description of the provision of feedback in order to measure the efficiency of the ageing management process;
- Rules for assessing the monitoring parameters, assessing the current state of SSCs selected for ageing management and predicting the future state of these SSCs;
- A list of corrective measures in respect of a failure to meet the acceptance criteria for monitoring parameters;
- Rules for the monitoring of the efficiency of measures in operation and maintenance of SSCs selected for AM to mitigate or eliminate the impacts of ageing and the effects of degradation mechanisms on these SSCs;
- Description of the provision of feedback in order to measure the efficiency of the ageing management process.

Also, within the type approval the DPC manufacturer, user, or importer has to submit to the regulator, within a complex suite of safety documentation, a preliminary AMP, which assesses the performance of key design components of the cask and specifies operational controls contributing to the ageing management of loaded casks.

An AMP has to be defined from initial service time, from the beginning DPC use. However, there are no plans to use DPCs substantially beyond their initial design period (40–60 years) as the deep geological repository is scheduled to be operational around 2065.

AMPs for DPCs consists of a set of operational controls performed through visual inspection of cask components, monitoring of pressures inside the cask and test of welds, if used in the design of casks. The ageing processes managed by these controls include corrosion of cask components, deformations and wearing out of trunnions, cask body, sealings and shock absorbers and functional degradation of pressure and temperature sensors.

AMP requirements are implemented within the operating documentation of the DPCs and SNF storage facilities. As of 2021, no ageing induced issues have been identified.

Since the first cask loading at the Dukovany ISFSF in 1995, no incidents or accidents have been reported. Operational occurrences, identified since first loading, were not related to any ageing processes and were had other causation factors, such as insufficient sealing of trunnion bolts, contamination of sealing surfaces during cask manufacturing, and SNF basket design flaws.

I.3. GUIDANCE ON AGEING MANAGEMENT: GERMANY

In Germany, SNF and high level vitrified waste (HLW) are kept in dry storage until the receiving facility at the future disposal site is ready to accommodate casks from the existing storage facilities. The technical concept and safety requirements for all dry cask storage systems for SNF and HLW are addressed in the recommendations of the German Nuclear Waste Management Commission (ESK) [56]. These requirements and recommendations are considered by the German licensing authority, the Federal Office for the Safety of Nuclear

Waste Management (BASE) and are implemented by site specific licenses. According to the principles of cask storage in Germany, the leakproof and accident resistant casks are the key safety element for dry storage. As confirmed in the site specific licensing procedures, the casks are suitable for at least 40 years of storage. Accordingly, the licenses for these dry storage facilities are currently limited to 40 years, starting from the first emplacement of a DPC, and will expire between 2034 and 2047. As a consequence of the disposal site selection procedure commencing in 2017, the programme for the responsible and safe management of spent fuel and radioactive waste (NaPro) takes the year around 2050 as a basis for the commissioning of a disposal facility [95]. Taking into consideration the duration of disposal for all SNF and HLW, the disposal process will end between 2080 and 2130 depending on disposal throughput. Thus, an extension of the storage duration of spent fuel in casks by an additional period of 40 to 80 years might be necessary, at least for a number of casks [96]. According to its discussion paper on SNF and other heat generating radioactive waste, the ESK likewise expects temporary dry storage for periods up to 100 years [97].

Based on the time frame given in the NaPro for spent fuel and radioactive waste management [95], it is to be assumed that until the expiry date of the licenses for dry storage facilities an operational disposal facility will not be available. Therefore, extended storage at the sites beyond the licensed period of 40 years is most likely but will be limited to the necessary period. As stipulated in § 6 (5) of the German Atomic Energy Act (AtG), a license may only be renewed on imperative grounds and after this issue has been discussed in the German parliament (Bundestag).

According to the German guidelines, DPCs needs to comply with the relevant safety functions regarding the following protection objectives:

- Safe enclosure of the radioactive inventory;
- Avoidance of unnecessary radiation exposure by shielding ionizing radiation;
- Safe removal of decay heat;
- Guarantee of sub‑criticality.

Not only do these safety functions need to be guaranteed under normal operating conditions, operators have to take into consideration external natural impacts such as storm, rain, snow, frost, lightning, flooding, landslides, and earthquakes, as well as human impacts from outside such as the effects of blast waves caused by chemical explosions, fires, mines caving in, and aircraft crashes.

In addition, the impact of site specific conditions such as interactions with a neighboring power plant, which could include, for example, the collapse of structures, need to be considered as far as debris from such events may affect the storage facility.

All these conceivable circumstances, which could occur during the operation of a dry SNF storage facility, have to be accounted for. Furthermore, the transportability of the casks has to be ensured at all times during the entire storage period.

In Germany, proofs of safety have been provided for a storage period of 40 years; this period has been applied for and granted in storage licenses to date. Within the licensing procedures, all safety related operational aspects have been completely evaluated and confirmed. The durations of storage licenses rely on are limited for administrative reasons, not due to physical or technical constraints. By imposing further conditions at a later stage during the operating lifetime, the competent authority may demand adaptations of the facility to comply with the state of the art in science and technology as far as this is necessary to fulfil the safety requirements.

The concept and design of SNF storage facilities include constructive and administrative measures to protect the public and the facility staff against hazards arising from the release of radioactive substances and ionizing radiation. The license holder is responsible for the planning, performance, and control of the effectiveness of these measures. The latter are oriented toward their significance for preventing damages caused by radiation exposure. The nuclear licensing procedure specifies the nature and scope of initial inspection and, where necessary, recurrent inspections by the supervisory authority, which also monitors compliance with the measures.

An extension of the current license for the storage of SNF requires a new license based on reliable data for the safety analysis. Currently, no regulatory specifications have yet been made pertaining to extended storage. In order to deal with the issues relevant to extended storage at an early stage, the ESK has recommended the implementation of a periodic safety review and ageing management (AM). The obligation for operators to carry out AM during the operational phase of the storage facility for SNF and HLW was first anchored in Ref. [56] and specified in greater detail by the ESK's 'Guidelines for Performance of Periodic Safety Reviews and on Technical Ageing Management for Storage Facilities for Spent Fuel and Heat Generating Radioactive Waste'[97]. The guidelines approved in 2022 are the outcome of a two year review phase that took accounted for the results of a practical test performed in the central storage facility at Gorleben and the onsite storage facility at Lingen NPP. Its overall aim is to confirm that the dry storage facilities are in compliance with their storage licenses, and to manage ageing effects.

The periodic safety review (PSR) is a holistic verification of the safety status at regular intervals of ten years, whereas AM includes continuous control of ageing effects during storage. The focus is on defining AM measures for safety relevant systems and compounds, as well as buildings accommodating safety relevant systems, and evaluating the operational experience of the storage facilities systematically.

Safety relevant systems and compounds are categorized according to their relevance to protection goals. For safety relevant compounds of DPCs with the highest requirements for quality, no failure is permissible. Ageing effects are controlled by taking them into account in the design concept. Accordingly, technical AM measures are limited to accessible cask areas, whereas non-accessible cask areas and inventories are not covered by the ESK guidelines.

PSR and AM interact with each other. Outcomes and findings from AM are being incorporated into PSR. Based on the findings from PSR, AM measures are reviewed, which may result in updated or additional AM measures.

At present, there is the expectation that the currently established concept of dry storage (DPCs and storage building) maintains its safety functions even for considerably longer periods than 40 years. This assumption needs to be confirmed by an explicit proof according to the state of the art in science and technology. The additional data required for the maintenance of operational safety, as well as for extended storage, can be generated both from the systematic evaluation of operating experience of the existing facilities during storage within the scope of a PSR and AM, and from additional investigation programmes.

National R&D funding programmes addressing nuclear waste management are initiated by the German government (especially the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU), the Federal Ministry for Economic Affairs and Energy, and the Federal Ministry of Education and Research). For example, the supreme supervisory authority, BMU, has initiated precautionary projects in which basic information and data on national and international experience are compiled to assess the safety issues related to the extended storage of fuel assemblies at an early stage, and to be able to make competent assessments of corresponding concepts and strategies for their future storage.

I.4. GUIDANCE ON AGEING MANAGEMENT: JAPAN

In Japan, ageing management of spent fuel storage facilities is generally stipulated within acts and ordinances for at reactor (AR) and away from reactor (AFR) storage, respectively. In addition, the Nuclear Safety Commission and the Nuclear and Industrial Safety Subcommittee of the government issued an acknowledgement and a report on metal cask storage. Related technical details are addressed by the Atomic Energy Society of Japan (AESJ). The following describes the abstracts of these documents.

I.4.1 **Act and ordinances**

The ‘Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors’ stipulates that maintenance of storage facilities needs to conform to the technical standards specified by the Nuclear Regulation Authority (NRA).

I.4.1.1 At reactor storage

The ‘Ministerial Ordinance for Commercial Nuclear Power Reactors concerning the Installation, Operation, etc. (Article 82–1–6)’ requires ageing evaluation of reactor facilities including storage facilities 30 years after starting operations. The evaluation is repeated every 10 years or less afterwards.

I.4.1.2 Away from reactor storage

The ‘Ordinance on Activity of Interim Storage of Spent Fuel (Article 32)’[98] requires annual self–inspection. It (Article 35–2) also requires ageing evaluation in 20 years after starting its operation. The evaluation has to be repeated every 10 years or less. The ‘Ordinance on Standards (Article 15)’ requires that a metal cask ensures the integrity of spent fuel during storage.

I.4.2 **Acknowledgement, report, and codes**

This section covers acknowledgement, report and codes for AR and AFR storage.

I.4.2.1 At reactor storage

The Nuclear Safety Commission [99] acknowledged that the first of a kind metal (bolted lid) dry storage cask needs to be inspected to assess the containment performance of the metallic gasket and other components, as well as the spent fuel cladding integrity, during storage.

The AESJ issued 'Code on Implementation and Review of Nuclear Power Plant Ageing Management Programs (2015)'[100] briefly addresses the mechanisms of ageing in metal cask components, and their management.

I.4.2.2 Away from reactor storage

The Nuclear and Industrial Safety Subcommittee of the Advisory Committee for Natural Resources and Energy, Interim Storage and Transportation WG (2009) issued a report on 'Long term Integrity of the Dry Metallic Casks and their Contents'[101]. It discussed understanding ageing mechanism and management of spent fuel and metal casks in detail. For spent fuel, it discusses chemical, thermal, mechanical, and radiation degradation and management. For metal casks, it also addresses the management (design) of ageing for systems, structures, and components (SSCs). It includes results of visual inspections of spent fuel assemblies and metal gaskets that have been stored and used in the metal casks. It also discusses the ageing and management of the main body, basket, and lid structure of the cask.

Based on the above discussions, post-storage transport was addressed. The concept of alternative inspection without opening the dual purpose metal cask was recommended by the working party to assure sub-criticality, spent fuel integrity, and atmosphere and pressure in the cask.

The AESJ issued 'Code on Standard for Safety Design and Inspection of Metal Casks for Spent Fuel Interim Storage Facility (2010)'[102], which addresses ageing of metal casks in terms of heat, radiation, and corrosion (attachment G reference [102]). In addition, it addresses inspections with respect to safety performance and the structural strength of dual purpose casks. The inspections specify locations (factory, nuclear power plant, and interim storage facility) and safety functions (containment, shielding, sub-criticality, and heat, attachment S reference [102]).

I.5. GUIDANCE ON AGEING MANAGEMENT: PAKISTAN

Pakistan's nuclear programme focuses on using nuclear technology in a demonstrably safe, reliable, and cost effective manner over the long term, for the benefit of society and stake holders. Efforts are being consolidated for the indigenous enhancement and development of the nuclear power programme in the country.

The Pakistan Nuclear Regulatory Authority (PNRA) is entrusted with the responsibility to control, regulate, and supervise all matters related to nuclear safety and radiation protection in Pakistan. It is empowered to develop rules and regulations, and issue guides for nuclear safety and radiation protection; develop and execute policies and programmes for the protection of life, health, and property against the risk arising from ionizing radiation; and regulate the nuclear and radiation safety aspects of nuclear installations, radiation facilities, and spent fuel storage systems.

PNRA regulates spent fuel management through a combination of regulatory requirements that apply to both spent nuclear fuel casks and spent nuclear fuel storage. In areas where PNRA regulations are not available, such as ageing management of dry storage systems, the latest U.S. Nuclear Regulatory Commission regulations are followed. Alternatively, the PNRA may also choose to follow the latest revisions of the applicable IAEA safety standards applicable to ageing management.

I.6. NATIONAL GUIDANCE ON AGEING MANAGEMENT: SPAIN

The strategy for SNF management in Spain is established in the General Radioactive Waste Plan (GRWP), of which the 6th Revision (dated 2006) is in force. This considers the development of a Centralized Temporary Storage (CTS) facility to gather all the high level radioactive waste (including SNF) generated in the seven operating NPPs (Almaraz I and II, Asco I and II, Vandellos II, Cofrentes and Trillo) at a site selected by the Spanish Government. In July 2018, the Government put the licensing process on hold for the facility in order to review the GRWP. The review forms the basis of the 7th Revision of the GRWP, which is awaiting governmental approval. Within this update, an NPP shutdown schedule was agreed between ENRESA and the Spanish utilities. The CTS facility will bridge the gap between shutdown of the NPPs and implementation of deep geological disposal. Due to the delay in commissioning of the CST facility, pool saturation and a lack of further re–racking possibilities, most NPPs have established specific away from reactor onsite (AFR–OS) dry storage facilities, that are currently implementing ageing management programmes.

In November 2018, Royal Decree 1400/2018 on Nuclear Safety Regulation for nuclear facilities [103] entered into force. This new regulation gathers the principles and foundations of the Council Directive 2014/87/Euratom of 8 July 2014 amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations.

Referring to ageing management, Royal Decree 1400/2018 on Nuclear Safety Regulation for Nuclear Facilitiesstipulates the following:

- Section 21. Design requirements for IS SSCs: “In the design of important to safety SSCs (Structures, Systems and Components), the maintenance, testing and aging management needs to be taken into account” [103];
- Section 33. Ageing Management: “The licensee needs to carry out an Ageing Management Program for the important to safety SSCs according to the specific applicable regulations, in order to ensure the maintenance of their functions under the conditions foreseen in the design bases during the period of operation of the facility” [103].

On the other hand, the Spanish Regulator, Consejo de Seguridad Nuclear (CSN), has issued three sets of Safety Instructions (IS, Instrucciones de Seguridad) applicable to the lifetime management of dry storage systems. These ISs and a summary of their most relevant clauses are listed below in Table 11.

TABLE 11. OVERVIEW SPANISH REGULATOR SAFETY INSTRUCTIONS

Safety instruction number	Title	Date of issue
20	Safety requirements related to spent fuel storage casks	January 2009
26	Basic nuclear safety requirements applicable to nuclear installations	June 2010
29	Spent fuel and high level radioactive waste storage facilities	November 2010

(i) CSN, SI–20,safety requirements related to spent fuel storage casks, January 2009 [104]

"This instruction is intended to establish the requirements of nuclear safety and radiological protection to comply with the design of spent fuel storage casks to be used in authorized temporary storage facilities and to define the necessary documentation to ensure that the interdependencies between the design, the manufacture and the use are carried out properly" [104].

Related to ageing management, SI–20 states the following:

- Paragraph 3.6.3. "Depending on the design bases and the fuel properties, the SSCs ageing will be considered establishing, if necessary, a program of maintenance, tests and inspections. The results from this program will serve as a base to the Periodic Safety Review" [104];
- Paragraph 5.11. "The licensee will develop the manual for maintenance, tests and periodic inspection, the surveillance system and the SSC ageing program, in accordance with the basis the Safety Analysis Report, that the user will implement at the installation. The results of the maintenance, tests and periodic inspections will be evaluated and registered" [104];
- Paragraph 5.12. "The licensee has to periodically review the manual for maintenance, testing and inspections to incorporate lessons learned from the experience, especially from incidents of secondary activities such as the preparation of the maintenance and the tests" [104].

(ii) CSN, SI–26,basic nuclear safety requirements applicable to nuclear installations, June 2010 [105]

"This instruction establishes the basic nuclear safety requirements applicable to:

- Any enrichment facility, nuclear fuel making facility, nuclear power plant, reprocessing facility, research reactor facility, spent fuel storage facility;
- Radioactive waste storage facilities found in the same enclosure and are directly related to the facilities listed in (a)" [105].

Related to ageing management, SI–26 states the following:

- Paragraph 7.19. "The nuclear facility needs to have an Ageing Management Program that identifies the degradation and ageing mechanisms of the SSCs important to safety, specifying their possible consequences, as well as determining their expected lifetime and the activities necessary to maintain their operability and reliability” [105].

(iii) CSN, SI–29,spent fuel and high level radioactive waste storage installations, November 2010 [106]

“The purpose of this instruction is to establish the basic safety criteria and requirements to be complied related to the design, manufacture, construction, testing, operation and safety analysis of the nuclear facilities for temporary storage of spent fuel and high radioactive waste” [106].

Related to lifetime management, this SI–29 states the following:

- “The design approval will have a validity period of 20 years’ maximum, counting from the date of expedition of the corresponding resolution. The application of extension or renewal of the Authorization should be done with at least one year of anticipation from the limit date, and it will be accompanied by a justification that the fuel storage has not affected adversely the structures, systems and components important to safety of the cask, according to the applicable requirements” [106];
- Paragraph 3.4.1. “The design of the temporary storage facility should take into account the following aspects:
  - The expected life of the facility. The design of the spent fuel and high-radioactive waste storage facilities should take into account the intended design lifetime to prevent and correct potential degradation phenomena in the long term, caused by the effect of corrosion, erosion and ageing of the SSC important to safety;
  - The inventory envisaged and a reserve capacity or justify that it is available otherwise, to allow repositioning manoeuvres with a view to maintenance, inspection, recovery or refurbishment of casks or packages of spent fuel and high radioactive waste that presents signs of degradation” [106];
- Paragraph 3.4.11. “The ageing of structures, systems and components will be considered, and the physical and chemical properties of the spent fuel and the radioactive waste, establishing, when necessary, the maintenance, tests and inspection forecasts. The inspection and maintenance program will consider the environmental conditions of the installation and the SSC ageing. The results obtained from this program should be used to review the adequacy of the design in the facility life cycle” [106];
- Paragraph 3.5.6. “The equipment and systems required for maintenance, periodic testing and inspection programs will be identified to ensure their proper operation” [106];
- Paragraph 4.4. “During the expected storage period, the holder will implement a lifetime program of the systems, structures and components defined as important to safety and define the intervals of preventive or corrective maintenance, periodic tests and inspections necessary to maintain safety of the temporary storage through reliability and required qualification” [106];
- Paragraph 4.5. “Surveillance, periodic testing and inspections of stored material and the facility needs to be carried out in accordance with a program based on written procedures, which needs to be available before the beginning of the operation of the facility” [106];
- Paragraph 4.6. “The results of the inspection, monitoring and maintenance programs of the stored material and the facility needs to be registered. These programmes should

be reviewed at appropriate intervals to incorporate lessons learned from experience, especially the incidents of secondary activities such as the preparation of maintenance and testing” [106].

I.7. GUIDANCE ON AGEING MANAGEMENT: UNITED STATES OF AMERICA

The U.S. NRC regulates the licensing, construction, and operation of independent spent fuel storage installations (ISFSIs) under Title 10 of the U.S. Code of Federal Regulations, Energy, Part 72 (10 CFR 72) [10]. When the regulations in 10 CFR 72 were created around 1980, the U.S. NRC chose 20 years as the term for a specific ISFSI license and later, for a dry storage system (DSS) certificate of compliance (CoC). However, shortly after the turn of the century, with still no federal programme for removing spent nuclear fuel (SNF) from reactor sites in the near term, it became apparent that most DSSs would remain in service at the ISFSIs beyond the initial 20 year license term. Operation beyond the original license term requires the specific ISFSI license or the DSS CoC (used at a general licensed ISFSI) to be renewed. The requirements for renewal of a specific ISFSI license or a DSS CoC include addressing age related degradation of structures, systems, and components that perform or support safety functions.

In May 2011, a rule change to 10 CFR 72 became effective, extending the terms for both general and specific ISFSI licenses and DSS CoCs from a term not to exceed 20 years to a term not to exceed 40 years. This 40 year maximum applies to both initial license/CoC terms and renewal terms. Beginning in 2014, industry and the U.S. NRC engaged in a collaborative effort to bolster the regulatory framework in this area, which has resulted in the development of additional regulatory guidance defining an operation based or ‘learning’ approach to DSS ageing management. The U.S. Department of Energy also participated in this effort through used fuel disposition research and development, which began in 2009 with a focus on extended storage and subsequent transportation.

Learning (or operations based) ageing management is a concept whereby ISFSI licensees and CoC holders pay close attention to operating experience, research, inspections, and monitoring related to age related degradation in the period of extended operation. ISFSI licensees and CoC holders use this information to learn more about age related degradation trends and modify ageing management programmes (AMPs) appropriately. To provide a structured process to implement learning ageing management, the Nuclear Energy Institute (NEI) created guidance that included the ‘tollgate’ process [49]. Such tollgate assessments allow ISFSI licensees to apply lessons learned from a wide variety of sources to their specific ISFSI AMPs. Because materials ageing is not unique to USA nuclear facilities, the tollgate process encourages ISFSI licensees to use international and non–nuclear age related degradation information in their tollgate assessments. To support efficient development of input for tollgate assessments and avoid duplicative effort, NEI also facilitated the development of the Aging Management Institute of Nuclear Power Operations database (AMID). AMID is a clearinghouse for ISFSI licensees and cask vendors (CoC holders) to collect ageing related information for the industry to search and use in preparing tollgate assessments. The U.S. NRC has endorsed the NEI guidance, with clarifications [107], and both the tollgate process and AMID are considered effective concepts for managing DSS ageing into the future and effectively implementing the concept of learning ageing management.

The U.S. NRC published a standard review plan for renewal of specific licenses and CoCs for dry storage of spent nuclear fuel [21]. This standard review plan provides guidance for the

NRC's safety review of renewal applications for specific licenses of ISFSIs and CoCs of dry storage systems. It also provides guidance on the review of the general information, scoping evaluation, and ageing management reviews included in a renewal application, as well as information on review of time limited ageing analyses (TLAAs) and AMPs, including learning AMPs that consider and respond to operating experience. The guidance provides considerations for CoC renewals and the general license framework, including guidance on general licensees' implementation of AMPs. The NRC expects to periodically revise and update this standard review plan to clarify the content and include new information, knowledge, and experience regarding ageing management considerations.

The MAPS report [15] provides detailed technical guidance for U.S. NRC technical reviewers and preparers of license/CoC renewal applications and establishes a generic technical basis for the safety review of renewal applications for specific ISFSI licenses and DSS CoCs. The MAPS report evaluates known ageing degradation mechanisms to determine whether they could affect the ability of DSS components to fulfil their safety functions in the 20 to 60 year period of extended operation. The guidance also provides examples of ageing management programmes that are considered generically acceptable to address the credible ageing mechanisms and effects to ensure that the design bases of dry storage systems will be maintained. An applicant for a license or CoC renewal may reference the information in the MAPS report to support its ageing management review and proposed ageing management programmes.

The USA has relied extensively on dry storage to meet the defueling needs of its operating fleet of power reactors since 1986. As of the end of 2020 the USA had loaded 3370 DSSs at 72 DSFs [6]. As many of these DSSs are now in their third or fourth decade of service, the USA has gained significant experience with AMPs. The success of these programmes highlights the importance of information exchange between the many involved parties.

Dry storage of SNF in the USA began as an unexpected solution to an unanticipated problem. The USA commercial nuclear reactor fleet was built under the presumption that the federal government would begin removing used fuel for reprocessing before the spent fuel pools inside the reactor facilities filled up. During the first two decades of the industry's history, the USA was actively working on the development of reprocessing facilities. But in 1977, USA President Jimmy Carter announced that the United States would defer reprocessing indefinitely [108].

Reprocessing Policy in the USA was subsequently replaced with a plan for direct disposal outlined in the 1982 Nuclear Waste Policy Act [109]. The Act's 1998 deadline for the removal of spent fuel from reactor sites would not be soon enough avoid the need for additional storage at several plants. The large scale buildout of dry storage became inevitable. Because the USA began loading DSSs in 1986, the U.S. NRC began considering license renewals in 2005. As the USA repository programme continued to experience delays before eventually being discontinued in 2011, the dry storage buildout expanded to nearly every USA commercial reactor site.

To date, U.S. NRC has renewed site specific licenses and certificates of compliance (CoCs), a license covering a given DSS design which may be deployed at multiple sites, for 1168 installed DSSs. Most of the USA experience with DSS ageing management is driven by the need to collect data to support license renewal. Most of the USA DSS designs either have license/CoC renewal applications under review at U.S. NRC or will have within the next 10 years.

Because over 90% of the DSSs loaded in the USA are modular systems where the fuel is placed in a welded stainless steel canister that is stored inside a concrete overpack or horizontal storage module, most ageing management activity in the USA has been focused on these types of systems. They are modular because the canister is placed in a transfer cask when removed from the pool, then a storage cask/module while being stored outdoors on a concrete pad, and then a transportation cask when shipped off site for disposal or subsequent storage at a Consolidated Storage.

APPENDIX II

POST–STORAGE INSPECTION CONCEPT

This summary explains the development of a potential concept for inspections after storage for DPCs in Japan. It was developed considering the specific need for inspections after storage and before transportation.

II.1. CURRENT INSPECTION REQUIREMENTS

Transport after ageing during storage requires careful inspection of each DPC [110]. When SNF is transported, generally the items displayed in Table 12 are inspected. Those inspections support the gap analysis, evaluating the status of the DPC after storage against its condition when placed in storage.

TABLE 12. EXAMPLES OF PRE–SHIPMENT INSPECTION AFTER STORAGE PERIOD

	Inspection item	Check	Measurement
(a)	External visual inspection	Corrosion, degradation of sealing material	
(b)	Leak tightness in relation to the transport regulatory requirements		✓
(c)	Pressure retaining		✓
(d)	Dose rates in relation to transport regulatory limits and values stated in the PDSR		✓
(e)	Subcriticality	Documentation and AMP reports	
(f)	External surface temperature and values stated in the PDSR		✓
(g)	Lifting and transport capability	Trunnions and attachment points	
(h)	Condition of contents	Documentation and AMP reports	
(i)	Surface contamination in relation to transport regulatory limits		✓
(j)	Storage environment records to demonstrate compliance with the storage conditions stated in the PDSR	Documentation	

Of these, items (b), (e), and (h) are difficult to perform after storage and the PDSR will endeavour to justify the performance characteristic in these areas without the need for an intrusive inspection regime that involves opening the cask. This may be a time constraint for the DPC storage facility, as the time to remove the DPCs from the storage facility may be due to unplanned events such as facility safety or security concerns at some future date. Consequently, if it is not the intention to provide the storage facility throughout its operational life with the capability to remove cask lids and inspect the contents, the plans and timescales to provide such a capability need to be developed and agreed with the licensing authorities as part of the operational plan for the storage facility. Alternate means of visual inspection can be performed by verification as described below, subject to acceptance and approval by the competent authority, which in combination with the ageing management evaluation, give assurance that the DPC can be safely transported. All this would be part of the PDSR submitted for package design approval certification for transport and storage. Taking into account these considerations the following potential concept was developed.

II.2. CONCEPT FOR ALTERNATIVE MEANS OF VISUAL INSPECTIONS

To confirm safety of transportation at the end of the storage period, the integrity of the DPCs and contents need to be assured. This could be achieved, for example, by a visual check of the condition of the contents and baskets that includes opening the lid of the DPCs, or by inspecting the atmosphere inside the DPCs in the storage facility prior to shipment. Taking radiation protection considerations into account, cask opening is an undesirable operation. This action is undesirable not only from the viewpoint of reduction of risks such as workers' exposure or release of radioactive materials but also may increase other risks caused by incidents during handling of the DPC package. Consequently, when the same level of safety in relation to spent fuel condition can be assured through alternative inspection means, it is more desirable to perform inspections in that manner.

The following 'Subcriticality inspection' is an example of the alternative inspection methods related to the above mentioned inspection items, subject to acceptance and approval by the competent authority taking into account the ageing and gap analysis as mentioned in the previous Section II.1.

Confirmation of subcriticality: There is reasonable assurance that there is no significant deterioration if the following points are confirmed:

- The baskets are manufactured in compliance with the design specified in the PDSR;
- Moisture is removed and inert gas is filled on completion of loading in accordance with the requirements of the PDSR;
- DPC packages comply with the PDSR referenced in the package design approval certificate;
- The inert atmosphere of the basket (neutron absorbing material) has been maintained during storage.

Therefore, when DPCs are shipped from storage facilities that have no fuel reloading equipment, subcriticality inspection during the pre-shipment inspection can be substituted by the documents which prove the above listed items.

APPENDIX III

RESEARCH CONTRACTS REPORTS

The following summaries the findings from the research contracts undertaken during the CRP.

III.1. ARGENTINA

The following summarises the progress reports from Argentina.

III.1.1. First progress report (2017).

There are two Dry Storage Facilities in Argentina for Spent Fuel Elements (SFE) used in power reactors. The first one is located at the site of Embalse NPP (ASECQ–CNE) and operates since the early nineties. The other is currently under construction at the site of the Atucha–I NPP (ASECQ–CNA–I).

(a) Dry storage at CNE.

This installation consists in an array of concrete silos lined with carbon steel at the interior. The 0.5 m long CANDU spent fuel bundles are placed into cylindrical metallic containers which are placed one on top of the other up to the top of the silo, which can hold up to nine of these canisters. Fig. 23 left shows a view of the site and construction details and depicts on the right a metallic canister which holds SFE. ASECQ–CNE was intended to hold SFE for a period not longer than a few decades; but since no decision was so far taken as for their final disposition, an extended storing time will be needed. Considering that CNE is undergoing a life extension process, this facility is expected to be active for a period close to a century at least. Hence, a study of possible materials degradation processes that could take place in this installation is necessary in order to evaluate its capability to withstand long term operation and safe SFE storage.

(b) Dry storage at CNA–I

The ASECQ–CNA–I is at present being built. The storage site consists in a dry concrete pool inside which up to 2016 of 6 m long SFE can be accommodated in 112 vertically hanging silo units containing 18 SFE each. It is intended to last for at least 50 years. A programme is currently running in order to ensure that its structural components will not suffer of any material degradation process that could hamper its ability to safely keep the stored SFE.

III.1.1.1. Evaluation of material degradation

(a) Dry storage at CNE

Two materials are considered when looking for degradation mechanisms in ASEC–CNE; they are the concrete used for the shielding cylinder and the carbon steel of the silo liner and lateral canisters wall.

(i) Concrete

The main issue related with concrete degradation is the corrosion of the reinforcement bars. In order to evaluate this, insertion of corrosion sensors developed at the Corrosion Laboratory of CNEA could has been proposed. Taking into account the possible effect on the shielding properties of the concrete wall, this activity has to be authorised by the ARN, who would only permit to work in silos being constructed (not those which are already loaded with SFE) and carried out with the collaboration of the NASA Company, which operates NPPs.

(ii) Carbon steel

Several activities have been carried out in relation with the study of carbon steel corrosion in the ASECQ–CNE Dry Storage Facility:

- Calculation of general corrosion in various conditions;
- Corrosion Tests in welds between CS and SS;
- In site Visual Inspection of canisters.

The measured corrosion rates of carbon steel exposed to various atmospheric conditions, as reported in [111] are shown in Fig. 49.

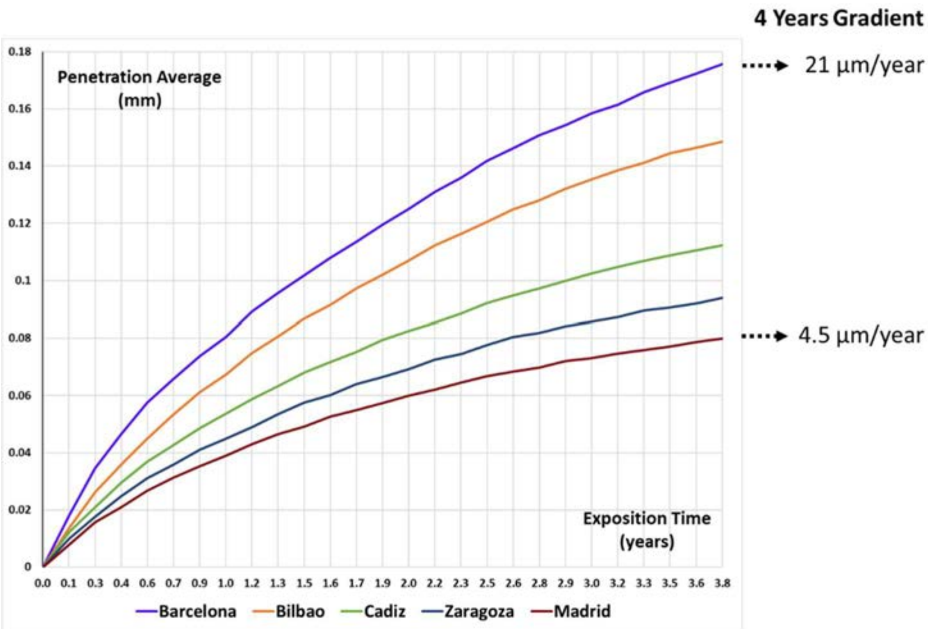


FIG. 49. Uniform Corrosion rate of carbon steel in various environments.

For the sake of a conservative calculation, the slope at four exposition years can be taken to extrapolate the long term corrosion level, knowing that this rate will be decreasing with time. Considering the worst case (humid, saline environment at Barcelona), the average penetration would reach a value of 2.1 mm, what constitutes a safe situation for a 9.5 mm thick wall.

A different situation could take place if there is some water ingress to the interior of the silos due to any eventual loss of tightness. For this reason, corrosion tests were performed in samples of the same material of the canisters, including welds. Test conditions were as follows:

Welds between CS F–24 and SS 304–L;

– Filler material: ER 309–L;
– Environment: pure water at 343 K;
– Immersion time: 73 days.

In Table 13, the results of this exercise are displayed. The expected corrosion level near the welds is found to be larger than the acceptable limits.

TABLE 13. RESULTS OF IMMERSION TESTS

Distance from weld	Corrosion penetration	40 years corrosion
> 6 mm	20 µm/year	0.8 mm
< 6 mm	1500 µm/year	60 mm

This activity is being continued with the purpose of evaluating the influence of weld area, water quality and temperature.

Given these results, a visual inspection of the facility would be necessary in order to establish the real situation and verify the existence of water. With this purpose, an attempt was made to introduce an endoscope into a mock up silo, through the ventilation lines, to get a close view of the canisters. This particular silo belongs to the first batch put into operation, is full with canisters but is not loaded with SFE. The first step was to introduce a testing cable to check for irregularities and obstructions. Then, the endoscope was inserted all the way to the point in which an image of the canister could be obtained from the ending point of the inferior ventilation line. As can be appreciated, there are no relevant corrosion associated features, only the expected normal oxidation. However, the welded region could not be reached due to the particular endoscope geometry. Other alternatives for a more comprehensive visual inspection are being studied.

(b) Dry storage at ASEICNA–I

Two materials are considered when evaluating components durability in this site, stainless steel and concrete. The degradation issues taken into account are the possibility of having localised corrosion in SS welds and the reinforcement bars corrosion.

(i) Stainless Steel Components

As for this issue, it is considered important that the welding procedure comply with all the specifications required for the case; for this reason, permission to supervise this practice has been requested, which is pending yet.

(ii) Concrete Structures

The main issue to be aware of with respect to the durability of concrete structures is the possibility of reinforcement bars corrosion, what can be assessed with the use of corrosion probes. Some that have been developed at the Corrosion Laboratory of CNEA [17] and have been used for various purposes, like monitoring of new and pre–existing structures and selection of best cement candidates. Two activities have been carried out in the ASEICNA–

I:

– Monitoring of a silo unit mock–up;
– Monitoring of the ICQ–CNA–I main structure.

This 1:1 model scale has been built with the purpose of simulating the SFE storage conditions, and measure thermal parameters before real fuel is transported. Two concrete corrosion sensors have been introduced in it, one in the inner side of the wall and the other in the outer side, to monitor the two different conditions.

Each sensor contains several electrodes which permit to measure several parameters, namely, Corrosion Potential, Corrosion Current, oxygen flow, chloride content, concrete resistivity and temperature. All these variables have to stay inside their respective safety ranges in order to ensure proper bar behaviour, as stated by specific standards and other recommendations [54, 55, 112]. Figure 29 depicts the reading operation on the silo unit mock–up concrete shielding, using probes embedded into the cement.

Figure 50 shows the evolution of the measured parameters during a period of several months. As it can be seen, all variables have been evolving to a safe condition in which the reinforcement bars will not suffer corrosion, thus ensuring the structural safety of the concrete structure. This activity has been discontinued for the time being, waiting for a decision to be made with respect to the final destination of the component.

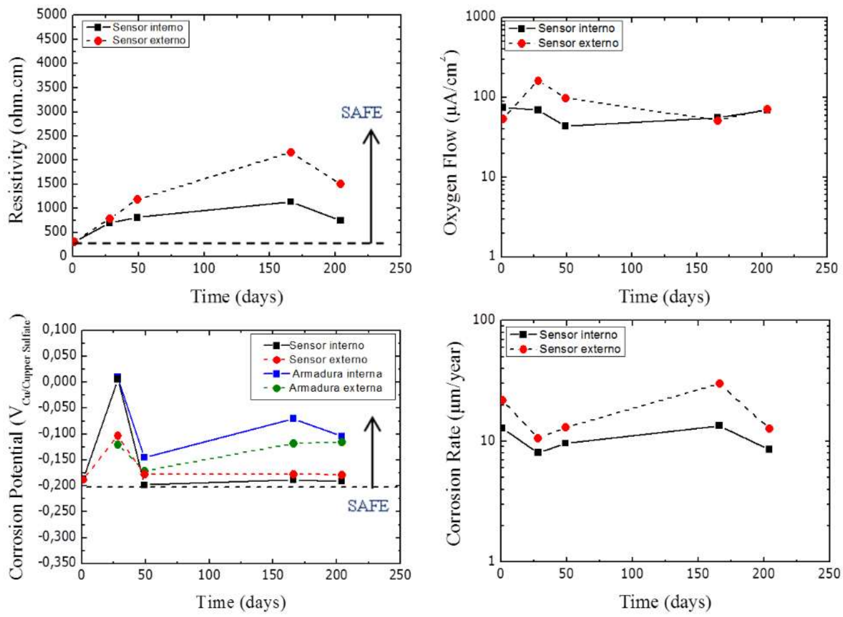


FIG. 50. Evolution of the various parameters measured on the silo unit mock–up (courtesy of CNEA).

In Fig. 30 the location of inserted probes can be seen. There are eight of them, four on the internal side of the shielding wall and four on the external one. They are distributed at two different heights in order to monitor the effect of conditions at different distances of the ground.

The results of the main ASEC–CNA–I structure can be seen in Fig. 51.

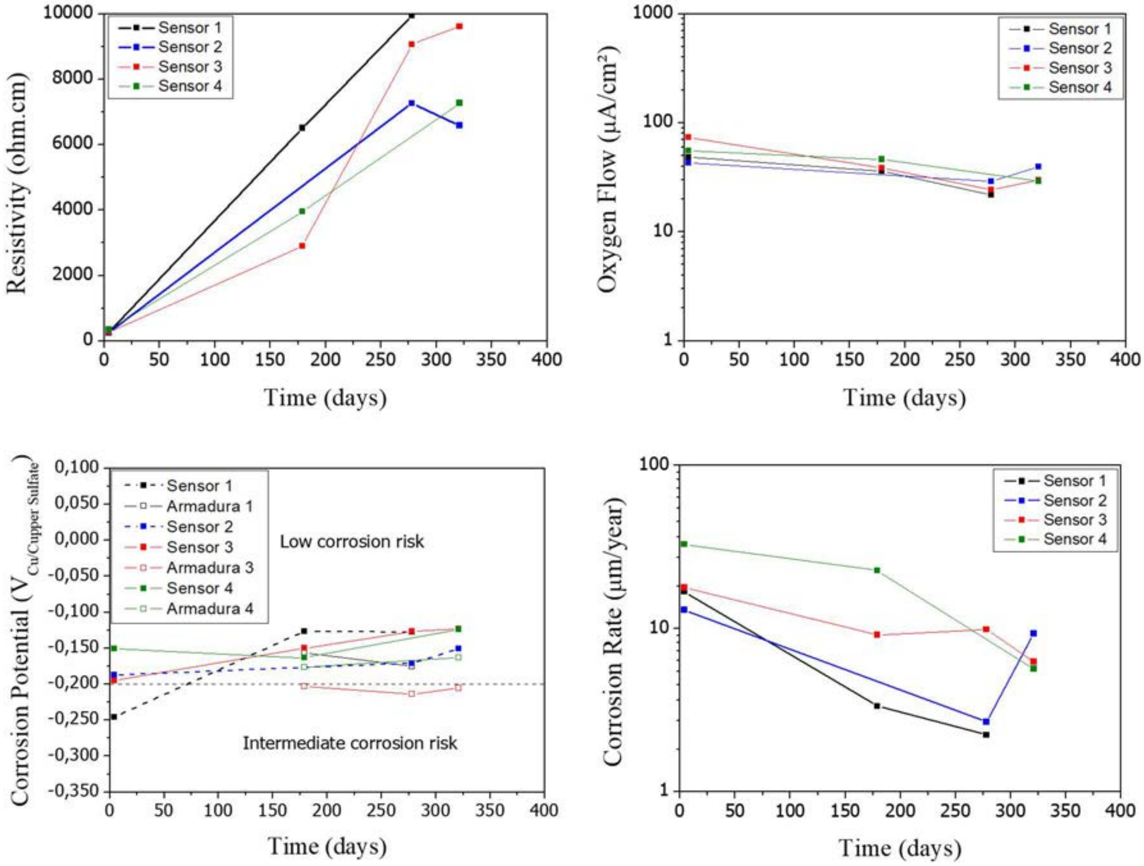


FIG. 51. Evolution of parameters in the main ASEC–CNA–I structure (courtesy of CNEA).

After almost a year, all measured parameters were entering the low corrosion risk zone. This activity is being maintained, totalising at present some 2 years.

III.1.1.2. Pending issues

The following sections cover the issues still to be resolved.

(a) Evaluation of ASECQ–CNE

Although the Regulatory Authority has not approved the insertion of reinforcement bars corrosion monitoring probes in the silos already loaded with SFE, there is still the possibility that their installation in new silos to be constructed be permitted. Laboratory tests will be continued to measure the corrosion rate in air with various moisture contents, study the influence of weld area, water quality and temperature. Finally, alternatives to visual inspection are being considered in order to carry on a sound assessment of the situation of the canisters.

(b) Evaluation of ASECQ–CNA–I

The reinforced concrete main structure monitoring is being continued, while a decision on the future of the mock–up silo unit is being awaited, to decide upon. The process to obtain authorization to supervise the welding of stainless steel components of the facility is currently ongoing.

III.1.2. Second progress report (2018)

The two Dry Storage Facilities for Spent Fuel Elements (SFE) burned in power reactors existing in Argentina were described in detail in the First Progress Report. The main durability related concerns were enumerated, as well as the actions taken to evaluate them. During the second period, the following activities were carried out:

– Sections of steel canisters coming from the ASECQ–CNE facility were subjected to corrosion tests to simulate the worst possible environmental conditions inside the silos.
– Following the advice obtained during the 1st RCM, steps were taken to address the feasibility study of a visual inspection of the silo's interior, using a videoscope.
– Monitoring of the main concrete structure at ASECQ–CNA–I Dry Storage Site was continued.
– Actions were continued before the ARN to obtain authorization for the insertion of concrete corrosion sensors in the silos of the CNE.
– Inspection of stainless steel (SS) components fabricated for this facility was initiated.

III.1.2.1. Description of the work carried out during the second period

The following sections provide a description of the work carried-out in the second phase of the research study.

(a) Corrosion tests of steel canisters

Figure 52 (a) shows a typical metallic container inside which the spent fuel elements (SFE) of the CNE CANDU reactor are placed. From such a container, an arc of about one third of circumference was cut, containing a weld string between SS cap and carbon steel (CS) body Fig. 52 (b).

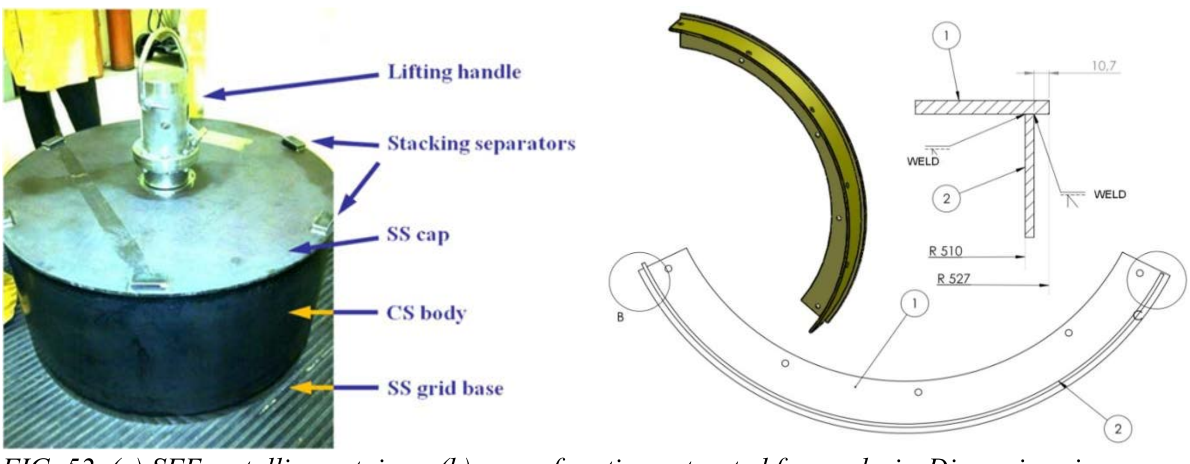


FIG. 52. (a) SFE metallic container. (b) map of section extracted for analysis. Dimensions in mm (courtesy of CNEA).

From this section, four samples of 30 mm x 45 mm x 6 mm in size were cut (‘L’ shaped), containing a weld string between carbon and stainless steel (identified as WS); additionally, another four rectangular samples were extracted from carbon steel material solely (CS), to be used as references. All pieces were weighed to 0.1 mg precision prior to the tests.

Two tests were carried out, lasting some 90 days each, trying to simulate a flood situation inside the silos. They consisted in samples immersion in pure (distilled) water at 343 K. Provisions were made to avoid loose of water, by means of a refrigerated condenser. The following procedure was followed with the coupons after completion of the exercises:

– They were photographed;
– Rinsed with distilled water;
– Brushed with a plastic tool;
– Pickled in ammonium citrate (150 g/l) aqueous solution at 346 K during 1 h, according to ASTM G1, procedure C3.4;
– Photographed again;
– They were re–weighed to calculate corrosion rate, assuming generalized corrosion.

Figures 53 and 54 show before (a) and after (b) surface cleaning, the CS and WS samples. A fair amount of corrosion products, as well as pitting, is observed. Consequently, test water became brown reddish due to accumulation of iron oxide. Resistivity decreased from 18 MΩ/cm to 10 Ω/cm.

(a)  (b)

FIG. 53. CS coupon after test before (a) and after (b) surface treatment (courtesy of CNEA).

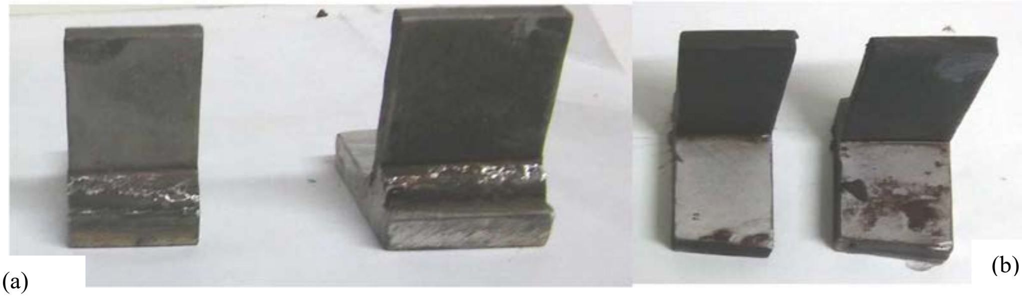


FIG. 54. Welded samples (WS) before (a) and after (a) surface cleaning (courtesy of CNEA).

Tables 14 and 15 show the results of weight determination for CS and WS respectively. In both cases, an average corrosion rate is derived. In the particular case of WS, only the CS area is taken into account for the calculations, assuming that this is the only material that dissolves.

The average corrosion rate obtained for CS and WS are 120 ± 15 µm/year and 154 ± 33 µm/year respectively. This difference would indicate a galvanic acceleration, although is computed as not relevant for an ANOVA calculation with a significance level of 0.5. Anyhow, its real implication has to be evaluated after detailed study of the morphology of attack, because it may be related with deep corrosion penetration.

TABLE 14. WEIGHT LOSS DATA AND CALCULATED CORROSION RATE FOR CS SAMPLES

Sample ID	Weight Loss (grams)	Corrosion Rate (µm/year)
CS1	0.8186	117
CS2	0.9697	139
CS3	0.8414	121
CS4	0.7108	102

TABLE 15. WEIGHT LOSS DATA AND CALCULATED CORROSION RATE FOR WS SAMPLES

Sample ID	CS Area (cm^2)	Weight Loss (grams)	Corrosion Rate (µm/year)
WS1	29.0	1.2100	198
WS2	37.3	0.9688	123
WS3	37.4	1.0513	133
WS4	39.7	1.3510	161

This work will follow with a detailed metallographic analysis of the tested samples, to establish the extent of the pitting attack.

(b) Feasibility study of a visual inspection of CNE silo interior

During the 1st RCM the case was raised of a visual inspection possibility at the ASECQ–CNE facility. In order to work into a radiation environment, the use of a device made for the occasion was considered, using disposable video cameras. At that time, testing cables and a videoscope probe were successfully introduced into an existing mock–up unit representing the actual dry storage silos, as a previous stage. Other CRP members supplied information on suitable equipment in use in other facilities. Having made contact with representatives of vendors, a study was started in order to establish their applicability. It includes a calculation of dose rate in contact with the canisters, based on the activity levels calculated for the spent fuel after a specific number of decay years, as shown in Table 16. Only the 10 most active elements out of 42 considered are shown. Using this information, the situation was modelled using Scale 6.1 Mavric Sequence, considering only caesium activity, which is about 90% of total.

TABLE 16: SFE SPECIFIC NUCLIDES ACTIVITY (IN BECQUERELS) AS A FUNCTION OF TIME.

	Time (years)				
Nuclide	1	5	10	30	50
Cs–137	8..936E14	8.147E14	7.258E14	4.572E14	2.880E14
Ba–137m	8.439E14	7.694E14	6.854E14	4.318E14	2.720E14
Am–241	1.715E12	6.683E12	1.164E13	2.246E13	2.601E13
Y–90	6.162E14	5.584E14	4.937E14	3.017E14	1.844E14
Sr–90	6.161E14	5.583E14	4.936E14	3.016E14	1.843E14
Pu–240	8.188E12	8.185E12	8.181E12	8.164E12	8.147E12
Pu–239	5.836E12	5.835E12	5.835E12	5.831E12	5.828E12
Pu–238	2.632E12	2.611E12	2.510E12	2.144E12	1.832E12
Sm–151	2.491E12	2.416E12	2.324E12	1.992E12	1.708E12
Pu–241	8.569E14	7.063E14	5.548E14	2.111E14	8.032E13
Tc–99	1.216E11	1.215E11	1.215E11	1.215E11	1.215E11

The results can be seen in Fig. 55. A Dose Rate of about 5 Sievert/h (500 Rads/h) is expected to be received by the camera/probe in contact with the metallic canister containing SFE.

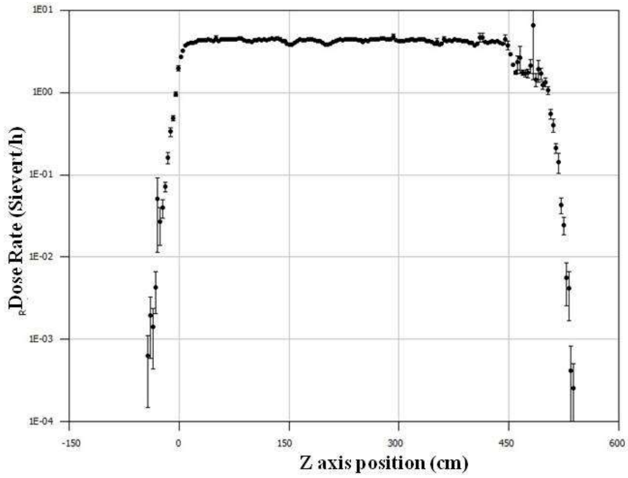


FIG. 55. Calculated Dose Rate as a function of silo height (courtesy of CNEA).

The videoscope used in the demonstration transmits images through a fibreglass line. Considering the materials resistance to radiation, it would lose 50% of transparency after receiving some 5 sieverts (Sv), what will happen in less than one hour, according with specifications supplied by the vendors, as seen in Fig 56.

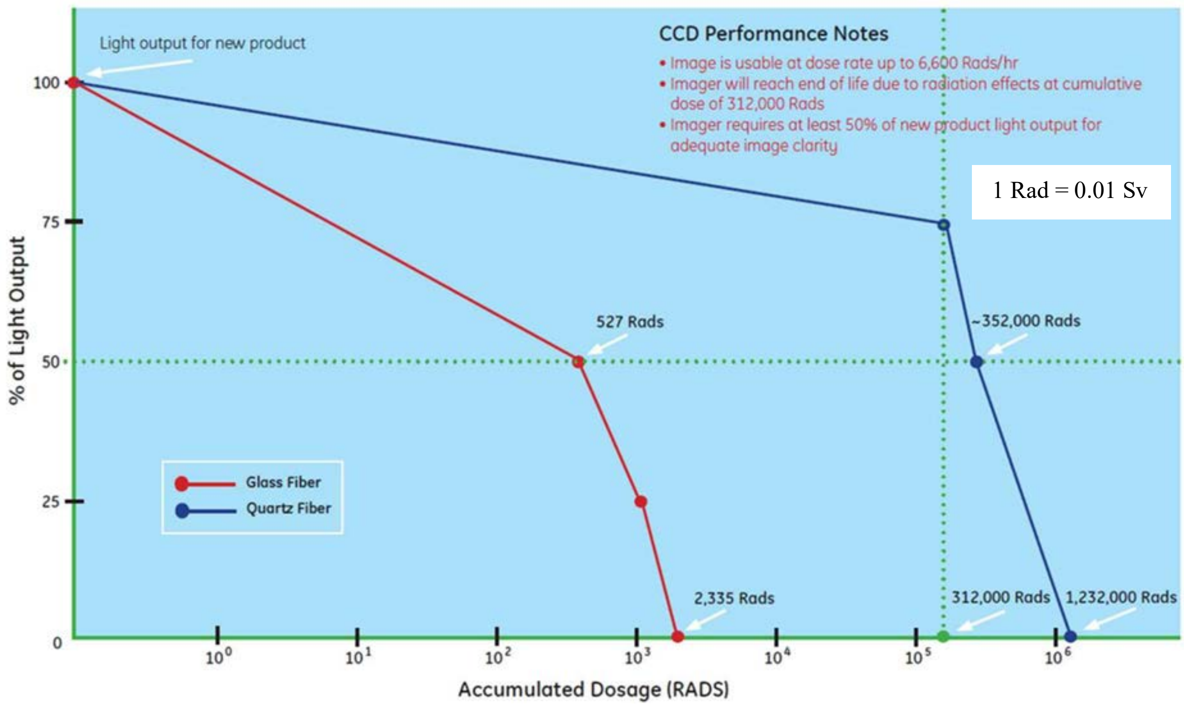


FIG. 56. Radiation effects on different type of fibres (courtesy Waygate Technologies).

Instead, there is an almost three orders of magnitude increase in resistance in devices bearing a quartz fibre, making it possible to use them for periods of about 1000 hours. Hence, two alternatives can be considered to perform the desired visual inspection: the acquisition of an instrument supplied with a radiation resistant video system or the use of replaceable cameras, to be used and disposed in short times. Whereas this choice would be more accessible from the economic point of view, the availability of a product specifically designed to be used in radiation fields would give the opportunity to inspect more silos in the future; it would be even possible to conceive this procedure as a routine practice.

(c) Monitoring of the main concrete structure at ASECQ–CNA–I dry storage site

The activities carried out in order to assess the durability of concrete structures were described in Appendix III.1.1. Among them, the monitoring of the ASECQ–CNA–I main structure is to be continued for years. Monitoring of the lower probes (-10.5 m) S1 to S4 had started some time ago (results were reported in the previous Progress Report). The upper ones, which are near ground level, could only started to be measured in May 2018, after the cement cast was performed. Fig. 57 shows the evolution of measured variables from the beginning of the project.

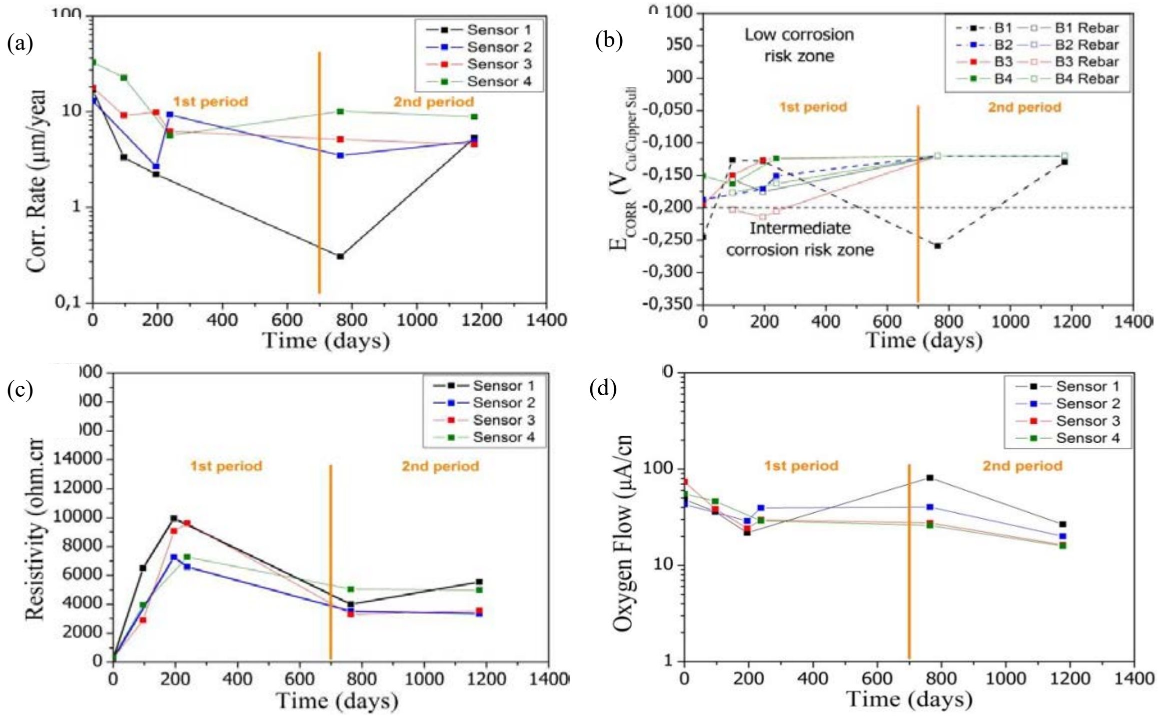


FIG. 57. Evolution of parameters Corr. Rate (a), E_{Corr} (b), Resistivity (c) and Oxygen Flow (d) in the main ASICNA–I structure (courtesy of CNEA).

The results of the first determination performed with the upper sensors S5 to S8 are shown in Table 17. Include: Sensed temperature (T); Concrete resistivity; Oxygen flux; Reinforcement bar corrosion potential (measured directly on a structural bar connection using the sensor's reference electrode); Sensor corrosion potential (measured on a bar embedded into the sensor); Corrosion rate.

TABLE 17. RESULTS OF FIRST MEASUREMENTS PERFORMED IN SENSORS S5 TO S8

Sensor ID	T (K)	Resistivity (Ω.cm)	Oxygen Flux (μA/cm2)	E_{CORR} bar (V_{CSE})	E_{CORR} sensor (V_{ESC})	CR (μm/year)
S5	301	112249	9	-0.205	-0.357	17
S6	311	46922	15	-0.293	-0.293	17
S7	302	28712	4	-0.195	-0.311	15
S8	302	92071	3	-0.184	-0.245	8

As the installation is under construction, there are several factors that strongly affect both the behaviour of the probes and the determinations themselves. The field is still open and subject to inclement weather, including a period in which the central pool was flooded with rain. In addition, the irruption of the construction machinery generates a lot of electrical and electronic noise, affecting the readings. These situations can explain the observed inconsistent fall in both the Corrosion Potential and Corrosion Rate at 770 days. Nevertheless, the general trends point to an acceptable behaviour, indicating a correct performance of the structure, from the Corrosion point of view. It is expected that once the construction is finished the determinations will stabilise and the trends will be in the sense of reducing the reinforcement bars corrosion rate, with increasing values of Corrosion Potential and Resistivity, whereas the levels of Corrosion Current and Oxygen Flux will tend to go down. The main purpose of the monitoring system is to assess the corrosion through the years to come once this transition stage is over.

(d) Monitoring of the main concrete structure at ASECQ–CNE dry storage site

Due to uncertainties related to shielding capability, the ARN denied authorisation to use concrete corrosion probes in silos at CNE, even in those yet to be constructed. They are concerned with the effect of having insertions of electrodes and cables, as well as of the degradation of the probes themselves.

(e) Inspection of stainless steel welds in ASECQ–CNA–I main components

The main concern about the degradation of stainless steel components is the likelihood of stress corrosion cracking (SCC). According to what has been established in numerous research works, the main factors that influence the SCC of stainless steels are the environment (humidity of the air, salt concentration, temperature, etc.), stresses and welds. Since the stainless steel components of ASECQ–CNA–I are currently being supplied, interest has been raised to evaluate the quality of the welds. The first step will be to make a visual inspection of the components that are already on the site. The permission of the authorities has already been obtained and this activity will be completed shortly.

III.1.2.2. Planned work

The following tasks will be completed during the following period:

- Evaluation of localized corrosion in coupons of ASEC–CNE welded canisters: Samples that have already been subjected to corrosion tests will be analyzed to look for evidence of localized corrosion. Also, an evaluation of the corrosion intensification due to galvanic coupling with stainless steel will be done;
- Visual inspection in the interior of silos at ASECQ–CNE: A decision has to be made on whether a videoscope be constructed that could be supplied with disposable video cameras or to acquire an instrument capable of resisting the radiation fields present in contact with the SFE canisters;
- Monitoring of the main concrete structure at ASECQ–CNA–I Dry Storage Site will be continued;
- Inspection of stainless steel (SS) components fabricated for this facility was initiated.

III.1.3. Third progress report (2019)

The two Dry Storage Facilities for Spent Fuel Elements (SFE) burned in power reactors existing in Argentina were described in detail in the First and Second Progress Reports. The main durability related concerns were enumerated, as well as the actions taken to evaluate them. These activities were partially continued during the third period; whereas several others could not be followed because the access to sites was restricted by the sanitary restrictions imposed by the COVID19 pandemic. Instead, a full revision of AMP at the ASECQ–CNE was carried out.

III.1.3.1. Description of the work carried out during the third period

The following sections provide a description of the work carried-out in the third phase of the research study.

(a) Corrosion tests of steel canisters at ASECQ–CNE

A metallographic evaluation was made on coupons of canister's material subjected to the corrosion immersion tests described in Appendix II.1.2. Fig. 58 shows polished cross section cuts of steel samples. General dissolution is observed, what erodes some 200 µm of the wall thickness. However, the same analysis in coupons containing welds show the presence of pits of various depths, up to 850 µm, as also seen in Fig. 58.

Taking into account the test duration, the measured penetration would produce container perforation in a matter of 3 to 4 years, provided the water incursion is continuously maintained.

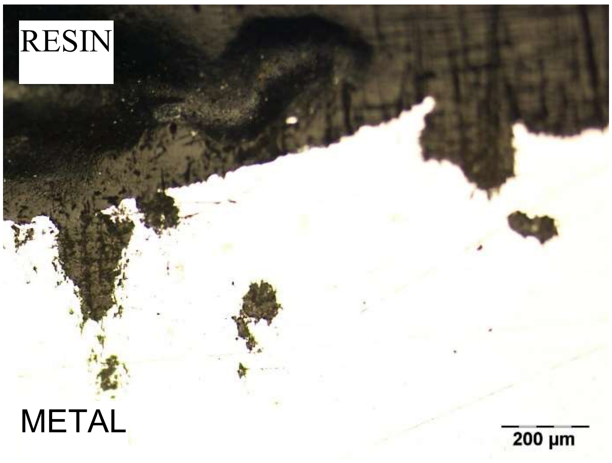


FIG. 58. Penetrating attack in welded samples (courtesy of CNEA).

In order to correctly assess the degradation state of the canisters it is crucial, then, to know the history of water ingress events and their duration. For this reason, a comprehensive revision of the historical data pertaining to storing conditions was done.

(b) Revision of AMP at the ASECQ–CNE

This analysis comprises the following: review of analysis performed on coupons at the mock–up silo, evaluation of mock–up canisters, regular air maintenance operations, possibilities of water ingress and surveillance of cracks evolution in concrete. Upon requirement, responsible personnel from the operating company (NASA) have provided all the necessary background needed to reconstruct the story of the installation.

(1) Evaluation of Coupons at the Mock–up Silo

As stated in Appendix III.1.1, this particular silo belongs to the first batch put into operation, approximately 30 years ago; is full of canisters but is not loaded with SFE. It contains also coupons of the various materials used in the facility components: silo liners, carbon steel from the side of the barrels containing SFE and samples of the welds between carbon steel side and stainless steel grid and/or lid. The same Report shows views of the canister's side obtained using a videoscope, depicting a relative clean surface without sings of penetrating attack.

With respect to the samples, they are periodically taken out to perform visual inspection and weight determinations. The results indicate they have not suffered from corrosion processes so far.

(2) Evaluation of Mock–up Canisters.

There are empty canisters fabricated in the same way that those containing spent fuel, which have been exposed to the atmosphere of the site since the beginning of operation. Although they look quite oxidised after some 30 years, there were no signs of any penetrating corrosion Fig. 26

(3) Air Maintenance Operations

One of the main aspects that were historically revised in relation with AMP activities was the regular procedures which are followed in order to check the quality of the air inside the silos. These consist in replacement of the internal air by other clean and dry. The air is vacuum pumped through filters and replaced. Determination of humidity and activity is performed on what was extracted. These actions are carried out twice a year.

All silos were monitored at the beginning of operation of the facility; however, it was noted that in many cases the extracted air contained humidity, even though it had been injected with 0 % humidity air six months earlier. It was decided, then, to extend the purging time from 24 hours to 15 days, with successful results. Relative humidity is kept below 30 % measured at 298 K. This translates into the range 2 to 10 % in the interior of the silos, what would produce corrosion rates in the order of 1 μm/year. Initially, all silos were monitored; however, due to the mentioned purging time extension only 20 silos are treated per campaign (approximately 10 % of the total), what means that air in all silos is renovated every 5 years.

Krypton activity is measured every 3 years; it is always found to be in the range 10^5 to 10^8 Becquerel. Total measured internal ^{85}Kr activity (adding up the content of all silos) amounts to 0.43 GBq, that is only 0.2 % of the permitted release amount for the whole of the site, including the reactor itself. No ^{137}Cs activity has been detected.

(i) Possibilities of Water Ingress

Water collected during construction. The silos are flooded with water in order to refrigerate during casting of the concrete. Once this operation is finished the water is evacuated through the lower inspection line, which is some 11 cm above the floor. The rest is fully dried by hand before loading.

Water collected during loading. The spent fuel loading operation takes about a week. A top hat like protection during pauses; however, this system in occasions failed to avoid rainwater ingress. This problem was solved by temporarily closing the silo using the definitive concrete cork with a rubber seal inserted, as seen in Fig. 59 (a) and (b).

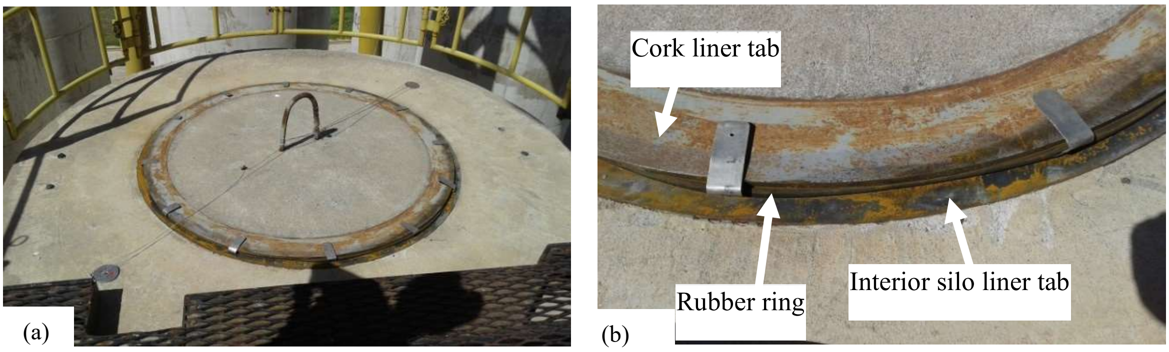


FIG. 59. Silo seal system used during SFE loading pauses (a): General view of concrete cap in position. (b): detail showing sealing area (courtesy of CNEA).

When the loading is finished, the rubber seal is taken out and both the cap and liner tabs are airtight welded.

Known case of water ingress. Water presence was verified in Silo No. 1. It was drained with pumps and let to dry spontaneously through evaporation. Current humidity content inside standard margins.

(ii) Surveillance of cracks evolution in concrete

As part of the surveillance routine, visual inspection is caried out on the external surfaces of the silo's concrete shielding. Any crack which is detected is registered and its evolution followed with time. When they reach a length of about 12.7 mm (½ inch) they are filled with cement in order. The maximum determined depth was around 25.4 mm (1 inch), getting close to the reinforcement bars.

(iii) Conclusions

Considering that:

– Water ingress events have been limited in duration;
– Humidity levels, which were high at the beginning of operation, are now kept inside desirable margins;
– There is no significant corrosion observed in mock–up canisters and samples;
– Krypton activity is minimum;
– No caesium activity has been detected.

It can be concluded that the general storing conditions have improved with the operational experience, and it is expected that no significant corrosion has occurred at the ASCEQ–CNE site.

(b) Monitoring of reinforcement bars corrosion at ASECQ–CNA–1

Description of activities related to the monitoring of the concrete structure at the DSF in the site of Atucha–1 NPP can be found in the previous Progress Reports. This work was continued up until the end of 2019; the site could not be accessed after that date, due to the restrictions associated with the sanitary conditions established in the frame of the COVID19 pandemic.

As it can be seen in Fig. 60, after some 4.5 years of operation, the majority of measured Corrosion rate values stay in the low corrosion risk zone and the measured Corrosion Rates are negligible. Resistivity is still not controlling the corrosion process.

Determinations are still affected by ongoing construction work and environmental conditions, mainly presence of water. It is expected that parameters will stabilise as the construction advances. This activity is being maintained. It has been running for ~2000 days so far and is intended to continue during the whole of the facility life.

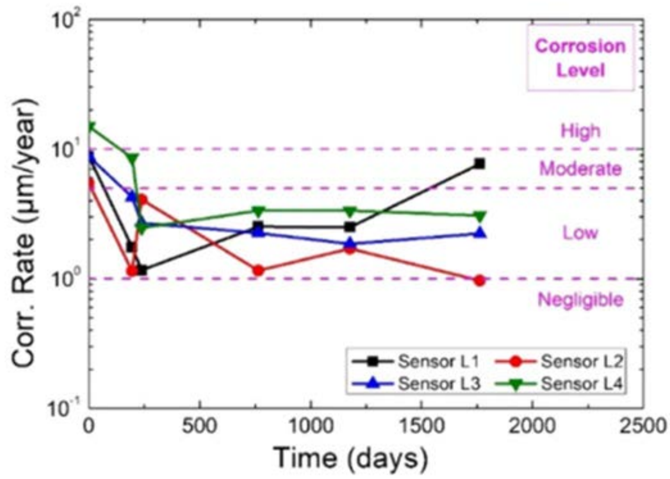


FIG. 60. Evolution of Corr. Rate at the ASECQ–CNA–1 concrete structure (courtesy of CNEA).

III.2. PAKISTAN

Since its commissioning in 1971, Karachi Nuclear Power Plant (KANUPP) has completed more than 17 Effective Full Power Years (EFPYs) of operation. KANUPP, the first commercial nuclear power plant of Pakistan, is 137 MW(e) CANDU type heavy water reactor. KANUPP has approached the end of its useful life after having a successful operation of about five decades. The irradiated fuel discharged from reactor is stored in Spent Fuel Storage Bay (SFSB). The spent fuel storage bay which was designed for plant's life period has approached full capacity. To generate additional storage capacity, high density tray rack (HDTR) system has been designed and is being successfully implemented at KANUPP.

As a long term solution of spent fuel storage problem (during plant life and beyond decommissioning), KANUPP has planned to construct and operate a spent fuel dry storage facility within the KANUPP boundary. The total expected inventory of spent fuel bundles from KANUPP at the end of its extended life (1972 — 2021) is estimated at around 36 000 bundles. It is planned to remove 10 years cooled spent fuel bundles housed in spent fuel bay and place in dry atmosphere in specially designed concrete cask at the facility. On successful completion, this facility would provide interim storage solution of approximately fifty (50) years. Ultimately the fuel along with the cask would be transported to national repository site intended to be constructed.

III.2.1. Objective

The research activity is aimed at revaluation of design and review of safety features of SC–108 dry storage cask (Fig. 61) designed for the KANUPP Spent Fuel Dry Storage Facility (KSFDSF), in the purview of ageing management.

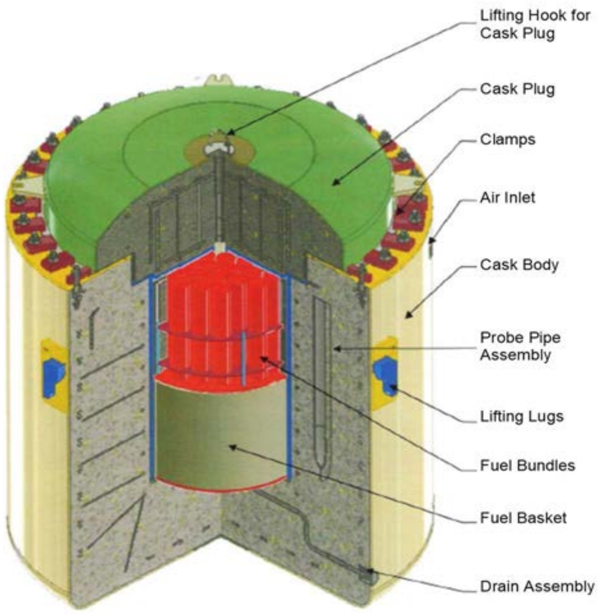


FIG. 61. Basic Schematic of Cask SC–108 (courtesy of PAEC).

III.2.2. Research approach

Several design and analysis tasks are performed in this perspective. A comprehensive plan for ageing management of spent fuel cask was established as per IAEA guidelines. Initially, several analyses with respect to design have been performed such as: shielding design, criticality safety, confinement and etc. The analyses with respect to ageing such as fatigue analysis, stress analysis, corrosion analysis of metallic components, moisture control and etc. have been conducted for the cask to ensure its safe operation over the entire design life and mitigation of its radiological releases. A number of tests have also been conducted and the tests results were found satisfactory as per design and the test record is maintained for ageing management purpose. Some of major analyses which are important from the viewpoint of ageing and have serious effects on the performance of cask are summarized as following:

III.2.2.1. Fatigue analysis of spent fuel cask

Fatigue analysis of the cask is performed to ensure the integrity of cask's inner liner in presence of thermal stresses and fatigue. The analysis is performed by modeling inner and outer shells and inner liner made up of stainless steel whereas concrete shielding is also provided between the shells using ANSYS software. Hex dominant method is used to mesh the model.

Thermal stresses over cask structure are evaluated with homogeneous heat source of strength 406.94 W. The heat source strength of the fuel basket is estimated as 117.23 W/m2, applied as heat flux at the inner surface of steel liner. Temperature distribution is shown in Fig. 62. Cask is to be backfilled with Helium which will act as a medium of heat transfer from fuel assemblies to cask outer shells. Backfilling of helium in SC–108 will exert a pressure of 0.2 MPa on the stainless steel inner liner. The static structure analysis is carried out by applying a constant pressure of 0.2 MPa at the inner liner. As a conservative approach, a stiffer model of cask is used by fixing the cask at the bottom. Thermal stresses are also imported into the static structural module, to incorporate the thermal stresses and accurately determine the cycles to failure. Static structural analysis reveals that the backfilling process will produce a very nominal amount of stress in the cask. Maximum stress generated in SC–108 is 465.73 MPa while the ultimate tensile strength of SS is 586 MPa [113]. Results are presented in Fig. 63.

Cask's inner liner would experience a cyclic load every time it goes through the helium backfilling process that will ultimately results in fatigue. Fully reversed loading follows sinusoidal pattern in which stress value reaches maximum and minimum value once per cycle. Fatigue analysis reveals that SC–108 can withstand 1735 cycles (Fig. 64). It means that SC–108 can be backfilled with helium 1735 times before it yields. No. of cycles are estimated from the point of failure i.e. the surface that is in contact with ground.

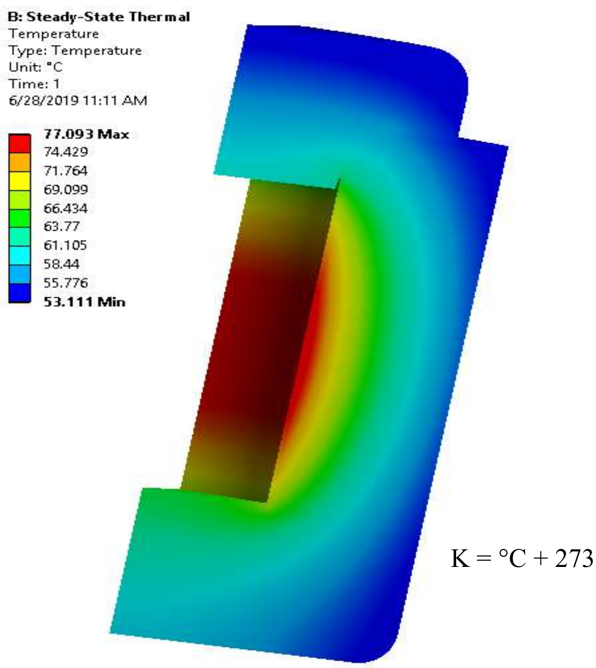


FIG. 62. Temperature distribution (courtesy of PAEC).

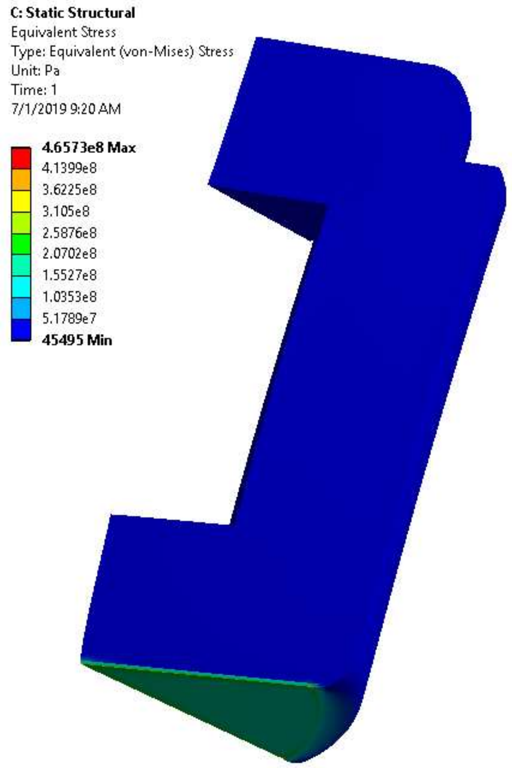


FIG. 63. Equivalent Von–Mises stress (courtesy of PAEC).

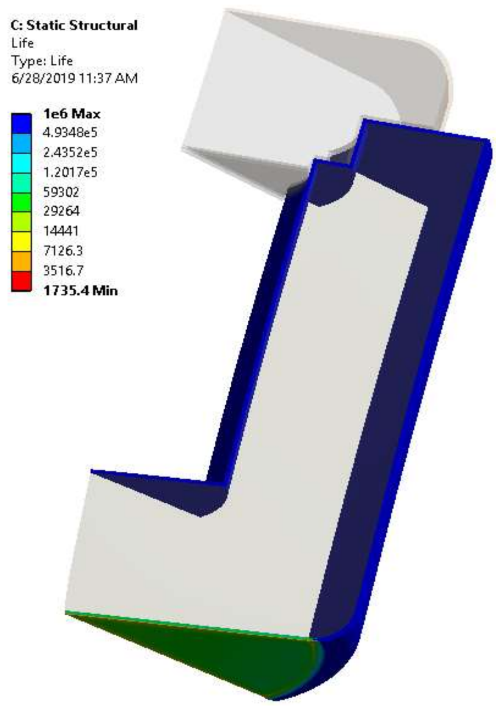


FIG. 64. Fatigue cycles to failure (courtesy of PAEC).

III.2.2.2. Establishment of allowable stress considering ageing deterioration in cask structure

Maximum possible stresses over the cask are evaluated considering the effect of high temperature resulting in deterioration of structural material (SS and concrete). Allowable stresses are calculated, considering the changes in material physical properties like modulus of elasticity and tensile/compressive strength, using ANSYS software.

Values of these parameters are established using the available literature and are presented in Table 18.

TABLE 18. PARAMETERS USED TO ESTIMATE THERMAL STRESSES

Parameter	Concrete	Stainless Steel
Young's Modulus (GPa)	19.5	181
Poison's Ratio	0.18	0.3
Tensile Ultimate Strength or Tensile Strength (MPa)	3.5	480
Compressive Ultimate Strength or Yield Strength (MPa)	3.4	150

Cask model developed for analysis mainly consists of inner/outer shells and inner liner made up of stainless steel with the provision of concrete within the shells for neutron and gamma shielding. To mesh the cask geometry, sweep and hex dominant method is used with an element size of 1E−2 m (Fig. 65). Homogenized source strength in fuel cylinder is 406.94 W. The heat source strength of the fuel basket is estimated as 117.23 W/m^2 and is applied as heat flux at the inner surface of steel liner. Convective heat transfer coefficient for steel air is usually taken as 10 $W/m^2 \cdot K$ [114] , however for a more conservative approach it is taken as 5 $W/m^2 \cdot K$.

Analysis show that peak temperature in the cask is 350 K as presented in Fig. 62. Very nominal amount of stresses in the cask are observed. The storage cask has been evaluated at Maximum Equivalent stress (Von–Mises). Maximum stress generated in SC–108 is 119.29 MPa while the ultimate tensile strength of SS is 586 MPa (Fig. 66).

FIG. 65. Meshed model (courtesy of PAEC).

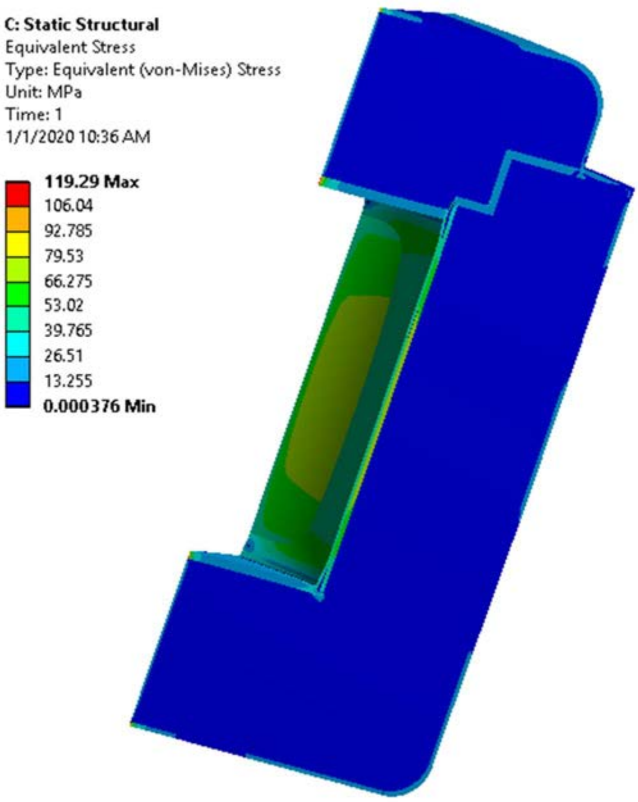


FIG. 66. Equivalent Von–Mises stress (courtesy of PAEC).

III.2.2.3. Evaluation of cask cooling performance due to gap between liner and concrete

Due to ageing, a gap may be developed between liner and concrete over a longer storage period. The full cask model is employed to assess cooling performance of the cask with gap in the form of air and vacuum. A homogenized source of 409 W with estimated heat source strength of the fuel basket of 117.23W/m^2 is considered. Heat transfer from cask to earth and thermal resistance of cask is neglected to develop an adiabatic model.

Maximum temperature is observed as 350 K within the cask under steady state as shown in Fig. 60. To simulate the condition of cask degradation, a gap of 1mm is created between liner and concrete. Thermal analysis shows that the gap between liner and concrete hinders the heat flow and the maximum temperature within the cask rises up to 350.23 K in case of air between liner and concrete whereas in case of vacuum the maximum temperature of cask is 408 K as presented in Figs 67 and 68 respectively.

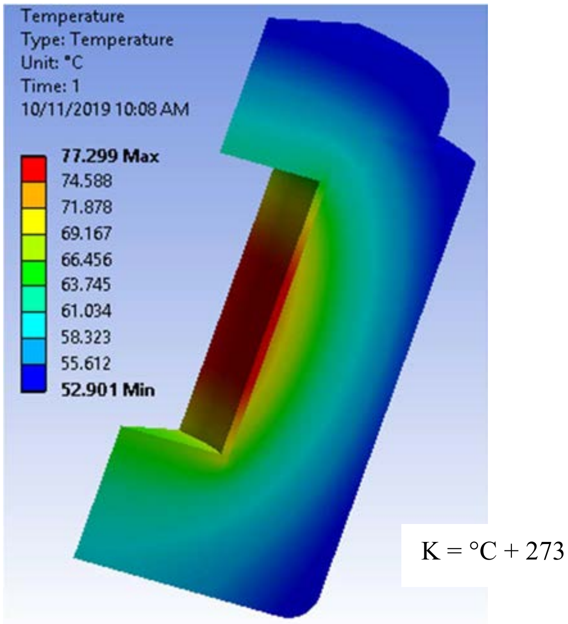


FIG. 67. Temperature distribution with air gap (courtesy of PAEC).

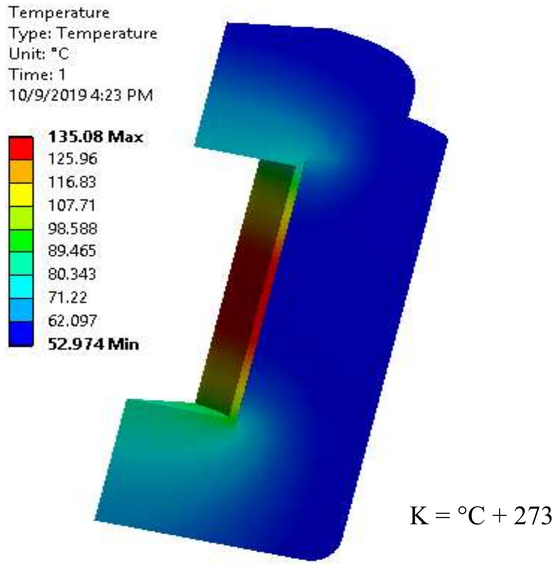


FIG. 68. Temperature distribution with vacuum gap (courtesy of PAEC).

III.2.3.4. Evaluation of corrosion effect on shielding performance of spent fuel dry storage cask

The cask has 12 mm thick outer liner made of carbon steel. Atmospheric corrosion is a severe hazard to all engineering structures, industrial installations and civil construction near Karachi's coastal line. Moreover, cask interior is exposed to gamma ray flux of 2.82E15 photons/s with cumulative heat load of about 410W. Effect of atmospheric and gamma rays induced corrosion on shielding performance of cask is evaluated through its design life.

The cask is exposed to heat and gamma ray flux far lower to initiate corrosion, therefore effect of gamma rays to cause corrosion in inner and outer structure of cask is negligible. A corrosion rate of 160 µm/yr for carbon steel at Karachi has been reported [115]. Shielding analysis for storage cask is performed by reducing the outer liner at the mentioned corrosion rate. The photon rate applied to cask structure for shielding analysis and resulted dose rate are shown in Fig. 69. It can be concluded that despite this corrosion rate and in absence of any protective layer/coating the dose rates remain in allowable limit of 1 mSv/hr.

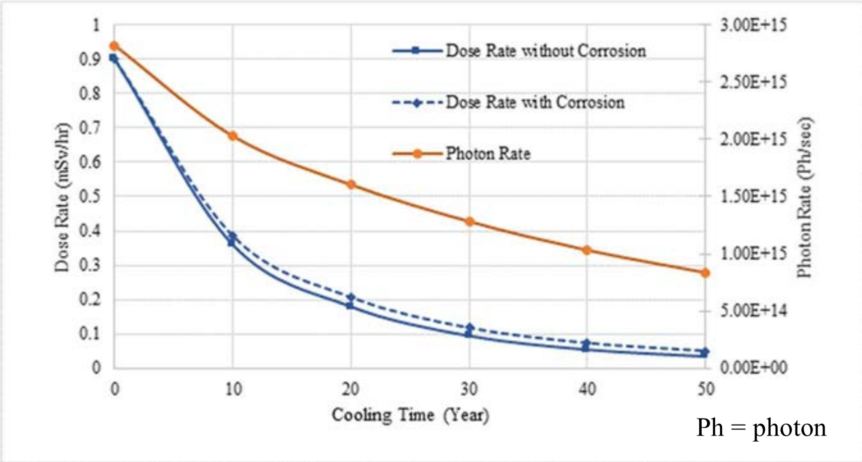


FIG 69. Photon rate and dose rate with/without corrosion variation with cooling time (courtesy of PAEC).

III.2.3.5. Evaluation of shielding performance of spent fuel dry storage cask trough the design life

Shielding performance of cask has evaluated for its design life of 50 year. Gamma source term for 10 year cooled KANUPP spent fuel bundles at discharge burnup of 9000 MWD/t(U) is generated using ORIGEN2 code [116]. The source term is used for shielding analysis by employing QAD–CG shielding analysis computer code5. Barite concrete with density 3.1E-3 g/mm3 is used as the shielding material whereas a margin of 10% is included in shielding dimensions of the concrete to account for uncertainty in modeling, cross–sections data and assumptions.

The applied gamma rate and resulted dose rates are presented in Fig. 70. It can be seen that the dose rates remain in permissible dose limit of 1 mSv/hr up to design life.

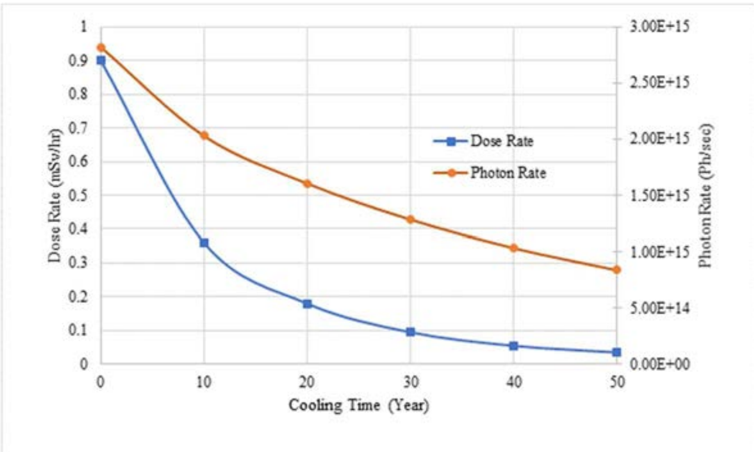


FIG 70. Photon rate applied for shielding analysis and dose rate with cooling time (courtesy of PAEC).

[5] QAD–CG Computer Code for Shielding Design of Radioactive Equipments, Shanghai Nuclear Engineering Research and Design Institute, May 1996.

III.2.4. **Research outcome**

The present project seems to be very useful in understanding the ageing degradation of dry storage cask, SC–108. The analyses performed in this research were helpful for the submission of safety analysis report (SAR) and certification of the cask from the regulatory authority. In addition, the work performed in this activity will be beneficial for the development of a comprehensive ageing management programme for SC–108 to mitigate the ageing effects over the cask for life extension.

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LIST OF ABBREVIATIONS

AE	acoustic emission
AMID	ageing management INPO database
AMP	ageing management programme
AMR	ageing management review
ASR	alkali silica reaction
BWR	boiling water reactor
CCC	California Coastal Commission
CISCC	chloride induced stress corrosion cracking
CNEA	Comisión Nacional de Energía Atómica, Argentina
CoC	certificate of compliance
CRP	coordinated research project
CS	carbon steel
DGR	deep geological repository
DPC	dual purpose cask
DSF	dry storage facility
DSS	dry storage system
DW	demineralized water
ECT	Eddy current testing
EPRI	Electric Power Research Institute
HSU	horizontal storage unit
HT	heat transfer
INPO	Institute of Nuclear Power Operators
ISFSI	independent spent fuel storage installation
LMP	Larson–Miller parameter
LPB	low plasticity burnishing
LWR	light water reactor
MPC	multipurpose canister
MS	Member States
NPP	nuclear power plant
U.S. NRC	U.S. Nuclear Regulatory Commission, United States of America

OD	outdoor air
PRIS	Power Reactor Information System
PWR	pressurized water reactor
QA	quality assurance
RAMM–TM	remote area modular monitoring for temperature measurement
RH	relative humidity
RIRP	regulatory issue resolution protocol
RS	radiation shielding
SFE	spent fuel element
SFP	spent fuel pool
SCC	stress corrosion cracking
SNF	spent nuclear fuel
SONGS	San Onofre nuclear generating station
SP	shot peening
SS	structural support
SSC	structures, systems and components
TEPCO	Tokyo Electric Power Company, Japan
TLAA	time limited ageing analysis
TRANSC	Transport Safety Standard Committee
UT	ultrasonic testing
VSU	vertical storage unit
VT	visual testing
WS	welded steel
Zry-4	zircaloy 4 alloy

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