Technical Bases for Extended Dry Storage of Spent Nuclear Fuel

Technical Report

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REPORT SUMMARY

Independent spent fuel storage installations (ISFSIs) are currently licensed for 20 years. However, delays in developing permanent spent fuel disposal capability require continued ISFSI storage beyond the 20-year term. This report provides a technical basis for demonstrating the feasibility of extended spent fuel storage in ISFSIs.

Background
ISFSIs use storage systems in which spent fuel is placed in a dry gas environment (typically helium) inside a metal container, which is sometimes placed inside a concrete cask. Because the Nuclear Regulatory Commission (NRC) limits the license term for an ISFSI to 20 years from the date of issuance and no final solution for spent fuel disposition is available, utilities must prepare for license extension.

Objective
To summarize potential dry storage system degradation issues that may affect the maintenance of ISFSI safety functions for time periods beyond 20 years along with approaches for managing these potential degradation issues.

Approach
Investigators examined documents containing information on long-term aging of ISFSIs (beyond 20 years). These documents included existing regulations governing extended spent fuel storage, 10 CFR Part 72, several documents that discussed specific aging issues, reports that provided data on the potential importance of aging issues, a summary of the first ISFSI license extension application to be filed with the NRC, and a discussion of possible aging management options. Investigators next evaluated the license renewal application for the ISFSI at Dominion Generation's Surry nuclear plant, focusing on approaches to identifying, evaluating, and managing aging issues associated with extended storage. In addition, a review of prior work provided data to evaluate the relative importance of degradation issues. This review focused primarily on the Dry Cask Storage Characterization (DCSC) Project during which a metal cask that had stored spent fuel for approximately 14 years was opened, and the cask and stored fuel assemblies were examined for signs of aging. Finally, investigators reviewed aging management practices to determine if current monitoring and inspection programs would be adequate for extended storage.

Results
This report reviews specific aging issues related to extended storage in ISFSIs. Among the aging issues discussed are the following: degradation of the spent fuel assemblies (including, for example, cladding creep, diffusion-controlled cavity growth, hydride reorientation, radiation embrittlement, and thermal annealing), corrosion of metals inside the sealed spent fuel canister,

corrosion of metals outside the canister, radiation damage to metals and concrete, neutron shielding irradiation and thermal damage, and concrete degradation. These aging issues have already been identified, evaluated, and managed as part of the initial 20-year ISFSI licenses granted by the NRC. Only a few aging issues associated with continued dry storage beyond 20 years need to be revisited.

Data collected in the DCSC project showed no evidence of significant degradation of metal cask systems important to safety from the time of initial cask loading in 1985 to testing in 1 999. DCSC project examinations of the spent fuel in this cask after approximately 15 years of storage suggested that there was little to no cladding creep or evidence of significant hydride reorientation and no rod failure. Residual creep testing on the spent fuel cladding suggested there was significant residual creep available such that maintenance of spent fuel integrity well beyond 20 years is likely. The surveillance and monitoring programs ISFSI licensees currently use appear sufficient for extended storage as well.

EPRI Perspective

The NRC has not fully developed a license extension approach for ISFSI licensees, and there has been some concern that aging issues associated with continued dry storage beyond 20 years will prevent ISFSI storage for several more decades. A summary of aging issues related to extended dry storage—and how to manage them—is needed to ensure the safety of long-term spent fuel storage, either at the surface or underground.

Keywords

Spent Fuel Storage
Independent Spent-Fuel Storage Installation
License Extension

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1 INTRODUCTION

1.1 Background

During the 1960s and 1970s when the nuclear power plants were designed that are in operation today in the U.S., it was thought there was little need to store significant quantities of spent fuel on site. At the time, it was thought that spent fuel would be removed from the spent fuel pools after only a few years of cooling, and sent for reprocessing. Furthermore, it was understood that a long-term interim storage and/or a permanent disposal facility would be available by some time in the 1980s. The subsequent elimination of the reprocessing option, along with significant delays in the availability of centralized interim storage or permanent disposal of spent fuel required nuclear utilities to expand their on-site spent fuel storage capabilities. Initially, utilities opted for reracking their spent fuel pools to accommodate larger numbers of assemblies. By the early 1980s, as delays in the availability of centralized interim storage and disposal continued, it became apparent that some of the spent fuel inventory would have to be stored on site, but out of the spent fuel pools.

Starting in the early 1980s, the U.S. Department of Energy (DOE) and EPRI developed technology, and the utilities installed dry storage capabilities to alleviate spent fuel storage problems and maintain a full core reserve while DOE was preparing a disposal facility. At the same time, the U.S. Nuclear Regulatory Commission (NRC) developed regulations governing the development and operation of Independent Spent Fuel Storage Installations (ISFSIs), now codified in 10 CFR Part 72 and associated review plans, regulatory guides, and interim guidance documents. While ISFSIs could also include wet storage facilities, all ISFSIs developed so far in the U.S. are dry storage systems.

Dry storage of spent light water reactor fuel in an inert atmosphere is currently licensed by the NRC for 20 years with the potential for an extension beyond 20 years. The general basis for a 20-year license was that certain dry storage system ageing issues may sufficiently degrade safety that a review of the continued safety of dry storage beyond 20 years would be required [10 CFR Part 72]. While analyses suggested that dry storage systems would likely maintain their capabilities significantly beyond 20 years, NRC was concerned about the growing uncertainty in potential dry storage system performance beyond 20 years. In addition, it was thought at the time that a permanent spent fuel disposal facility would surely become available within 20 years.[1] Thus, there was likely to be little need to continue to use dry storage systems for more than 20 years.

[1] At the time of the development of 10 CFR Part 72 in the early 1980s, Congress was enacting legislation requiring DOE to develop either centralized interim storage and/or permanent disposal capability such that DOE would begin accepting spent fuel from nuclear utilities no later than January 31, 1998.

Some 20 years after the establishment of 10 CFR Part 72, neither centralized interim storage nor a permanent disposal facility is yet available. DOE's current estimate of when the Yucca Mountain disposal facility will begin accepting spent fuel is late 2010. Due to these delays, some dry storage facility licenses will need to be extended beyond 20 years. For example, the 20-year license for Dominion's ISFSI at the Surry nuclear plant will expire in 2006. Earlier this year Dominion submitted an application for a 40-year license extension for the ISFSI at Surry [Dominion, 2002]. Several other ISFSI 20-year licenses are also due to expire ahead of 2010. These affected utilities are also developing plans for extending their ISFSI licenses beyond the first 20 years.

1.2 Summary of Work Addressing ISFSI use beyond 20 Years

Over the last 10 years, a number of utilities have obtained U.S. Nuclear Regulatory Commission (NRC) approval for 20-year dry storage. Lack of progress on the repository points to the need for storage periods much longer than 20 years and possibly as long as 100 years. Flow diagrams to extend fuel storage capabilities from 20-year dry storage to a nominal 100-year dry storage are shown in Figures 1-1 through 1-3. In some cases, the fuel would be able to stay in its original cask for the full 100-year period. In other instances, such as a significant change in the cask materials properties when continued operation is not feasible, or for movement to a different storage location, the fuel might need to be put in different casks via either a wet or dry transfer. The behavior of the system during each step of the process must be evaluated to ensure that the criteria imposed on the system are met for the 100-year period. Should an accident or off-normal event occur during the initial storage period, the subsequent action may depend on the timing of the event. The sequences shown in Figures 1-1 and 1-2 assume that the event occurs any time within the 20-year period and that the fuel remains in the original cask for the remainder of the 20 years. Even though licenses have been granted for only 20-year periods, many of the methodologies developed and support data obtained should be able to support the longer storage periods envisioned.

The initial work to show that specific dry storage systems are likely to continue to function safely well beyond 20 years was performed by dry cask storage vendors and individual utilities and documented in the original Safety Analysis Reports accompanying the original ISFSI license applications.[2] Since then several more recent activities aimed at providing technical bases for extended dry storage are underway or are completed. The American Society for Testing and Materials (ASTM) is nearing completion of guidelines for evaluation of ISFSIs for extended use. At present, they are resolving final comments on "Draft 16" of this guideline [ASTM, 2002].

In 1998, EPRI issued a report on "Identification of Data Needs to Dry Store LWR Fuel for 100 Years" [EPRI, 1998]. This report addressed a variety of potential spent fuel ageing issues and concluded that

> [t]he highest priority needs are to 1) determine whether diffusion-controlled cavity growth is viable in zircaloy cladding; 2) determine temperature limits, evaluate cladding degradation mechanisms and post-irradiation mechanical properties of the newer high burnup claddings and fuels; and 3) evaluate the change in the mechanical properties of the cask system with emphasis on deterioration of the polymer neutron shields and seals.

[2] For example, the NRC states the "design life of the Standardized NUHOMS system is 50 years as Described in the [Safety Analysis Report]." [Federal Register, 59 (245), December 22, 1994, pp. 65902]

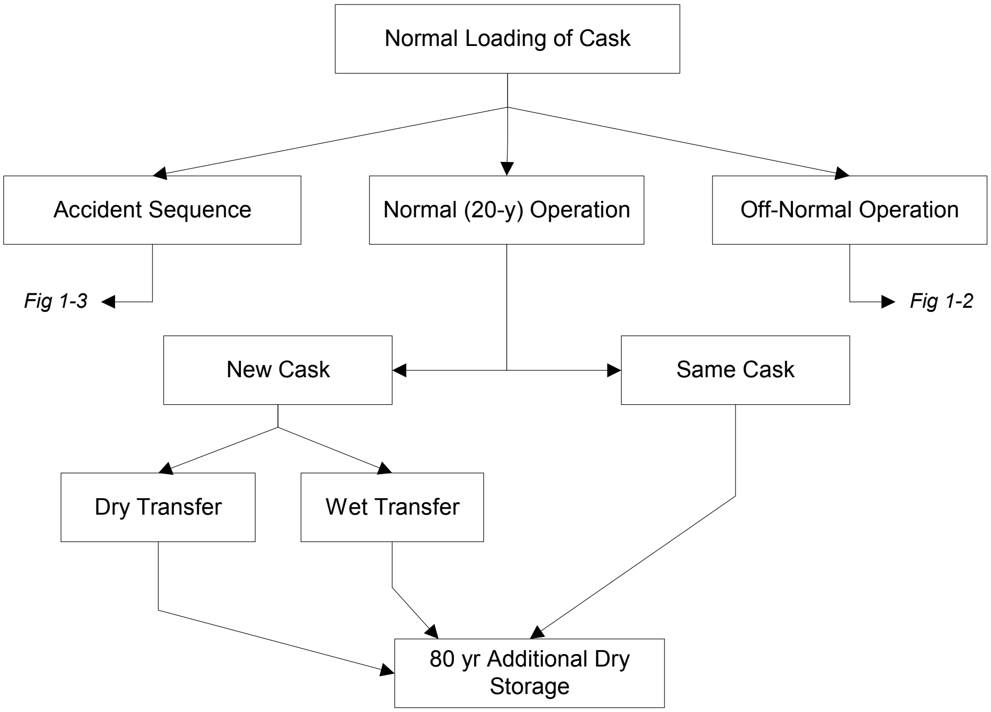

Figure 1-1
Potential Sequences of Events

(Adapted from EPRI [1998])

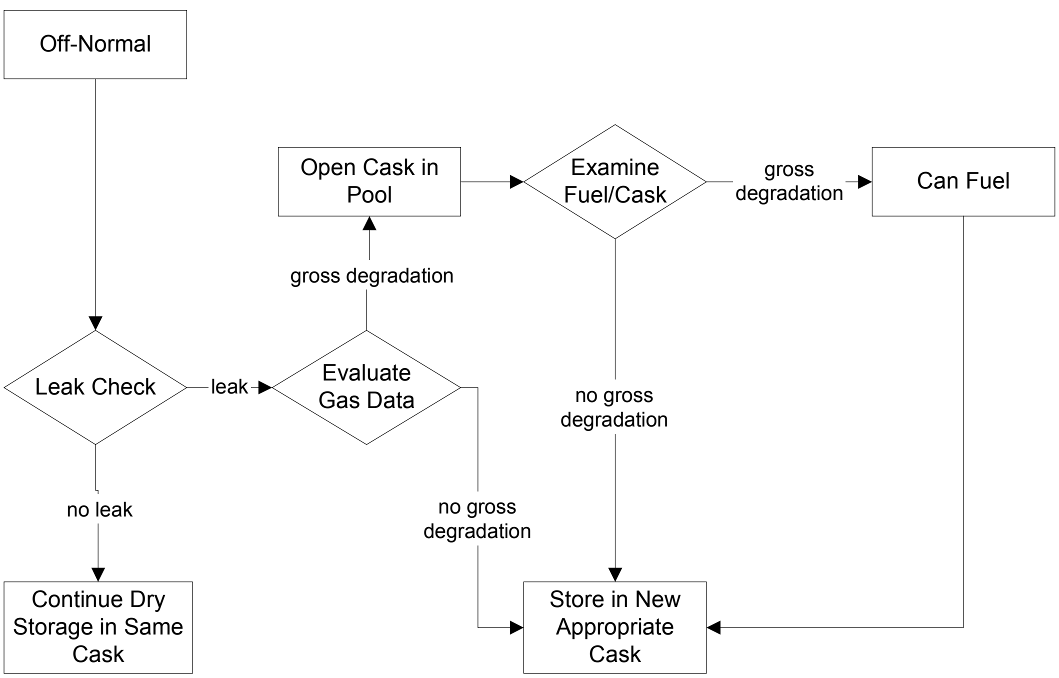

Figure 1-2
Potential Course of Action if Off-Normal Operation Occurs. If there is no Cask Leak then Fuel Oxidation is Precluded. If there is an Erroneous Backfill with Air, then the Fuel Degradation can be Dealt with in 100 Years as well as within 20 Years.

(Adapted from EPRI [1998])

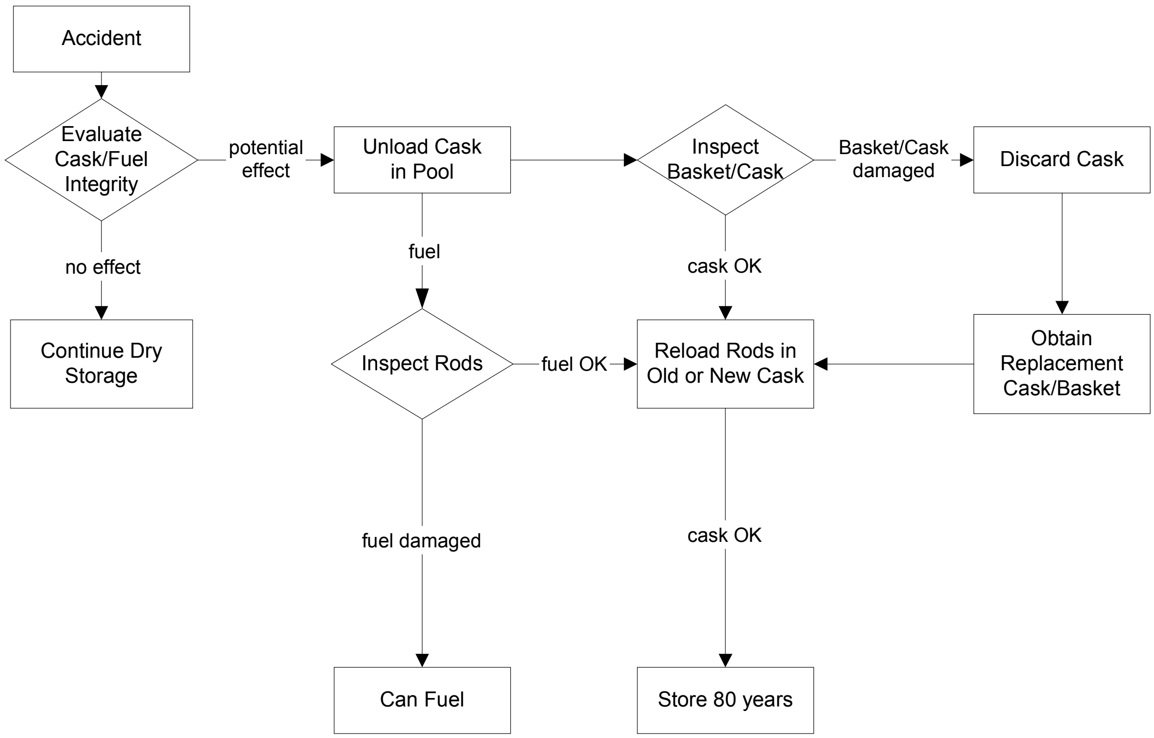

Figure 1-3
Potential Course of Action after an Accident Event

(Adapted from EPRI [1998])

While there is a large amount of literature available on materials degradation, these two references [ASTM, 2002; EPRI, 1998], along with two others on concrete ageing [NEA, 2002; TN, TN, 2001] provide a good summary of the main potential dry storage system degradation issues that may need to be addressed from a technical standpoint.

1.2.1 ASTM Draft Guide for Evaluation of Materials used in Extended Service of Interim Spent Nuclear Fuel Dry Storage Systems

The American Society for Testing and Materials (ASTM) is nearing completion of guidelines for evaluation of ISFSIs for extended use. At present, they are resolving final comments on “Draft 16” of this guideline [ASTM, 2002]. The purpose of the guide is as follows:

> This guide addresses many of the factors affecting the time-dependent behavior of materials under ISFSI service [10 CFR Part 72.42]. These factors are those regarded to be important to performance, in license extension, beyond the currently licensed 20-year period. Examples of these factors are given in this guide and they include materials alterations or environmental conditions for components of an ISFSI system that, over time, could have significance related to safety. For purposes of this guide, a license period of an additional 20 to 80 years is assumed.

> This guide addresses the determination of the conditions of the spent fuel and storage cask materials at the end of the initial 20-year license period as the result of normal events and conditions. However, the guide also addresses the analysis of potential spent fuel and cask materials degradation as the result of off-normal, and accident-level events and conditions that may occur during any period.
>
> This guide provides information on materials behavior to support continuing compliance with the safety criteria, which are part of the regulatory basis, for licensed storage of SNF at an ISFSI. The safety functions addressed and discussed in this standard guide include thermal performance, radiological protection, confinement, sub-criticality, and retrievability. The regulatory basis includes 10 CFR Part 72 and supporting regulatory guides of the U.S. Nuclear Regulatory Commission. The requirements set forth in these documents indicate that the following items were considered in the original licensing decisions: properties of materials, design considerations for normal and off-normal service, operational and natural events, and the bases for the original calculations. These items may require reconsideration of the safety-related arguments that demonstrate how the systems continue to satisfy the regulatory requirements. Further, to ensure continued safe operation, the performance of materials must be justified in relation to the effects of time, temperature, radiation field, and environmental conditions of normal and off-normal service. Arguments for long-term performance must account for materials alterations (especially degradations) that are expected during the service periods, which include the periods of the initial license and of the license renewal. This guide pertains only to structures, systems, and components [SSCs] important to safety during extended storage period and during retrieval functions, including transport and transfer operations. Materials information that pertains to safety functions, including retrieval functions, is pertinent to current regulations and to license renewal process, and this information is the focus of the guide. This guide is not intended to supplant the existing regulatory process. [ASTM, 2002]

The draft Guide identifies performance requirements and lays out a materials performance evaluation process. Several annexes to the Guide provide helpful information on factors that affect materials performance, potential degradation mechanisms for the dry storage system including the spent fuel, and consequences and degradation mechanisms for off-normal and accident conditions. Sections of this draft Guide will be summarized later in this report.

1.2.2 Dominion's Surry ISFSI License Extension Application

Another reference of significant interest is the recent Surry ISFSI license extension application and Safety Analysis Report (SAR) [Dominion, 2002]. These documents put into licensing perspective the potential ageing mechanisms discussed in ASTM [2002] and EPRI [1998]. For example, the SAR separates ageing mechanisms that do not affect safety from those that may. Next the SAR provides analyses for those that may to indicate either that the additional amount of degradation will not be sufficient to sufficiently diminish the large margins of safety in the original system performance, and/or to identify actions Dominion proposes to take (regular surveillance, special inspections) to identify and mitigate negative ageing effects. This report is summarized in somewhat more detail below.

Dominion's Surry ISFSI license extension application begins by adopting a regulatory philosophy for license renewal as discussed in the statements of consideration by NRC in 10 CFR Part 54 [Federal Register, 1995]:

> "The first principle of license renewal was that, with the exception of age-related degradation unique to license renewal and possibly a few other issues related to safety only during the period of extended operation of nuclear power plants, the regulatory process is adequate to ensure that the licensing bases of all currently operating plants provides and maintains an acceptable level of safety so that operation will not be inimical to public health and safety or common defense and security. Moreover, consideration of the range of issues relevant only to extended operation led the Commission to conclude that the detrimental effects of aging is probably the only issue generally applicable to all plants. As a result, continuing this regulatory process in the future will ensure that this principle remains valid during any period of extended operation if the regulatory process is modified to address age-related degradation that is of unique relevance to license renewal."

> "The second and equally important principle of license renewal holds that the plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term. This principle would be accomplished, in part, through a program of age-related degradation management for systems, structures, and components that are important to license renewal....."

Thus, Dominion notes that, based on these principles, "license renewal is not intended to impose requirements beyond those that were met by the facility when it was initially licensed by the NRC. Therefore, the current licensing basis for the ISFSI will be carried forward through the renewal period." [Dominion 2002, page 2-1]

Dominion [2002] developed a scoping process to identify the structures, systems, and components (SSCs) of the dry cask storage system that are both within the scope of license renewal and would require evaluation as part of an aging management review (AMR). The AMR process involved the following four (4) major steps:

1. Identification of in-scope subcomponents requiring AMR

2. Identification of materials and environments

3. Identification of aging effects requiring management

4. Determination of the activities required to manage the effects of aging.

The above approach is consistent with general guidelines proposed in ASTM [2002]:

> The license renewal technical evaluation consists of four parts. *First*, the component evaluation basis is established, which includes component descriptions, the general component design bases, and relevant component operating history. *Second*, the aging-related degradation mechanisms that could affect this component are described and their potential significance to component safety function performance as defined in the

original license of the DCSS is evaluated. *Third*, the effectiveness of existing programs, which also helps manage the facility aging phenomena, is examined to determine if credit can be taken for the existing programs. *Fourth*, for case where the existing programs cannot be shown to adequately manage the effects of age-related degradation, aging management options for plant specific programs are recommended for the extended operation.

Dominion first determined the in-scope subcomponents requiring an AMR by evaluating the SSCs that comprise an ISFSI against the scoping criteria definition provided in the comments on the Preliminary Guidance for License Renewal for Site-Specific Independent Spent Fuel Storage Installations (ISFSIs) [NRC, 2001]:

> Any SSC that meets either of the criteria shall be evaluated further in the aging management review (AMR) process described later. The categories of SSCs are those that are:
>
> 1. Important to safety; that is, the SSCs are relied on to perform any of the following functions:
>   i. Maintain the conditions required to store spent fuel safely.
>   ii. Prevent damage to the spent fuel during handling and storage.
>   iii. Provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public, as identified in the current licensing basis (CLB).
>
> These SSCs ensure that these important safety functions are met: (1) criticality, (2) shielding, (3) confinement, (4) heat transfer, and (5) structural integrity.
>
> 2. Classified as not important to safety, but, according to the CLB, whose failure could prevent an important to safety function from being fulfilled or whose failure as a support SSC could prevent an important to safety function from being fulfilled.
>
> The function performed by an SSC that causes it to be within the scope of license renewal is its intended function.
>
> Also, SSCs which perform ISFSI support functions are generally not within the scope of license renewal.

Applying the above criteria, Dominion concluded that only the dry storage casks and spent fuel assemblies fell within the scope of requiring an aging review for ISFSI license renewal. Other components not considered to have met these two criteria include [Dominion, 2002 Table 2.3-1]:

- Reinforced concrete pad
- Transporter and supporting equipment
- ISFSI pressure monitoring system
- Lighting

- Backup diesel generator
- Diesel fuel tank collection trench
- Security fence

The second step of the AMR process involved the identification of the materials of construction and the environments to which these materials are exposed, for the subcomponents that require an AMR. As suggested by ASTM [2002], Dominion has provided a series of tables identifying the materials of construction.

The third step in Dominion's AMR process involved the identification of the aging effects requiring management. Dominion notes that "[a]ging effects are the manifestation of aging mechanisms. In order to effectively manage an aging effect, it was necessary to first determine the aging mechanisms that are potentially at work for a given material and environment application. Therefore, the AMR process addressed both the aging effects and the associated aging mechanisms." [Dominion 2002 page 3-2]

Dominion determined that there were no activities required to manage the effects of aging.

1.2.3 Examination of a Dry Storage Cask and Components in Dry Storage for ~14 Years

A recent EPRI report has been issued summarizing data collected on a dry storage cask (including the spent fuel inside the cask) that had been in storage for approximately 14 years [EPRI, 2002]. This report summarizes work co-sponsored by EPRI, NRC's Office of Research, DOE-EM, and DOE-RW. The storage system chosen for evaluation was a CASTOR V/21 metal cask design containing 21 PWR Zircaloy-4™ spent-fuel rods with burnups of up to ~35 GWd/MtU. The broad objectives were to search for signs of degradation in the cask containing the spent-fuel assemblies and in the spent-fuel assemblies, particularly the Zircaloy™ cladding enclosing the fuel pellets. The results from this work, provided in detail in EPRI [2002], will be summarized later in this report insofar as aspects of the report address specific aging issues.

2 ISFSI SAFETY-RELATED FUNCTIONS REQUIRING ASSESSMENT

In support of 10 CFR Part 72, NRC's NUREG-1536 [NRC, 1997] identifies the functions important to safety that the dry cask SSCs must maintain:

- thermal performance
- radiological protection
- confinement
- sub-criticality
- retrievability

Depending on the specific dry cask storage design, particular SSCs are relied upon to maintain each of these functions.

Thermal performance is a function of the ability of cask internal systems (usually the metal components inside the container along with some contribution from the internal gas) to transfer heat to either the outer surface of the internal container or through the outer portions of the metal cask for metal cask designs. Examples of components used to meet the thermal performance criteria are cooling fins, which, for metal casks, are usually fabricated from carbon steel (SA 283 or SA 285 Grade A), copper, or stainless steel (SA 240 Type 304), so as to increase heat transfer. Dry storage system designs using concrete for shielding require heat from the internal basket to be transferred to air that circulates in the cavity between the internal basket and the surrounding mostly concrete portion of the dry storage system. This part of the heat transfer system requires maintaining good air circulation from the vents near the bottom and top of the concrete structure. In systems employing this internal air circulation, licensees are required to perform frequent inspections of the air vents to ensure they remain free from obstructions.

Thus, aging issues related to maintaining long-term thermal performance could include anything causing degradation in the ability of heat transfer from the assemblies to the outside of the internal structure and, for some designs, the ability of external air to freely circulate in the gap between the internal canister and the outer, mostly concrete shielding. Potential long-term degradation mechanisms for the internal canister section would most likely be associated with entrance of air into this section. Air could corrode metal surfaces responsible for heat transfer, and is a poorer heat conductor than helium (for those systems using helium backfill).[3] Air ingress

[3] Note also that prevention of air ingress and egress from the internal portion of the dry storage system is the primary function of "confinement".

could be caused by failure of seals in bolted cask systems, or corrosion through the canister wall in welded metal systems. In the case of bolted systems, double seal systems are employed with a continuous pressure detection system between the seals. Thus, any failure of the seals would be detected. Corrosion of the welded systems is slow, and will likely continue to be slow under extended service as temperatures decrease.

Radiological protection functions (neutron and gamma shielding) are maintained by the continued presence of adequate shielding without any gaps in the shielding. In metal cask systems, gamma shielding is maintained primarily by the thick metal cask walls. Neutron shielding in metal cask designs is provided primarily by organic polymers or other hydrogen- or carbon-bearing materials. Neutron poisons for criticality control also provide indirect neutron shielding. For these designs, long-term performance of the polymers or other neutron shielding materials must be demonstrated. Issues such as shrinkage or cracking of these neutron shielding materials need to be addressed. Concrete provides both neutron and gamma shielding in concrete systems. Cracking, spalling, or excessive dry-out of the concrete could lead to diminished neutron shielding and potential neutron streaming. External inspections of the concrete cask designs are required, and should detect significant cracking or spalling.

The confinement function is to ensure no radioactive gases or solids are released from the containment system. Air ingress into the cavity containing the spent fuel is to be avoided. Potential mechanisms leading to air ingress were briefly discussed above relative to maintaining thermal performance.

Maintaining sub-criticality requires maintaining the initial storage geometry and, in some designs, maintaining sufficient neutron poison capability. Storage geometry requires continued structural integrity. Barring off-normal or accident scenarios leading to mechanical damage, it is likely that only corrosion could lead to loss of structural integrity. As discussed above, significant internal corrosion requires air ingress. Maintenance of sufficient neutron poison will need to be demonstrated.

Ensuring adequate spent fuel retrievability after extended storage requires maintenance of geometry of both the cask internals, as well as the spent fuel itself. Again barring off-normal or accident scenarios leading to mechanical damage, only internal corrosion might cause degradation of internal cask geometry or adequate buildup of corrosion products that may prevent easy removal of spent fuel. Potential spent fuel geometry changes and internal corrosion have been discussed in EPRI [1998], and will be summarized later in this report. The EPRI [1998] report concluded that these two processes were unlikely to prevent spent fuel retrievability.

ASTM [2002] notes:

> Materials for extended service must meet the design and performance requirements given in 10 CFR 72. The DCSS has been designed to store spent fuel safely for a minimum of 20 years and to permit maintenance as required in the original licensed term. Structures, systems and components important to safety have been designed, fabricated, erected and tested to meet standards commensurate with their function and their importance to the safety of the overall system.

ASTM [2002] continues to suggest that additional storage time is unlikely to place additional demands on performance, but is more likely to reduce such demands:

> The service conditions for the renewal period may be less severe than those of the initial licensing period. If the cask contains its original SNF, then the demands on materials properties for an additional 20 – 80 years of storage may be reduced due to lower temperatures and radiation levels. The general assumption put forth here regarding decreases in thermal and radiation conditions are based on the expectation that reloading of SNF does not occur. It is assumed that at the time of license renewal, the reloading of casks (with SNF different from that originally stored in a cask) is very unlikely.

ASTM [2002] also discusses the process for determining and dealing with potential degradation issues deemed “significant”:

> Any age-related degradation mechanism is considered significant if, when allowed to continue without any additional prevention or mitigation measure, it cannot be shown that the component would maintain its safety function during the license renewal period or extended operation following the initial license. The potential for significant age-related degradation of specific component evaluated is dependent upon design features, design basis, operating history, and the extent to which they are susceptible to the age-related degradation mechanism(s). If it can be shown that a DCSS component is not affected by the degradation mechanism under consideration, or is only affected to such a small degree that the component safety function is not adversely affected throughout the license renewal term, then the component/degradation combination is not significant. Otherwise, the combination is potentially significant. If a potentially significant problem (component/degradation mechanism combination) is adequately addressed by effective existing programs, then the issue is not a license renewal concern as it is considered to be resolved on the basis that the degradation is managed acceptably. Combination of mechanisms and components for which existing programs cannot be shown to manage potentially significant age-related degradation will require plant specific enhanced monitoring program to effectively manage the age-related degradation phenomena. [ASTM, 2002, Appendix E]

3 POTENTIAL DEGRADATION MECHANISMS – A REVIEW

There are a variety of potential degradation mechanisms that could conceivably compromise the ability of the ISFSI to maintain its safety functions. While there is a large literature on degradation mechanisms affecting dry storage system SSCs, a few references summarize the more important potential degradation mechanisms. ASTM [2002] and EPRI [1998] provide a review of potential spent fuel degradation mechanisms in dry storage. ASTM [2002] and Dominion [2002] discuss potential degradation mechanisms for metal storage systems. TN [2001] (among others), ASTM [2002], NEA [2002], and EPRI [1994] discuss potential concrete degradation mechanisms at nuclear facilities.

The following sections in this chapter provide a review of some of the potential degradation mechanisms that may be important to address when considering extended ISFSI service. This review largely draws upon the documents cited above.

3.1 Potential Spent Fuel Degradation Mechanisms [EPRI, 1998]

At the time of initial storage, the ability of the system to meet the regulatory requirements is determined by applying benched mark codes using the expected conditions of storage (temperature distribution, atmosphere), and the condition of the fuel as it was irradiated. During the initial storage period (20 y) the condition of the fuel rod might change, and the storage conditions will become less severe (lower temperature and radiation field). The bench-marked codes will still be valid, although some of the potential effects in the initial storage period will no longer be active in extended storage. For example, any radiolysis effects on residual water or fuel oxidation from residual air will have long been exhausted. To determine if continued storage meets the regulatory requirements, the codes for those mechanisms still active must be applied using the condition of the fuel rods at the end of the initial storage period and the storage conditions expected during extended storage.

In most cases, the fuel will be subjected to "normal" conditions during both the initial storage period and extended storage. The fuel initially will be dried to remove the majority of the cask moisture. There will be either a He or N_2 inert atmosphere. The temperature, governed by the decay heat and loading pattern will be no higher than 400°C [USNRC 2002] and decrease with time. The radiation field will be < 10^5 R/hr (γ) with a neutron flux of ~ 10^4 to 10^6 n/cm^2-s, which is substantially less than in-reactor fluxes. After 20 years the temperature will be in the 120-150°C range and the radiation fields will be lower.

In rare instances during either storage period, "off-normal" events might occur due to a seal leak, or an over-temperature excursion due to a blocked vent. In this case air might get into the cask. If per chance, the cask was mistakenly misfilled with air, the condition is also off-normal.

Even rarer are accidents such as cask drop, cask tip-over, airflow blockage, and fire that could lead to fuel damage, or leakage of the confinement boundary. Some natural phenomena events include flood, tornado, earthquake, burial under debris, lighting, seiche, tsunami, and hurricane. Of particular concern due to the events of 9/11 are the effects of intentional projectiles. If one of these accidents occurred, the condition of the system would have to be reviewed to determine if storage could continue for the remainder of the initial storage period or if the fuel needed to be put into a new canister at that time.

Under all the above conditions, there are mechanisms that may change the condition of the fuel rod, increase the radiation source term in case of a cladding breach, or compromise the ability to meet the requirements and functions stated earlier in this report. If off-normal/accident conditions exist, the storage conditions of the SNF could change due to either air or water ingress, or excessive temperature excursion. In addition, the possibility of fuel damage by mechanical trauma also may occur. If there is air ingress, oxidation of the Zircaloy™ cladding and fuel may occur. These potential changes are shown in Table 3-1. The mechanisms that may lead to these changes are discussed briefly. These mechanisms must be analyzed to determine if they: 1) have caused fuel rod change during the initial storage period, 2) are still active during the extended storage period, and 3) sufficiently degrade the fuel to the point where the regulatory concerns are not met during the extended storage period.

Table 3-1
Potential Rod Changes and Mechanisms during Storage

Storage Condition	Fuel	Cladding
Normal	Increased stress from fgr	Breach (creep, DCCG, H_2 reorientation)
		Change in mechanical properties (annealing, H_2 migration)
		Crud contamination (spallation)
Off-Normal	Particle size reduction (oxidation)	Wall thinning (oxidation)
	Gas-release (oxidation)	Excessive creep
	Increased cover gas conductivity	Mechanical property change (annealing)
Accident		
Impact	Fuel fracture	Breach
	Gas release (oxidation)	Oxidation (corrosion)
	Particulate formation (oxidation)	Crud contamination (spallation)
Fire	Stress increase due to thermal expansion; strength decrease	Breach (stress rupture)
		Change in mechanical properties (annealing, H_2 reorientation)

DCCG = Diffusion Controlled Cavitation Growth
fgr = fission gas release

The extent that these mechanisms are expected to be active for either initial storage or extended storage is given in Table 3-2. **Yes** indicates that the mechanism could be expected to be active for the fuel or cladding during that time frame. In the case of the initial time frame the mechanism should have been evaluated in the original license submission. This initial evaluation along with any new support material since the submission should establish the initial fuel conditions for evaluation of the extended storage period. **No** indicates that the mechanism is not expected to be active either because the atmospheric conditions or material condition is not suitable. In some cases the mechanism is potentially active only for certain events. These entries are indicated with a **DE**. Justifications for the entries in Table 3-2 are given in the next section

Table 3-2
Potentially Active Degradation Mechanisms

Mechanism	Fuel		Cladding	
	Initial	**Extended**	**Initial**	**Extended**
Normal				
Oxidation	N	N		
fgr	Y	N		
Creep,			Y	N
DCCG			N	N
H_2 reorientation			Y	N
DHC			N	N
SCC			May be	N
H_2 embrittlement			N	Y
H_2 migration			Y	N
Annealing			Y	N
Crud Spallation			May be	N
Off-Normal				
Oxidation due to air ingress	Y	Y	Y	Y
Creep			Y	DE
Annealing			Y	DE
Accident-impact				
Fracture	DE	DE		
Oxidation	DE	DE		
Impact Breach			DE	DE
Crud spallation			DE	DE
Accident-fire				
Stress rupture			Y	Y
Annealing			Y	Y
H_2 reorientation			Y	Y

DHC = Delayed Hydrogen Cracking
DCCG = Diffusion Controlled Cavitation Growth
Fgr = fission gas release
SCC = Stress Corrosion Cracking
DE = Depends on the event

3.1.1 Cladding Degradation Mechanism

3.1.1.1 Creep

One criterion for consideration of cladding integrity during inert dry storage is creep. To avoid degradation of cladding, the strain calculated to occur in storage should be less than the creep strain to failure. In creep tests at temperatures between 250-400°C of Zircaloy™ cladding irradiated up to burnup of 64 GWd/MtU, no failures have been observed below 2% strain [Spilker 1997, Goll 2001, EPRI 2002]. Therefore, a conservative cladding strain limit of one percent has been used in several countries, such as Germany.

Creep is the progressive deformation of a material under an applied stress. Creep occurs in three stages. The primary stage has rapid deformation and a decrease in creep rate over time, the secondary stage has a constant creep rate and the tertiary stage has a rapid creep rate increase with time until fracture occurs. The creep strain rate and strain at failure of spent nuclear fuel cladding are affected by material parameters like alloy composition, fabrication steps (e.g., cold work, solution anneal, recrystallization anneal), hydride content, and radiation fluence. For irradiated cladding, radiation effects overshadow these fabrication and chemical effects. The two principal factors in the creep behaviour of irradiated cladding are the hoop stress and the temperature. The hoop stress results from the rod internal pressure, and the temperature results from the decay heat of the fuel assemblies. In general, the creep strain can be calculated from creep equations, but their applicability for a particular material or set of materials parameters should be questioned and not applied without consideration of all important factors

At low temperatures and stresses, the deformation (strain) is negligible and can be ignored; at high temperatures and stresses the strain can be substantial. For typical fuel cladding hoop stresses, strain may be detected at temperatures above about 300°C although significant strain (for zirconium alloy cladding) is not expected to occur until the temperature is well in excess of 350°C. Over long storage times, both the pressure and the temperature decrease such that the strain rate tends to zero. At temperatures below 300°C, creep may be considered to be immeasurably slow and is not a factor in extended storage under normal operation.

3.1.1.2 Diffusion Controlled Cavity Growth (DCCG)

Theoretically, diffusion controlled cavity growth is a potential mechanism for mechanical degradation, but one that has never been observed on actual SNF cladding and its potential effect on degradation of the cladding in a ISFSI is expected to be very small. The occurrence of failures from DCCG has not been observed to-date, either in Zircaloy™ or in any components of dry storage facilities. It has been calculated that if DCCG does not occur during the initial storage period, it will not occur during extended storage due to the lower temperature. The NRC no longer considers this a viable mechanism for spent fuel cladding breach in dry storage and need not be considered for extended storage.

3.1.1.3 Hydrogen Effects

As a result of corrosion during irradiation the hydrogen concentration can increase to values in excess of 300 microgram per gram (for higher burnup fuels, the concentrations may be considerably higher) and hence result in hydride formation and precipitation. This may increase if there is any radiolysis of moisture in the cask. Approximately 10-20% of the hydrogen produced by the radiolysis and corrosion will enter the cladding. The distribution of the hydrogen in the rod can change if there is migration down the temperature gradients during the initial storage period. The brittle zirconium hydrides formed can impact the mechanical properties of the Zircaloy™, generally increasing the strength and decreasing the ductility. This may be exacerbated if the hydrides reorient to the radial direction during the storage. In addition, the hydrogen may activate such breach mechanisms as hydride embrittlement and delayed hydride cracking (DHC) [Huang 1994. Northwood 1983, Simpson 1979a, b].

Hydride embrittlement occurs when there are sufficient hydrides to cause detrimental effects to mechanical properties, including tensile ductility, fracture toughness and ultimate fracture strength. The amount of ductility degradation depends on hydride orientation, concentration and distribution. Hydride platelets oriented normal to the stress direction cause large reductions in strength and ductility, whereas hydride platelets oriented parallel to the stress direction have little effect.

Commercial SNF is manufactured with hydrides predominately oriented in the circumferential direction [Kawanishi 1974]. However under sufficient stress, hydrides will reorient to the radial direction [Kawanishi 1974, Coleman 1985, Einziger 1984, Cunningham 1987]. The radial hydride orientation is more detrimental than circumferential hydrides. As the cladding is heated during the drying of the cask and initial storage at higher temperatures, much of the hydrogen introduced into the cladding during irradiation goes into solid solution in the Zircaloy™ matrix. As the cladding cools during storage, hydrogen in solid solution will precipitate as hydrides as the solubility limit is exceeded. Depending on the stress levels in the cladding, texture and cooling rates, these hydrides may be circumferential or radial in orientation. The hoop stress needed for hydride reorientation in unirradiated Zircaloy is not well defined (35 to 140 MPa at 300 to 400°C) and depends on such things as fabrication history, basal pole texture, hydrogen content and temperature. Recent unpublished evaluations of reorientation work on irradiated cladding indicate that the stress threshold is temperature-dependent and higher than in unirradiated material.[4] The stress required for reorientation appears to increase with decreasing temperature.

Recently, NRC [2002] suggests that hydride reorientation may be an important mechanism adversely affecting spent fuel cladding integrity for spent fuel with higher burnup levels. This mechanism is partially dependent on the amount of zirconium hydrides that have reprecipitated from their initial solution within the cladding at the initially high temperatures. The lower the temperature, the more zirconium hydride that can reprecipitate. Thus, the successively lower temperatures associated with extended storage can lead to more hydride reprecipitation.

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[4] Based on unpublished analysis of reorientation experiment performed independently by K Gruss (NRC) and Eric Siegmann (Framatome)

The amount of hydrogen necessary to severely reduce ductility of SNF cladding depends on the storage service temperature and the orientation of the hydrides. The current understanding suggests that at temperatures around 300-400°C radiation damage determines ductility loss, while at room temperature the effects of hydrogen and radiation damage are additive. [Wisner 1998] As low as 30 ppm hydrogen may be required to reduce ductility at room temperature [Bai 1991, Huang 1994] while over 600 ppm [Bai 1991] may be required at 300°C. Hydrogen embrittlement is one of the few mechanisms where decreased temperature is detrimental. This mechanism probably is not an active during the initial storage period but must be considered during extended storage where the temperature is low. This embrittlement must be also considered when evaluating the potential cladding fracture during an impact accident in extended storage.

Delayed hydride cracking (DHC) is occurs by diffusion of hydrogen to a flaw region. The fracture process requires the formation of a hydride zone, growth of a flaw, and subsequent fracture of the brittle hydride zone. The tensile stress region provides the driving force for diffusion of hydrogen atoms. When the hydrogen concentration exceeds the solubility limit, hydrides will start to form and grow. When a critical size under a sufficient tensile stress is reached, fracture through this zone can occur eventually leading to failure. Initiation of DHC occurs only if the stress intensity is above a threshold value, and stress intensities in inert dry storage conditions are expected to be lower than this critical stress intensity. This mechanism is not important for extended storage.

3.1.1.4 Stress Corrosion Cracking

Stress Corrosion Cracking (SCC) is similar to DHC except migration of chemical impurities such as fission products, instead of hydrogen, migration to the cladding flaw is necessary in addition to an applied stress. Pescatore [1994] reviewed the testing done on SCC of irradiated Zircaloy™ and determined that under conditions expected for dry storage that failure would not occur by this mechanism. Even under conditions of higher than expected stress, laboratory testing only produced non-detrimental pinhole breaches. The stress on the cladding is highest at the start of storage and decreases with storage time due to a decreasing temperature. Due to the lower fuel temperature in storage than in reactor, no corrosive fission products will be released from the fuel pellets to the cladding gap during the storage period. Therefore if the unexpected SCC is going to occur in a particular fuel rod, it will occur early in storage, when stress is highest, not later during the extension period.

3.1.1.5 Oxidation

Excessive oxidation of the Zircaloy™ cladding combined with an internal stress can potentially breach the cladding. Cladding failure could result in a risk of contamination of the interior of the confinement vessel and needs to be evaluated for retrievability.

Oxidation of the Zircaloy™ [Rothman 1984] is a thermally-facilitated process. Dry storage under inert gas conditions leads to no further increase in the oxide layer over and above the condition of final discharge from the reactor, since the storage conditions rule out the presence of oxidizing substances. In the event of air ingress, dry oxidation at storage temperatures below

300°C is of no concern since the extent of oxidation is much smaller than the cladding thickness [EPRI 1998].[5] Oxidation of the cladding needs to be evaluated to determine the ramifications of a fire that might exceed 300°C should a concurrent cask breach admit an air atmosphere. Since the consequences would be independent of when the fire occurred, the analysis conducted for the initial storage period should suffice for the extended storage period.

Any water left in a sealed system due to incomplete drying will result in minimal oxidation of the cladding and hydrogen input into the cladding [EPRI 1998]. This would happen in the initial storage period and be of no additional concern to extended storage. Furthermore, this mechanism has been addressed in the initial license application, so presents no new issues for extended storage as long as the system remains sealed. Since oxidation of the cladding and solubility of hydrogen in the cladding both decreases with temperature, water ingress is a degradation mechanism during extended storage only in the case of an accident that results in a water ingress concurrently with a fire that raises the temperature significantly. This is highly unlikely.

3.1.1.6 Annealing

When in the reactor, the cladding is subjected to both a neutron and gamma flux. The mechanical properties of materials used in fuel assemblies need to be evaluated to determine: 1) whether substantial changes in these properties would be expected to occur during the initial 20-year period after closure; and 2) whether substantial changes would be expected to occur in the 20- to 100-year period after closure. Mechanical properties of these materials could be affected by the elevated temperatures expected during the initial storage period.

The properties of Zircaloy cladding are affected by irradiation. The effects depend on composition, heat-treatment, cold work, and texture. However, the radiation fields to which the cladding is exposed during storage are many orders of magnitude lower than those already encountered in-reactor and do not change the effects that take place in the reactor, which usually saturate during the first or second irradiation cycle. For this reason, changes of yield strength, elastic modulus, and fracture toughness due to the radiation field from the fuel are expected to be small during the storage period [Sanders 1991a, b]. The fracture toughness may be slightly increased by irradiation [Walker 1974], and this would be expected to improve the cladding's resistance to fracture under dynamic loading, such as in an accident scenario. In addition, any improvement in fracture toughness would help to limit the growth of existing cracks or pinhole leaks.

Although the strength of the cladding may be increased by irradiation, the temperature may be high enough during the initial part of the storage period to cause some annealing of radiation induced hardening to occur. The result of annealing is to decrease the strength and increase the ductility of the material. It may also assist in relaxing internal stresses [Sanders 1991a, b]. The extent of annealing depends on the irradiation fluence, temperature, and time. Although the annealing curves are not well established, if annealing occurs it will be in the initial storage

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[5] Spent fuel in a test dry storage cask at INEEL was exposed to a partial air environment for approximately six months, before the air was replaced with helium [EPRI, 1986]. While not measured, a reasonable estimate of container temperatures during the time air was present would be in the range of 300°C. Subsequent examination of the spent fuel yielded no evidence of significant additional oxidation [EPRI, 2002].

period when the fuel is hot. During extended storage, the fuel will be too cool for additional annealing to occur [Einziger 1982].

3.1.1.7 Crud Spallation

During reactor operation, a coating of crud developed from corrosion of the materials in the primary system forms on the fuel rod. This crud contains a number of isotopes, but after a short time, the main constituent is ^{60}Co. The crud tends to be very tenacious on PWR rods and much more flaking and easily dislodged on BWR rods. In some instances, BWR rods have been known to dislodge so much crud when moved around in the pool that diminished pool clarity occurred. A definitive compilation on the properties of crud is given in Sandoval [1991]. Information is given on composition, phases, radionuclide content, and distribution. Once the crud spalls, only a small fraction will be available for gaseous transport. The remainder will settle quickly on other surfaces within the cask. The degree of settling will depend on the particle size distribution of the crud.

The NRC mandates the use of a spallation factor of 0.15. [USNRC 1997] There is no published systematic study of the adhesion or cohesion of crud from dry out or impact. The tenacity of the crud may be affected by the length of time the fuel is stored in the water pool and then, subsequently, the amount of time the crud is dried during dry storage. Estimates of crud spallation from rods stored at 230°C ranged from 1.6 to 4.8%. These rods had seen quite a bit of movement before the spallation measurements, and it is not possible to determine how much crud spalled off during the movements. These measurements are probably reasonable for static dry out. A more complete description of the supporting data is given in Sandoval [1991].

Crud spallation will be a concern if it occurs excessively during normal operation during the initial storage period where fuel rod drying occurs. No additional crud spallation would be expected during extended storage and the analysis for the initial license application should be sufficient.

3.1.2 Fuel Degradation Mechanisms

3.1.2.1 Fragmentation

An accident may exert sufficient force on the fuel to shatter the cladding and fragment the fuel. This will result in respirable particulate that may impede retrievability. This is of special concern in the case of a man-made projectile. Very little is known about the energy needed to fragment spent UO_2 or the size distribution of the particulate. Analogies are made to other ceramic materials. Data on this subject has been collated in Sanders [1992]. Fragmentation concerns during extended storage are the same as during the initial storage period and would be covered in the initial license application.

3.1.2.2 Oxidation

Oxidation of the fuel is a thermally-activated process governed by the time-at-temperature that fuel is exposed directly to air in the DCSS, and the amount of oxidant available. A comprehensive review of the mechanisms and kinetics of fuel oxidation has been performed [McEachern 1998]. Oxidation typically occurs first along the fuel grain boundaries. Thus, the release of fission gases, such as ^{85}Kr, to the DCSS relatively early during this transition is possible. Upon further exposure to air, the fuel is oxidized to U_3O_8, a phase which is approximately 36% less dense than the original fuel. As the fuel continues to oxidize it swells and splits the cladding [Einziger 1985, Einziger 1986]. U_3O_8 formed by oxidation of the fuel is a fine powder that spalls from the fuel surface.

Limited access of air can be assumed if the seals of the DCSS failed during the first (original licensed) dry storage period or if the system assessment indicates the potential of the sealing system leaking in the license renewal period. Small leaks due to seal failures, where the amount of air ingress has been shown to be minimal, would result in only minor oxidation of either the fuel or cladding, with limited consequences [Sanders 1992]. Water ingress, most likely with an air atmosphere, may raise the potential for additional oxidation of the fuel due to the radiolysis of a humid atmosphere. Water ingress related to oxidation processes will be a concern only during the initial storage period when the temperature is elevated and should not be a concern during the extended storage period when the temperature is low.

During the initial license period, when temperatures within the DCSS are higher, the potential for adverse consequences due to oxidation is greater than during the extended license period. Should there be failed rods in the DCSS at this time and a large leak due to an accident or the cask is misfilled with air, there is sufficient oxidant to oxidize 1-2 rods depending on the size of the cask [EPRI 1998] in a short time frame. At the lower temperatures expected during extended storage, oxidation of even low-burnup fuels will take sufficiently long, at least months or years, that detection and mitigation of the off-normal condition should prevent fuel oxidation from being a concern. Of course fuel oxidation is not a concern during normal operation during extended storage.

3.2 Potential Dry Storage System Degradation Mechanisms

3.2.1 Temperature-Related Degradation

Decay heat during storage will increase temperatures in the ISFSI. Temperature-related issues have been addressed in the safety analyses supporting the initial ISFSI licenses. Unless heat removal capabilities are significantly degraded during storage, it is expected that temperatures will decrease with prolonged storage. For example, Dominion [2002] states:

> Following initial cask loading, the fuel temperature inside the fuel cladding is expected to be less than 662°F (350°C) except in some localized areas where the temperature may be slightly higher during the first few years after cask loading. After 20 years of dry storage, the fuel cladding temperature is expected to be less than 347°F (175°C) and to decrease to less than 248°F (120°C) after several years of extended storage.

Higher temperatures will enhance thermally-activated degradation processes. Relevant processes for ISFSIs include, for example:

- Fuel cladding creep caused by increased cladding ductility and increased stress (due to higher temperatures causing higher pressures inside the cladding);
- Hydride reorientation in the spent fuel cladding;
- Increased corrosion of metals;
- Degradation of neutron shielding materials; and
- Concrete dryout and cracking.

Temperature-related issues for the spent fuel were discussed in Section 3.1.

3.2.1.1 Corrosion at Elevated Temperatures

Corrosion as a general mechanism will be discussed in Section 3.2.2. Enhanced corrosion due to higher temperatures has already been taken into account in the safety analysis reports supporting the initial ISFSI licenses [see, for example, TN 2002]. Decreased temperatures associated with extended ISFSI operation will lower corrosion rates even further.

3.2.1.2 Neutron Shielding Degradation due to High Temperatures

The nature of the degradation of neutron shielding materials at higher temperatures depends on the particular material. For example, polyethylene rods may experience some shrinkage, which could lead to significant local loss of neutron shielding. Other neutron shielding materials can experience loss of hydrogen or water at higher temperatures. Similarly, degradation of neutron shielding materials due to high temperatures has also been taken into account in the initial ISFSI licenses. The lower temperatures associated with extended storage will likely lead to little to no additional degradation.

3.2.1.3 Concrete Degradation due to Higher Temperatures

A review of concrete behavior under sustained elevated temperatures was provided in TN [2001], and ASTM [2002]. A summary of the discussion in these two reports is provided here.

Changes in concrete properties at elevated temperatures are primarily caused by loss of free water from the concrete pores, surfaces, or matrix. As temperatures increase, water is lost from the following five regions within the concrete, approximately in this order:

1. Water in capillary pores;
2. Water in gel pores;
3. Water adsorbed on crystal surfaces;
4. Adsorbed water confined between adjacent crystal surfaces; and
5. Zeolitic intracrystalline water.

At a temperature of 175°F (~80°C) some of the capillary water will be lost, although cement hydration will continue. Loss of pore water is complete at a temperature somewhat above 100°C. Loss of adsorbed water occurs over a wide range of temperatures with all evaporable water lost at approximately 300°C [Allen, 1984]. Microstructural changes in the concrete occur at this temperature.

Concrete strength often improves with somewhat elevated temperatures due to loss of pore water. This is caused by an increase in capillary forces and a loss of lubrication causing more resistance to sliding. Concrete degradation, characterized by crack formation and subsequent spalling, is mostly attributable to loss of chemically combined water (dehydration). Lankard et al. [1970] note that, for tests on unsealed concrete, chemically combined water content of the cement phase increases somewhat at 175°F (~80°C), while a 10, 20, and 40 percent loss of this water occurs at 250, 375, and 500°F (121, 191, and 260°C), respectively. The dehydration reaction leads to weakening of the bond between the gel and cement phases, leading to a decrease in strength, affecting the compressive strength, modulus of elasticity, tensile strength, Poisson ratio, and creep.

Tests performed by Bertero and Polivka [1972] report that, "If the free moisture is allowed to escape during heating to 300°F, the mechanical characteristics of the concrete are very little affected by the heat treatment". TN [2001] notes that several other tests reached the same conclusion.

Peak concrete operating temperatures in concrete dry storage systems are below approximately 200°F (~95°C). Thus, it appears that little, if any strength loss will have occurred during normal and probably most off-normal conditions, as well. Furthermore, the dehydration reactions leading to reduced strength are reversible if water is reintroduced. Reintroduction of water is likely to occur during extended outdoor storage as storage temperatures decrease with time and as the casks are exposed to moisture. Thus, it appears that elevated concrete temperatures during the initial storage period do not pose any loss of strength issues during extended storage operation. In fact, there may be some concrete strength improvement during extended storage.

3.2.2 Corrosion

Corrosion of materials is an issue for materials exposed to oxygen – especially in humid environments. Corrosion inside a sealed container was discussed in EPRI [1998] and summarized in Section 3.1. As long as no air has entered the container, even small amounts of residual water are not likely to cause significant degradation of the spent fuel cladding or other components inside the container.

Corrosion of metal components exposed to air and humid environments will occur. These mechanisms have been addressed in the safety analysis reports accompanying the initial ISFSI licenses. Corrosion has occasionally been an issue in ISFSIs. For example, corrosion around the external seals on a few welded containers led to loss of the external seal, as discussed in Section 4.1.2 below. However, the loss of seal was detected through the use of continuous pressure monitoring between the inner and outer seals. The problem causing corrosion of the seal in this instance was linked to a weather cover design that allowed moisture to collect next to the seals. The weather cover design has since been changed to avoid this occurrence.

However, for the relatively thick metal components of the ISFSIs corrosion rates are thought to be sufficiently slow to be of little concern. For example, NRC is sufficiently confident that corrosion of the inner metal container of welded container designs will not occur that it does not require any monitoring of the container:

> [T]he NRC does not consider continuous monitoring for … double-weld seals to be necessary because:
>
> 1. There are no known long-term degradation mechanisms which would cause the seal to fail within the design life of the [canister containing the spent fuel][6]; and
>
> 2. The possibility of corrosion has been included in the design.
>
> These conditions ensure that the internal helium atmosphere will remain stable. Therefore, an individual continuous monitoring device for each [canister] is not necessary. However, the NRC considers that other forms of monitoring, including periodic surveillance, inspections and survey requirements, and survey requirements, and application of preexisting radiological environmental monitoring programs of 10 CFR Part 50 during the use of the canisters with seal weld closures can adequately satisfy NRC requirements. [Federal Register, 1994]

This is because NRC notes that the general corrosion rates for stainless steels like those used above were less than 0.00001 inch per year [Federal Register, 1994]. Assuming this upper-end corrosion rate was constant in time, the corrosion depth after even 100 years would be 0.001 inch. Thus, additional stainless steel corrosion during extended storage is not a concern.

The CASTOR metal cask design includes a cask body and cooling fins made of cast iron. Cast iron corrosion rates are generally higher than that for stainless steels. Dominion [2002] notes that regular inspection of the outside of the cask can be used to check for corrosion that could lead to degradation of some safety functions.

Similarly, Dominion [2002] notes that metal trunnions and trunnions bolts can be visually inspected for signs of excessive corrosion. This is also the case for the metal cask designs using carbon steel primary seal covers.

Dominion [2002] considered corrosion of the polyethylene neutron shielding. Dominion determined this corrosion would be arrested due to loss of oxygen as the small amount of oxygen remaining inside the region into which the polyethylene was inserted at the time of cask construction reacts with the surrounding metal or the polyethylene itself. Therefore, it is not a long-term degradation issue.

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[6] As was mentioned earlier, design lifetimes are typically significantly longer than 20 years.

3.2.3 Metal Fatigue

Metal fatigue under normal operations is caused by daily temperature fluctuations. Dominion [2002] did calculate fatigue factors for extended storage and concluded the additional fatigue will still be within safe limits.

3.2.4 Radiation-Induced Degradation

Radiation may cause enhanced corrosion due to radiolysis. Neutron radiation will cause a portion of any neutron shield to be used up. ASTM [2002] notes that effects of radiation on neutron shielding and steels are expected to be small:

> After 20 years of dry storage, the fast neutron fluence at the interior of the DCSS [dry cask storage system] is typically on the order of 10^{14} n/cm^2 and the cumulative gamma dose is on the order of 10^9 rad. The radiation shielding within a DCSS absorbs neutrons and decreases the exposure levels and the potential damage to the materials of the exterior components but, in general, at this fluence level the effects on materials of interest are small. These levels of neutron fluence could potentially have some effects on mechanical properties of steels, but not for any austenitic materials. The ferritic materials would require at least several orders of magnitude greater neutron fluence to have any significant effect on mechanical properties [Steel-1996] and the effects would be limited to those on impact properties, i.e. on either on the upper-shelf energy absorption or on the transition temperature behaviors.

Dominion [2002] determined that the amount of neutron poison used up by the neutron fluence during the first 20 years of storage was negligible. Thus, extended storage under neutron fluxes significantly lower than during the first 20 years will also likely have only a minimal effect on the available neutron poison:

> A small fraction of the original Boron-10 (neutron poison material) could be consumed over time by the $B^{10}(n,\alpha)Li^{7}$ reaction, resulting from spontaneous fission within the spent fuel. A calculation was performed to demonstrate that there is sufficient neutron poison material remaining over the additional 40-year life with the pure water present in the cask cavity and that the TSAR conclusions do not change for the license period of 60 years. The affect on criticality due to depletion of the Boron-10 in the Boral™ plates due to spontaneous fission has been re-analyzed and projected to be valid for the license renewal period.

Radiation-induced hardening of metals can affect their null ductility transition (NDT) temperature. Dominion [2002] states, “Based on testing, no shift is expected in NDT temperature below the irradiation value of 10^{17} Neutrons /cm^2. Since the neutron fluence for sixty years is calculated to be 2.2×10^{14} Neutrons/cm^2, it is concluded that there will be no shift in NDT temperature.”

Concrete also exhibits degradation at high radiation levels. For example, concrete may exhibit aggregate growth if exposed to high neutron levels, while high gamma radiation may affect the cement pasts producing heat and driving off free water [ASTM, 2002]. However, ASTM [2002] also notes that "the level of irradiation [in concrete ISFSI systems] over the extended operation is not expected to reach a level that is sufficient to cause significant mechanical strength reduction of concrete, therefore, irradiation effect is not considered to be a significant aging degradation mechanism."

3.2.5 Other Concrete Degradation Mechanisms

The concrete typically used in DCSS consists of Type II Portland cement because of its higher sulfate resistance and lower heat of hydration relative to the general purpose Type I Portland cement. All concrete used in an ISFSI is reinforced with embedded bars, wires, strands, or other slender members. The safety functions of concrete include maintaining subcriticality, containing radioactive material, providing radiological protection, and maintaining retrievability of the spent fuel.

ASTM [2002] Annex E provides an excellent review of concrete degradation mechanisms:

> Portland cement has been used in concrete structures since the early 1800s, and it's proven durability to aging-related effects is one of the main reasons for its widespread use in buildings and industrial facilities. However, concrete that is exposed for an extended period of time to extreme environments such as freeze-thaw, aggressive chemicals, flowing abrasive fluids, and elevated temperature may experience significant aging-related degradation in performance. The aging-related degradation effects that are evaluated for all concrete and the steel reinforcements include loss of material, cracking, and change in material properties.
>
> Loss of material in concrete structures or components is characterized as scaling, spalling, rust staining, pitting, and erosion. These effects result from one or more of the following aging-related degradation mechanisms: freeze-thaw, abrasion and cavitation, elevated temperature, aggressive chemical attack, and corrosion of reinforcement steels.
>
> Cracking in concrete structures or components is characterized as general cracking, map cracking, hairline cracking, pitting, and erosion. These effects result from one or more of the following aging degradation mechanisms: freeze-thaw, reaction with aggregates, shrinkage, settlement, elevated temperature, irradiation, and fatigue.
>
> Change in material properties is evidenced in concrete structures or components as increases in permeability and porosity of concrete material, and reductions in pH value, tensile strength, compressive strength, modulus of elasticity, and bond strength of concrete. These effects result from one or more of the following aging degradation mechanisms: leaching of calcium hydroxide, aggressive chemical attack, elevated temperature, irradiation, and creep.
>
> Freeze-thaw damage typically occurs on relatively flat surfaces such as pavement, where water remains in contact with the concrete. Freeze-thaw damage often originates at the surface and therefore is readily detected by visual inspections. The effects of freeze-thaw damage are local in nature and by itself will not affect the strength of the concrete.

When water passes through cracks, inadequately prepared construction joints, or inadequately consolidated areas during concrete placement, the calcium compounds in the concrete can be dissolved due to presence of small amount of calcium ions in water. Since the most readily soluble calcium compounds in concrete is calcium hydroxide or lime, water can leach lime from concrete. When calcium hydroxide or lime in concrete is leached away, other cementitious constituents become exposed to chemical decomposition, eventually leaving behind silica and alumina gels with little or no strength.

Concrete is highly alkaline (pH > 12.5) and therefore is vulnerable to degradation by strong acids. Acid attack can increase porosity and permeability of concrete, reduce its alkaline nature at the surface of the attack, reduce strength and render the concrete subject to further degradation. … Sulfates solutions of potassium, sodium, and magnesium sometimes found in groundwater may attack concrete over time. Sulfate attack can produce significant expansive stresses within the concrete, leading to cracking, spalling, and strength loss. Groundwater chemicals may also damage the foundation concrete [EPRI, 1994].

A dense concrete designed and constructed in accordance with [ACI (American Concrete Institute) and ASTM standards] will result in a concrete with low permeability and high resistance to aggressive chemical attack [EPRI, 1994]. If concrete structures or components are not exposed to an aggressive chemical environment, the aging degradation of concrete strength due to chemical attack is considered to be insignificant [EPRI, 1994].

Chemical reactions may develop between certain mineral constituents of aggregates and alkalis that compose the Portland cement. … Seawater and solutions of deicing salt can also inject alkalis into concrete by action of penetration. Three types of chemical reactions may occur depending upon the composition of the aggregates. They are alkali-aggregate reaction, cement-aggregate reaction, and expansive alkalis-carbonate reaction [EPRI, 1994]. Operating history of nuclear power plant concrete structures do not indicate that structural integrity of these concrete structures is significantly affected by alkali-aggregate reactions.

Corrosion of embedded steel can be an aging degradation mechanism for concrete structures. If the concrete is degraded by other aging mechanisms, which may reduce the protective cover of the steel, corrosion may occur at a significantly higher rate. Adequate management of the other aging mechanisms will help control the corrosion of the embedded steel [EPRI, 1994].

When exposed to high levels of fast and slow neutrons, concrete may exhibit aggregate growth, decomposition of water, and thermal warming of concrete. … ANSI/ANS-6.4-1985 reports that for incident energy fluxes less than 10^{10} MeV/cm2-sec, nuclear heating is negligible. … In DCSS systems, the level of irradiation over the extended operation is not expected to reach a level that is sufficient to cause significant mechanical strength reduction of concrete, therefore, irradiation effect is not considered to be a significant aging degradation mechanism.

Excess water is typically added to the concrete mix to improve its workability during forming of concrete structure. Shrinkage of concrete occurs initially during curing as the result of excess water leaving the concrete. This curing period typically continues several months after placement. As excess water evaporates into the surrounding environment, tensile stresses are induced in the concrete due to internal pressure from the capillary action of water movement, and cracks develop. Subsequent drying and shrinkage occurs in concrete for up to about 30 years. Over 90 percent of the shrinkage occurs during the first year and about 98 percent in the first five years [ACI, 1997]. Therefore, shrinkage is not a significant aging degradation mechanism for licensing renewal.

4 EXAMINATIONS AND TESTS TO SUPPORT EXTENDED DRY STORAGE

This chapter discusses current surveillance programs to detect potential ISFSI degradation. It also summarizes the results of a study that was conducted to determine whether degradation in a dry storage system in use for over 14 years had occurred.

4.1 Ongoing Surveillance Programs

Foreign and domestic dry-spent-fuel storage programs are reviewed to see if any experience has been gained, either positive or negative, that will shed light on the expected behavior over a 100-year period.[7]

4.1.1 Surveillance Programs Outside the United States

Six foreign countries are storing UO_2 spent fuel under dry storage conditions. Argentina has stored CANDU fuel in 80 concrete canisters since 1993 [Marticorena 1995]. The temperature is below 200°C; no problems have been identified, and no data on the performance of the fuel are available. Belgium has chosen to store its pressurized water reactor (PWR) fuel in metallic Transnucleaire casks with a helium atmosphere [DeValkeneer 1995]. Fuel has been recently loaded at one of its plants, and no performance data are available. India has been storing boiling water reactor (BWR)-type spent fuel in indigenous casks since 1987 using an air atmosphere. No performance data are available [Schneider 1992, Srinivasan 1995]. Former Russian republics have VVER PWR fuel dry stored in two demonstration casks [Schneider 1992], and Japan is just starting to dry store LWR fuel commercially [Kuriyama 1995]. No performance data are available from these programs, but no adverse or unexpected performance has been reported.

The two major foreign dry storage programs are in Canada and Germany. Canada has both a long-term testing program and a commercial dry-storage program. CANDU fuel has been dry stored in concrete canisters [Pare 1994. Wasywich 1995] using an air atmosphere since 11 canisters of fuel were loaded at Gentilly-1 in 1985. Since then, fuel has been stored at six other sites: Douglas Point (1987–47 canisters), Chalk River (1990–11 canisters), Point Lepreau (1991– ~60 canisters), Wolsong (1991– ~60 canisters) and Gentilly-2 (1995) [Schneider 1992]. Other than temperature, no routine surveillance is conducted. No adverse or unexpected performance has been reported.

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[7] The experience base, other than the discussion of the Dominion seal leaks, has not been updated since the 1998 EPRI report [EPRI 1998].

A program has been underway for the last 18 years at the Whiteshell Nuclear research establishment. Both intact and intentionally defected spent CANDU fuel has been stored in a saturated air or helium atmosphere at 150°C. Periodically, the casks are opened to examine the fuel. A few rods are removed for destructive examination. No degradation or additional failure of the intact rods has been observed. Some oxidation of the fuel from UO_2 to U_4O_9 has been observed in the defected rods. Only trace surface oxidation to the lower density U_3O_8 was seen. The extent of the oxidation depended on the linear heat rating of the fuel. No cladding bulging due to strain from oxidizing fuel was observed, and no handling problems were incurred. The water in the casks became very acidic due to radiolysis of the air and water [Wasywich 1993]. The extent of fuel oxidation was in agreement with oxidation measurements made in the laboratory on used fuel.

Germany has been routinely putting LWR fuel in dry storage using a dry inert atmosphere since 1992 (IAEA 1992). Over 200 metallic casks have been loaded with no problem [Dreisvogt 1995]. There is a 410°C temperature limit for long-term dry storage. No surveillance data are available about the condition of the fuel in these casks. In addition, Germany has been conducting full-scale dry-storage cask demonstrations with irradiated PWR and BWR fuel since 1982. Tests lasted up to two years, at which time cladding creep stopped due to pressure, temperature, and decay heat drops in the fuel rod. Consequently, the Germans think that, since the driving force for cladding degradation has been sufficiently removed after two years, it is of no consequence whether the storage period is 20 or 100 years in an inert atmosphere [Schneider 1992].

Due to the lack of fuel surveillance coupled with the short storage duration the foreign commercial spent nuclear fuel dry storage programs have been positive but provide little support to long-term dry storage. The German argument for support of long-term storage is based on the cessation of the creep process. Although there are obvious differences in burnup and internal gas pressure, both CANDU and LWR fuel are UO_2 based with Zircaloy cladding. Oxidation tests have indicated that a reasonable similarity exists between the behaviors of the two fuels [Taylor 1996]. Thus, the long-term results from the Canadian program are encouraging for the long-term storage of older fuel that remains quasi-isothermal at a fairly low temperature, or fuel that has been stored successfully for 20 years and has a temperature below 150°C. It also indicates that during off-normal and accident events that damages fuel and admits an air atmosphere, oxidation of the fuel and cladding will not occur to any significant extent in extended storage when the temperature remains below 150°C. The behavior of the defected rods is consistent with the behavior expected based on laboratory testing.

4.1.2 Surveillance Programs in the United States

As of early 2002, the United States has licensed dry-storage cask systems deployed in independent spent fuel storage installations (ISFSIs) at approximately 20 sites. Additional sites are expected to have ISFSIs in the near future. The longest operating dry storage has been in 29 casks at the Surry plant and eight at the HB Robinson plant, which were loaded in late 1985 [Dominion 2002].

The requirements for monitoring U.S. dry storage of spent nuclear fuel (SNF) can be found in NUREG-1536 [NRC 1997]. The requirements along with the pages in the document are (paraphrased):

- 2-11: Continuous monitoring of Safety Protection Systems is defined as routine surveillance programs and/or active instrumentation. The applicability will be determined at the time of licensing.
- 4-2: Fuel and cask materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions to ensure components can maintain their safety functions.
- 7-3: describes the monitoring capability and surveillance plan for mechanical closures. For welded closures, no closure monitoring system is required, but lack of closure monitoring is usually coupled with a periodic surveillance program.
- 9-6: Usually a cask has at least one monitoring system (temperature, pressure, dosimetry). The main reason is to detect an off-normal event or accident. The licensing SAR describes the periodic visual surface and weld inspection limited to readily accessible surfaces. The SAR should describe periodic tests to verify shielding and thermal capabilities and neutron poison effectiveness. Neutron poison can have an environmental qualification of the material instead.

For the short duration, NRC puts confidence in the quality of the double closure welds and the knowledge of the behavior of the materials as a function of temperature. In the concrete NUHOMS-type system, the specifications require that the temperature in the system be measured the first few weeks after loading to see that it does not rise out of range. After that, temperatures are no longer measured. Passive radiation monitoring is conducted on the outside of the system. Daily visual checks are made of the inlet air vents to ensure they are not clogged (with bird nests, etc.). No monitoring is done on fuel canisters that have two welded seals. Continuous monitoring of the helium pressure in the gap between the two lids is conducted on the metal casks that have bolted closure. A pressure transducer that looks for a pressure rise, indicative of a leak, sends a signal back to a control room.

Recently, six casks at Surry were determined to have leaking outside seals by monitoring. A Castor-X cask had galvanic corrosion of the secondary metallic seal due to the presence of chloride probably from residual cleaning fluid. Five TN-32 casks experienced galvanic corrosion of the secondary metallic seal due to water leakage though the protective cover. Based on the monitoring and subsequent examination of the seals, corrective actions were taken to prevent a recurrence. The design of the protective cover was improved and aluminum was replaced by silver secondary seals on new loadings.

The monitoring of the operational casks has indicated minimal storage operational problems such as leaking casks The gas monitoring is designed to determine cask leaks and does not measure the composition or pressure of the cover gas directly. As such, it gives information about the integrity of the seals but says nothing about the condition of the fuel inside. The monitoring has not indicated anything that would preclude a long-term storage extension.

Since 1984, a monitored dry-storage demonstration program has been conducted at the Idaho National Environmental and Engineering Laboratory (INEEL). Five casks were loaded with

PWR fuel to determine the thermal characteristics of the casks. A cask containing BWR fuel was instrumented at the General Electric Morris Facility. The casks were backfilled with nitrogen or helium or a vacuum, and the temperature was measured in a variety of configurations. The fuel was characterized visually and/or by sipping to determine the integrity of the rod cladding. No breached rods were put into the casks tested. A complete description of the tests is in McKinnon and DeLoach [1993].

The cover gas in the casks was monitored and analyzed to determine changes in atmospheric composition and for ^{85}Kr, which, in sufficient quantity can indicate a leaking rod. Krypton-85 indications were found during the three-month testing of the Cooper BWR fuel in the REA-2023 cask and in the TN-24P cask when it held unconsolidated fuel. It is thought that the leak in the TN-24P cask occurred when the cask was rotated. In neither case was the leaking fuel rod visually identified at the end of the test. The size of the ^{85}Kr indication along with the lack of visible identification indicates that the breaches were very small and would not be considered as gross damage as defined in 10 CFR 72 (1985). No leakers have been identified in the Castor-V/21 cask holding 21 PWR assemblies since September 1985, or the MC-10 cask holding 12 to 24 PWR assemblies since 1986. Other than the short time when the thermal tests were being conducted, the temperature of these casks has been unmonitored. While the tests tell little about the defect mechanisms and the potential for failures in long-term storage, it is important to remember that no gross breaches occurred, and fuel assemblies/canisters were able to be pulled out of the casks with no adverse effects, signs that longer-term storage is feasible.

4.2 EPRI/NRC/DOE Dry Cask Storage Characterization Program

Extensive testing has been conducted on as-irradiated material that spent some time in wet storage. This testing formed the basis of support for the initial 20 y storage period. In addition, some post storage examination has been conducted on CANDU fuel. This fuel was stored at a lower temperature than expected for LWR fuel, and is of sufficiently lower burnup, so the results are of limited support for extended storage of LWR fuel. The Dry Cask Storage Characterization (DCSC) Program, which has been recently completed, is the only test to-date that has examined fuel that has been in dry storage for a significant time, at conditions realistically representative of those expected for the storage of LWR fuel.

The DCSC Program final report produced by EPRI [EPRI, 2002], summarized briefly in the following subsections, is divided into two parts. Part I of this report, summarized below in Section 4.2.2.1, documents visual examination and testing conducted in 1999 and early 2000 at the Idaho National Engineering and Environmental Laboratory (INEEL) on the CASTOR V/21 pressurized water reactor (PWR) spent fuel dry storage cask stored there for over 14 years. The purpose of the examination and testing was to develop a technical basis for renewal of licenses and Certificates of Compliance for dry storage systems for spent nuclear fuel and high-level waste at independent spent fuel storage installation sites. The examination and testing was conducted to assess the condition of the cask internal and external surfaces, cask contents consisting of 21 Westinghouse PWR spent fuel assemblies from the Surry Power Station, and the concrete storage pad. The assemblies have been continuously stored in the CASTOR cask since 1985. Cask exterior surface and selected fuel assembly temperatures, and cask surface gamma and neutron dose rates were measured. Cask external/internal surfaces, fuel basket components including accessible weldments, fuel assembly exteriors, and primary lid seals were visually examined. Cask interior crud samples and helium cover gas samples were collected and analyzed. The results of the examination and testing indicate the concrete storage pad, CASTOR V/21 cask, and cask contents exhibited sound structural and seal integrity and that long-term storage has not caused detectable degradation of the spent fuel cladding or the release of gaseous fission products between 1985 and 1999. Selected fuel rods were removed from one fuel assembly, visually examined, and then shipped to Argonne National Laboratory (ANL) for further examination.

Part II of this report, briefly described below in Section 4.2.2.2, includes a summary of the work in progress at ANL on the fuel rods shipped from INEEL. Examinations of the fuel rods included profilometry of the rods to determine if any cladding creep has occurred, and internal pressure and fission gas analysis to determine the fraction of fission gas released to the plenum inside the cladding. Additional examinations at ANL included: metallography to examine the general condition of the cladding including the thickness of the interior and exterior oxide layers; concentration, orientation, and distribution of the hydrides within the cladding to determine if significant embrittlement may have occurred due to the presence of hydrides; and cladding fracture toughness measurements also to look for signs of embrittlement. Cladding creep and tensile stress experiments at ANL investigated how much the cladding will creep prior to rupture at various temperatures and tensile stresses. Results of the work conducted at ANL suggest: little to no creep of the cladding has occurred; the amount of fission gas released from the fuel pellets into the cladding plenum compares well with other fuel and will not appreciably increase the pressure inside the rods; there is no evidence of hydrogen pickup or hydride reorientation during the storage period; and little, if any cladding annealing occurred during storage.

4.2.1 Description of the DCSC Program Test

In the mid-1980s, the U.S. Department of Energy (DOE) procured a Castor V/21 dry-storage cask for testing at the Idaho National Environmental and Engineering Laboratory (INEEL). The cask was loaded with as-irradiated assemblies (burnups in the 35 GWd/MTU range) from the Surry Nuclear Station and then tested in a series of configurations using a variety of cover gases (He, vacuum, N_2). During the tests, the temperature (350 – 415°C) within a number of the fuel assemblies was monitored, and the cover gas was periodically analyzed to determine if any rod leaks had developed. No leaks were found. The details of these tests have been reported in several documents [EPRI 1986, McKinnon 1993, McKinnon 1997]. Subsequently, the cask sat on the storage pad at the INEEL for ≈15 years with the fuel in an essentially inert atmosphere (He/<1% air). During this time the temperature dropped from about 344°C to at or somewhat above 155°C [EPRI-2002][8].

The NRC, EPRI, and DOE were interested in obtaining information relevant to extended storage by examination and testing of the cask components and fuel rods that had undergone prototypical long-term storage. These organizations jointly funded a project for examining the concrete pad, opening the Castor cask, and visually inspecting the cask internals, seals, and fuel.

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[8] The temperature measurement of 155°C was made with the cask lid off. Estimates of the temperature using an exponential time dependence of the initial temperature or decay half life are somewhat higher [EPRI 2002]

Cover gas samples taken from the cask over the 15-year period were evaluated, and radiation measurements were taken at the outside surface of the cask. After opening the cask, the seals, basket welds and fuel rod surfaces were visually examined. Crud samples from the rod surface were analyzed.

Nondestructive and destructive examinations of the fuel were made in an effort to determine if any degradation had occurred during the storage period. After visual examination, profilemetry was conducted on 12 rods. Four of these were punctured for gas analysis and two were metallographically examined. In addition, post-storage creep tests were performed. Details of the tests and examinations can be found in the final EPRI DCSC Project report [EPRI 2002].

4.2.2 Examination Findings

4.2.2.1 Cask/Pad Components Behavior

The examinations of the cask pad and components reported in EPRI [2002] included:

- Radiation survey of the cask surface and at 1 and 2 m (3 and 6 ft) with comparison to the 1985 radiation survey.
- Gas analysis of the internal atmosphere to check for the presence of air ingress into the container past the lid seal and for the presence of fission gases that would suggest cladding failure.
- Integrity of the concrete pad upon which the CASTOR V/21 has rested since 1985.
- Integrity of the inner and outer O-rings on the lid for signs of corrosion or wear.
- Assembly lifting force measurements to see if there is any resistance to pulling the assemblies back out of the cask basket channels due to corrosion or excessive rod bowing.
- Visual inspection of the outside of each assembly for indications of additional corrosion, crud spallation or other damage.
- Visual inspection of the cask basket welds and internals for indications of additional corrosion or degradation.

There was no evidence of degradation of the CASTOR V/21 cask systems important to safety from the time of initial loading of the cask in 1985 up to the time of testing in 1999. Supporting evidence for this lack of significant degradation are summarized as follows.

The 1999 and 2001radiation survey suggests that doses are generally lower than in 1985—as would be expected. Doses are now well below the 200 mrem/hr contact limit. Due to potential errors or uncertainties in some of the 1985 and 1999 and 2001 measurements, trends in doses between 1985 and 1999 are not always easy to determine, however. Nevertheless, the dose trend is down suggesting that radiation shielding materials in the cask—both gamma and neutron—are maintaining their function.

Gas analyses show neither signs of air ingress into the container nor signs of cladding failure leading to fission product release.

Visual examination of the cask lid O-rings suggest they were, indeed, in adequate condition to maintain a seal.

There was no evidence of major crud spallation from the fuel rod surfaces.

The concrete pad did not show any sign of failure. Strength measurements did not show any evidence of strength loss. Only small cracks typical of a normal, small amount of shrinkage were noticed. Concrete conditions immediately below the cask were similar to that of other areas.

All materials inside the cask—including the assemblies themselves—appeared the same as they did in 1985.

The stitch welds in the triangular air channels appear to be nonstructural welds and were only intended to add additional stability for holding the fuel basket in place during 1985 fuel loading and vertical and horizontal thermal testing. The 1985 cask heat transfer performance testing thermal analysis [EPRI, 1986], concluded that the tightly fitted fuel basket expanded and came in contact with the basket barrel and the fuel barrel came in contact with the inner wall of the cask. The results of the thermal analysis indicated that thermal stresses in the weld joints at the eight crack locations identified in 1985 were well above yield stresses measured during tensile testing of applicable weld samples. Based on these results and speculations, it appears that all new cracks identified in 1999 occurred during the 1985 thermal testing. Further, it is concluded the cracks are not relevant to normal long-term storage and presents no adverse safety implications on the cask or components to perform their safety related function.

4.2.2.2 Fuel Rod Behavior

4.2.2.2.1 Fuel Pellet Cracking, Fission Gas Release and Cladding Stress

Changes in the condition of the fuel pellets were not expected because (i) there was no cladding breach, and (ii) the temperature of the fuel during storage was lower than the temperature of the fuel during reactor operation. No additional pellet cracking was found.

Cladding creep during storage is driven by the stress induced by the rod internal gas pressure. This pressure is due primarily to the initial He fill gas and, to a lesser extent, the fission gas released from the fuel into the rod void volume. The measured fission gas release in the Surry rods is within the range one would expect for the in-reactor release of fission gas. In all likelihood, most, if not all, of the release occurred in-reactor. Within experimental uncertainty, there appears to be no additional fission gas release during the storage period. Any further release that may have occurred during further extended storage would be substantially lower due to the lower temperature and would have an insignificant effect on internal pressure and cladding stress. Thus, the stress on the cladding would decrease with time as the internal pressure decreases with temperature.

4.2.2.2.2 Cladding Creep

No pretest profilometry on the actual rods was available. Comparison of post-storage profilometry with sibling rods resulted in too much systematic error to yield meaningful creep information. Creep was estimated by the use of benchmarked creep codes for the conditions of storage and comparison of post-storage profilometry with maximum possible cladding creep down during irradiation. The creep during the 15 years storage was <0.1%, well below the original regulatory limit of 1%. This limit has since been removed by ISG-11 Rev 2 [USNRC 2002]. If any thermal creep occurred, it was probably during either the performance testing phase or the initial storage years when the temperature was the hottest and the stress was highest.

In addition, creep tests under accelerated conditions on segments of the post-storage cladding indicated that significant creep life remained. The Surry cladding has residual creep strain >1% for thermal creep test temperatures of 380°C (220 MPa) and 400°C (190 MPa). Since the current storage temperature in the Castor cask is substantially lower then the initial storage temperature and will continue to drop during extended storage, the creep testing indicates that the fuel could be stored considerably longer before creep became life limiting for cladding breach.

4.2.2.2.3 Cladding Hydride Redistribution and Reorientation

As the cladding is heated during the drying of the cask and initial storage at higher temperatures, much of the hydrogen introduced into the cladding during irradiation goes into solid solution in the Zircaloy matrix. As the cladding cools during storage, hydrogen in solid solution will precipitate as hydrides as the solubility limit is exceeded. Depending on the stress levels in the cladding, texture and cooling rates, these hydrides may be circumferential or radial in orientation. Excessive hydrogen content or radially-oriented hydrides may degrade the mechanical properties of the cladding.

There is no evidence of hydrogen pickup during the storage period. The limited hydrogen-content data suggest that axial migration of hydrogen from above the fuel mid-plane to the cooler upper end of the rod may have occurred. This migration will decrease with time as the temperature gradient becomes smaller. The effects of such migration on the long-term stability of the rods in dry storage are as yet undetermined. It may be significant in higher burnup rods using Zircaloy cladding and having higher post-irradiation hydrogen content, especially of the migration continues into the end cap weld region, but probably is not of concern for these medium burnup rods with much lower hydrogen content.

There is no evidence of radial reprecipitation of the hydrides when the fuel cooled. The stress required to reorient the hydrides is not well defined, and the stress in the Surry rods may have been below the reorientation stress threshold. The stress required for reorientation appears to increase with decreasing temperature, and the stress in the fuel rods decreases with decreasing temperature. As cooling from 355°C to >155°C precipitated hydrides that are circumferential, any additional hydride formation during further cooling should also be circumferential.

4.2.2.2.4 Cladding Annealing

Less then 20% cladding annealing occurred during the pre-storage performance period or the extended storage period. Any recovery that may have occurred was probably during the performance testing when the temperature reached as high as 415°C for 72 h, and not during the extended storage. At the even lower temperatures expected for continued storage (<150°C), no additional recovery is expected. While this annealing will decrease the yield strength hence resulting in more creep, as seen from the creep measurements this could only be minor.
The annealing will be beneficial as it also reduces the hardness, thus making the cladding less vulnerable to fracture during an accident event. The annealing is small though and not worth considering in any consequence analysis.

5 ISSUES ADDRESSED BY EXISTING TESTING AND MONITORING

The requirements that the spent fuel dry storage system must meet are the same for the initial storage period and extended storage. The materials issues that must be addressed to assure the system works have been identified. Some of these issues can be resolved for extended storage by analysis since the conditions during extended storage or less demanding (lower temperature and field). Others are resolved by testing and monitoring programs. The current monitoring programs of gas, temperature and radiation fields shed light on the ability of the cask system to perform as desired but say nothing about the fuel itself.

5.1 Summary of Potential Degradation Issues during Extended Dry Storage

Table 5-1 shows the potential ageing problems that need to be addressed for extended storage. Most of these issues have been addressed in the safety analysis reports accompanying the original ISFSI license applications. Other issues are addressed by on-going or completed research programs, such as the EPRI/NRC/DOE Dry Cask Storage Characterization Program [EPRI, 2002]. Some issues are being addressed by the monitoring programs used at dry storage facilities. These programs are very effective determining that the seals, heat rejection and radiation shielding systems continue to work. Some of the potential ageing issues are moot since the temperature and radiation field have dropped so much during the initial storage phase that the mechanism is no longer operative. In these cases, the initial license application should suffice to address these concerns.

The third column in Table 5-1 indicates if there have been tests to at least partially address the potential mechanism. If so, these tests can be used to assess the importance of these degradation mechanisms during extended storage. The fourth column indicates whether the mechanism was addressed in the original license application. If "yes", then two approaches may be used to evaluate the effects of the degradation mechanism during extended storage:

- Provide evidence to suggest that the mechanism becomes inactive during extended storage. For example, decreasing temperatures may remove the necessary driving energy for degradation. Or reaction of residual oxygen or water inside the sealed canister during the initial storage period could eliminate the possibility of further internal corrosion.
- Similar analyses would be required if it appears the mechanism would continue into the extended storage period. For example, the extent of corrosion, if significant, should be re-evaluated. Metal fatigue analyses, as was done by Dominion [2002] to deal with additional daily temperature fluctuations is another example. Although decreasing, additional radiation damage may also occur. However, in the cases discussed above, additional radiation fluence levels during extended storage were not considered high enough to cause degradation.

Table 5-1
Ageing Issue Resolution for Extended Storage Normal Burnup Fuel (BU <45 GWd/MTU)

Potential Ageing Issue for Extended Dry Storage	Data Type	Addressed with Existing Test Data?[a]	Addressed in Initial Application?[b]	Addressed with Monitoring?[c]
Normal Operation/Fuel				
Increased rod stress	Fission gas release	DCSC [EPRI, 2002]	Yes	No
Normal Operation/Cladding				
Breach	Creep	G,C, DCSC	Yes	No
	H_2 reorientation	DCSC	Yes	No
	H_2 embrittlement		No	No
Change in mechanical properties	Annealing	DCSC		No
	H_2 migration		No	No
Crud contamination	Spallation	DCSC	Yes	No
Off-Normal Operation/ Fuel				
Particle size reduction	Oxidation	R	Yes	No
Gas-release	Oxidation	R	Yes	No
Increased cover gas conductivity	Oxidation	R	Yes	No
Off-Normal/Cladding				
Wall thinning (oxidation)	Oxidation	R	Yes	No
Excessive creep	Creep Eqn.	R	Yes	No
Mechanical property change	Annealing	OG		No
Accident Operation/Fuel				
Fuel fracture	Impact fracture	NRC	Yes	No
Gas release	Oxidation	NRC	Yes	No
Particulate formation	Oxidation	R	Yes	No
Accident Operation/Cladding				
Breach	Cladding toughness	NRC		
Oxidation	Oxidation	R	Yes	
Crud contamination (spallation)	Spallation		Yes	
Breach (stress rupture)	Stress rupture limits			
Change in mechanical properties	Annealing			
	H_2 reorientation			

Table 5-1
Ageing Issue Resolution for Extended Storage Normal Burnup Fuel (BU <45 GWd/MTU) (Continued)

Potential Ageing Issue for Extended Dry Storage	Data Type	Addressed with Existing Test Data?[a]	Addressed in Initial Application?[b]	Addressed with Monitoring?[c]
Normal Operations/Cask				
Concrete degradation	Spalling/cracking due to freeze/thaw, chemicals, rebar corrosion	DCSC (pad only)	Yes	Yes
	Dehydration		Yes	No
Cask internals degradation	Corrosion	DCSC	Yes	No
	Fatigue	DCSC	Yes	No
	Radiation embrittlement	DCSC	Yes	No
	Loss of neutron poison		Yes	No
	Null ductility transition		Yes	No
Shielding	Metal corrosion	DCSC		Yes
	Neutron shielding thermal degradation	DCSC (polyethylene)	Yes	Yes
External corrosion	Cask body corrosion	DCSC	Yes	Yes
	Trunnion/cooling fin corrosion	DCSC	Yes	Yes
Radiation-induced metal hardening	Ductility		Yes	No
Bolted lid seals	Loss of seal	DCSC	Yes	Yes
Off-Normal Conditions/Cask Behavior				
Excessive temperature	Temperature		Yes	Yes
Air ingress	Oxidation	DCSC[d]	Yes (bolted lid designs)	Yes (bolted lid designs)
Accident Conditions/Cask Behavior				
(no new accident scenarios for extended storage)				

[a] programs addressing or have addressed issues
DCSC = Dry Cask Storage Characterization Program, OG =on-going NRC program at ANL
R = other research programs, NRC = NRC accident program at Sandia
G = German programs
C = Canadian Programs
[b] the initial application addresses these issues adequately for extended storage, only need to indicate that storage conditions are less severe
[c] monitoring during initial 20 y addresses these issues for extended storage
[d] air inside cask for approximately six months resulted in no visual indications of cask or fuel corrosion

The fifth column indicates whether existing monitoring programs would be able to detect potential degradation. For example, maintaining radiation monitors at the ISFSI will help assess if there has been significant degradation of shielding or radioactive release from the canisters. Visual observations will help ensure that air vents are clear, thereby assuring temperature limits are met. Visual observations will also detect signs of metal corrosion or concrete degradation. Also, for bolted lid systems employing double O-ring seals, continuing pressure monitoring in the annulus between the seals will assure the seal is maintained.

Only in the cases where there is a mechanism that only occurs during extended storage that was not addressed in the initial license application, would there need to be new analyses. Table 5-1 indicates only one such case – potential embrittlement of cladding due to hydrides. However, this mechanism has been addressed recently in their Interim Staff Guidance 11, Revision 2 [NRC, 2002]. For spent fuels with low to moderate burnup this mechanism will not be important. For accident conditions, cladding brittleness increases as temperatures decrease. This is one are of ongoing research.

All license applications address the accident situation of a projectile striking the cask. All tests to date, both foreign and domestic, along with all analysis have indicated that this is a non-event for dry storage casks. Public concern though has driven the NRC to institute programs to evaluate the potential and consequences of these events in much more detail. As a result issues such as fuel fracture, cladding breach of potentially embrittled Zircaloy are included as issues for both initial storage and extended storage.

Monitoring of some ISFSI SSCs has not been required in the original licenses. As mentioned earlier, confidence is high in welded canister systems such that no monitoring for leakage from these systems is required. For example, NRC is sufficiently confident that corrosion of the inner metal container of welded container designs will not occur that it does not require any monitoring of the container:

> [T]he NRC does not consider continuous monitoring for the Standardized NUHOMS double-weld seals to be necessary because:
>
> 1. There are no known long-term degradation mechanisms which would cause the seal to fail within the design life of the [canister containing the spent fuel]; and
>
> 2. The possibility of corrosion has been included in the design.
>
> These conditions ensure that the internal helium atmosphere will remain stable. Therefore, an individual continuous monitoring device for each HSM [horizontal storage module] is not necessary. However, the NRC considers that other forms of monitoring, including periodic surveillance, inspections and survey requirements, and survey requirements, and application of preexisting radiological environmental monitoring programs of 10 CFR Part 50 during the use of the canisters with seal weld closures can adequately satisfy NRC requirements. [Federal Register, 1994]

For the case of concrete degradation, most concrete degradation is likely to be observable from the outside of the concrete systems. However, any thermal damage would likely occur on the inside as temperatures are highest there. Design approaches are used to avoid thermal damage to avoid the need for monitoring internal concrete surfaces. For example, for the NUHOMS design, TN [2002] reports that assumed concrete structural properties for temperatures at or somewhat above even accident conditions were assumed to exist at all times:

> In addition, the maximum HSM [horizontal storage module] concrete temperature of 222°F for the 125°F short tem extreme ambient temperature case are well within the [relevant American Concrete Institute code] short-term temperature of 350°F. The same can be said for the worst case accident condition with the HSM vents assumed to be blocked with an extreme ambient temperature of 125°F for which the maximum HSM concrete temperatures are less than 350°F, except at localized areas near the center of the roof, wall, and floor slab. At these locations the maximum concrete temperatures are 441°F, 414°F, and 479°F, respectively, which are well below the [ACI code] short-term temperature limit of 650°F. It is noted that average temperatures through the thickness of the roof and floor slabs are less than 350°F for this worst case. … Nevertheless, the concrete compressive strength and rebar yield stress … at 500°F were used in calculating the concrete section capacities … to provide a conservative result. The conservatism on the concrete properties and temperatures … eliminates the need for surveillance inspection of the interior concrete surfaces following [initial] emplacement.

Thus, it seems unnecessary to monitor concrete internal surfaces during extended storage, as well, since temperatures will be lower and concrete strength potentially improved (as discussed in Section 3.2.1.3).

The Dry Cask Storage Characterization Program data [EPRI, 2002] described briefly in Section 4.2, provided information on the internal state of a CASTOR (metal) cask along with the spent fuel stored in the cask. The spent fuel had been stored in the cask for 14 years. Since inner canister internals and the spent fuel itself are not part of monitoring programs for ISFSIs, the fact that the results of the DCSC program showed essentially no evidence of cask internals or spent fuel degradation during 14 years of normal (and somewhat off-normal[9]) storage provided confidence that these parts of the system are functioning as expected. It also provided confidence that extended storage presents no new safety concerns.

5.2 Conclusion

This report provided an overview of potential ISFSI degradation issues important to extended storage. A review was made of approaches to addressing these issues for the purpose of ISFSI license extension, including the first license extension application to be submitted to the NRC [Dominion, 2002]. A summary of currently available data related to extended storage and current world-wide monitoring programs was also provided. It appears that essentially all potential degradation issues that may occur during extended storage have either been addressed by these data, by analyses provided to support the initial license applications, and/or by continuation of existing monitoring programs.

[9] Air had entered the cask internals for approximately six months prior to being removed and replaced with helium.

The only case identified that is not covered by any of the above approaches is potentially that of increased cladding brittleness due to decreasing storage temperatures. This is not expected to be of concern for either normal or off-normal operations. Under certain severe accident conditions, more brittle cladding may lead to increased fuel failures. This is currently being actively investigated by EPRI. NRC's ISG-11, Rev. 2 also helps to partially address this issue. However, analyses provided in all initial license applications for cask drop scenarios show that the canister will not fail. Thus, if extended storage was followed by a cask drop accident, the primary barrier to radionuclide release, the cask (in metal systems) or the inner canister (in concrete systems) will prevent radionuclide release whether or not some of the cladding fails. From a safety standpoint, therefore, it appears that this is also not a significant issue.

The review conducted in this report finds no technical reason why extended storage (beyond 20 years) would lead to loss of any of the required safety functions. The technical bases for extended storage are in place. Finally, the Dominion [2002] license extension application provides a procedure for addressing potential degradation mechanisms during extended storage.

6
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