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March 24, 2021
E-58432

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: Supplemental Response to Request for Additional Information for the TN Americas LLC Application for Renewal of TN-68 Dry Storage Cask, Certificate of Compliance No. 1027 (Docket No. 72-1027, CAC No. 001028, EPID: L-2020-RNW-0014)

Reference: [1] Letter E-58026, from Prakash Narayanan (TN Americas LLC) to Christian Jacobs (NRC), Response to Request for Additional Information for the TN Americas LLC Application for Renewal of TN-68 Dry Storage Cask, Certificate of Compliance No. 1027 (Docket No. 72-1027, CAC No. 001028, EPID: L-2020-RNW-0014), dated February 9, 2021

[2] Conversation Record - Clarification Call between NRC and TN Regarding CoC 1021 Renewal RAI-5 held March 3, 2021, ADAMS ML21064A050

TN Americas LLC (TN) hereby submits a supplement to Reference [1] above, and provides a revised response to a specific Request for Additional Information (RAI). As a follow-up to Reference [1], the NRC and TN held a conference call on March 3, 2021 for the purpose of clarifying TN's response to RAI-5 as documented in Reference [2]. The referenced clarification call correlates to RAI-4 for the CoC 1027 renewal application.

Enclosure 2, herein, provides the revised response to RAI-4. The RAI response has an impact section that describes any changes made to the enclosed Application for Renewal of Certificate of Compliance (CoC) No. 1027 for the TN-68 Dry Storage Cask as a result of the revised RAI response.

Enclosure 3 provides a proprietary version of the entire CoC renewal application, updated to Revision 2. Enclosure 4 provides a public version of the entire CoC renewal application, updated to Revision 2. Changes in the Revision 2 CoC renewal application are tracked and indicated by revision bars. Changes based on the RAI response are annotated with the RAI number.

Portions of this submittal include proprietary information which may not be used for any purpose other than to support NRC staff review of the application. In accordance with 10 CFR 2.390, I am providing an affidavit (Enclosure 1) specifically requesting that you withhold this proprietary information from public disclosure.

Should you have any questions regarding this submittal, please do not hesitate to contact Mr. Douglas Yates at 434-832-3101, or me at 410-910-6859.

A. Prakash

Prakash Narayanan
Chief Technical Officer

cc: Christian Jacobs, NRC DFM

Enclosures:

1. Affidavit Pursuant to 10 CFR 2.390
2. RAI-4 and Revised Response
3. Application for Renewal of Certificate of Compliance No. 1027 for the TN-68 Dry Storage Cask, (Docket No. 72-1027), Revision 2 (Proprietary Version)
4. Application for Renewal of Certificate of Compliance No. 1027 for the TN-68 Dry Storage Cask, (Docket No. 72-1027), Revision 2 (Public Version)

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

TN Americas LLC)
State of Maryland) SS.
County of ~~Howard~~ BALTIMORE)

I, Prakash Narayanan, depose and say that I am the Chief Technical Officer of TN Americas LLC, duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in Enclosure 3, listed below:

- Enclosure 3, Portions of the CoC 1027 Renewal Application, Revision 2

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by TN Americas LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure involves operating experience with spent fuel storage system aging and details of spent fuel storage system aging management studies, associated with the renewal application for the TN-68 Dry Storage Cask, which are owned and have been held in confidence by TN Americas LLC.

2) The information is of a type customarily held in confidence by TN Americas LLC, and not customarily disclosed to the public. TN Americas LLC has a rational basis for determining the types of information customarily held in confidence by it.

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of TN Americas LLC, because the information consists of operating experience with the aging degradation of TN-68 loaded casks, and details of spent fuel storage system aging management studies and planned aging management strategies associated with the renewal application for the TN-68 Dry Storage Cask, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with TN Americas LLC, take marketing or other actions to improve their product's position or impair the position of TN Americas LLC's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

A. Prakash

Prakash Narayanan
Chief Technical Officer, TN Americas LLC

Subscribed and sworn before me this 12th day of March, 2021.

Notary Public

My Commission Expires 7 / 16 / 2022

MARK JOSEPH KRIEGER, JR. NOTARY PUBLIC BALTIMORE CO. MD

MARK JOSEPH KRIEGER, JR.
Notary Public, State of Maryland
County of Baltimore
My Commission Expires July 16, 2022

RAI 4:

Provide the following additional information on Radiation Monitoring for the TN-68 Cask Aging Management Plan (AMP) and update the AMP as necessary. Because the shielding material is made of borated polyester and polypropylene, there is the possibility that heat from spent fuel coupled with age may degrade this material and reduce its shielding effect. The applicant proposes to use radiation monitoring at the perimeter fence to ensure that the shielding material maintains its' integrity.

Please provide:

1. Parameters Monitored or Inspected: A description of how the parameters monitored will be capable of identifying degradation or potential degradation before a loss of intended function for a specific cask considering that the number of TN-68 casks which may be loaded and placed on the independent spent fuel storage installation (ISFSI) storage pad can vary with time.

2. Detection of Aging Effects: The technical basis to show that detection of gamma and neutron radiation by the use of thermoluminescent dosimeters (TLDs) at the ISFSI perimeter fence has sufficient resolution to identify degradation of the shielding effectiveness for the individual casks. Section 4.3.4.2 of the renewal application states, "Detection of gamma and neutron radiation is accomplished by the placement of the TLDs at the ISFSI perimeter fence, or between the ISFSI and locations used to show compliance with 10CFR20.1301 and 10CFR72.104. While the TLDs may not be capable of detecting all neutrons (i.e., the very high energy neutrons) they are effective in detecting adverse trends in neutron dose rates."

3. Detection of Aging Effects: Identify the personnel qualifications for the staff that evaluate the radiation monitoring results.

4. Acceptance Criteria: The acceptance criteria (e.g., level of increase) that will be used to determine the annual upward increasing trend in neutron or gamma quarterly TLD readings at the ISFSI perimeter fence that indicates a loss of intended function of the shielding, including a description of how the criteria account for decay of the spent fuel source term and consideration for the use of the TN-68 system at an operating reactor or ISFSI where the number of storage casks is increasing with time.

5. Operating Experience: A summary of the available operating experience for similar cask designs such as those used at the North Anna ISFSI and the Prairie Island ISFSI. Include a description of how the TLD data trends at the Prairie Island ISFSI with the cask survey results described in the Prairie Island ISFSI AMP (ML15285A007) will be considered in the Operating Experience AMP element.

This information is needed to determine compliance with 10 CFR 72.240(c).

Response to RAI 4:

Response to RAI 4(1) Parameters Monitored or Inspected:

The purpose of an AMP is to manage aging effects prior to a loss of intended safety function. Therefore, when selecting the parameters to be monitored or inspected, for managing the aging effects of the borated polyester and polypropylene, it is necessary to understand what would constitute a loss of the shielding intended safety function. As stated in Section 4.3.3, the shielding intended safety function is defined as meeting the requirements of 10 CFR Part 20 and 10 CFR Part 72.104, i.e., doses to *individuals* outside the ISFSI, and are not tied to a specific cask. As long as the dose to *individuals outside the ISFSI* is maintained within these limits, the shielding intended safety function is not lost even if the polymer material of an individual cask degrades. Interim Staff Guidance (ISG-13) states “Compliance with the dose limits in 72.104 will be verified by an environmental monitoring program using direct radiation measurements (such as TLDs) and/or effluent measurements, as appropriate.” Therefore, it is more appropriate to monitor parameters that are related to the total dose from the ISFSI to *individuals outside the ISFSI*, e.g., TLD at the ISFSI fence, rather than the dose rates from an individual cask.

While not part of the formal aging management program, radiation surveys conducted as part of the ALARA programs for developing radiation work permits provide assurance that dose to radiation workers is kept as low as reasonably achievable. If anything unexpected is noted during these surveys, the standard practice would be to enter the condition into the corrective action process for evaluation.

Response to RAI 4(2) Detection of Aging Effects:

As stated in the response to RAI 4(1), the shielding intended safety function of the TN-68 cask is defined as meeting the requirements of 10 CFR Part 20 and 10 CFR 72.104, i.e., doses to *individuals* outside the independent spent fuel storage installation (ISFSI) and not tied to a specific cask. Therefore, it is not necessary for the aging management program (AMP) to identify degradation of the shielding for individual casks.

To provide additional assurance that the TLDs will be capable of detecting an increasing trend in the neutron dose rate, Section 4.3.4.2 of the TN-68 AMP *has* been revised to require that the TLDs be capable of detecting low, intermediate, and high energy neutrons (e.g., using a CR 39 polycarbonate chip or equivalent dosimetry). In addition, the TN-68 AMP has been revised to allow the licensee to determine the placement of the TLDs on the ISFSI parameter fence taking the following into consideration:

- The objective is to monitor for increasing dose rates that could approach the 10 CFR 20.1301 and 10 CFR 72.104 regulatory limits*, and dose to individuals outside the ISFSI*.
- Casks that have been in service the longest would generally be expected to be more susceptible to potential degradation.
- Shadowing or shielding by other casks or structures.

To provide additional monitoring for degradation of the radial neutron shield, Sections 4.3.4.2, 4.3.5, and 4.3.6.2 of the TN-68 AMP have been revised to include annual neutron radiation surveys around the perimeter of the storage pad(s).

Response to RAI 4(3) Detection of Aging Effects:

The personnel qualification for collection, reading, and trending of the TLDs will be specified by the licensee in accordance with their administrative controls as described in Section 4.3.9.

Response to RAI 4(4) Acceptance Criteria:

Quantifying an acceptable level of increase in the annual upward increasing trend could lead to situations where there is an obvious increasing trend but the condition does not meet a numerical increase criterion and, thus, not entered into the licensee's corrective action program. By specifying an increasing trend as the acceptance criteria, once a trend is identified (regardless of the magnitude of the trend) it would be entered into the licensee's corrective action program.

As the spent fuel source term decays, it would be expected that the dose rates at the ISFSI fence would decrease with time (assuming no other changes such as adding casks or degradation of the shielding material). Obviously, if there is a decrease in the dose rate at the independent spent fuel storage isolation (ISFSI) fence over time, the requirements of 10 CFR Part 20 and 10 CFR Part 72.104 *for individuals outside the ISFSI* will continue to be met and there would be no loss of the shielding intended safety function.

Although an increasing trend would not be expected, this condition should be entered into the licensee's corrective action program for evaluation. The evaluation in the corrective action program would determine if the increasing trend is due to degradation of the shielding material or due to some other cause, e.g., storing more spent fuel in the ISFSI.

Response to RAI 4(5) Operating Experience:

TN Americas does not have access to the TLDs readings from the Prairie Island or the North Anna ISFSI and therefore cannot compare TLD data trends to individual cask survey results. In any event, the comparison would not be meaningful since the addition of casks to the ISFSI could have more of an effect on the TLD readings than on the individual cask surveys depending on the location of the additional cask. In addition, the radiation surveys shown in Figure A2.10-1 and Figure A2.10-2 of the Prairie Island ISFSI Site-Specific license renewal [1] illustrate a large scatter in individual cask radiation surveys. This scatter can be assumed to be attributed to the non-standard techniques used by inspectors for individual cask readings and makes it very difficult to get a meaningful indication of degradation that could lead to a loss of the shielding intended safety function. However, the TLDs at the ISFSI perimeter fence or at locations used to show compliance with the regulations provide for consistent, uniform and predictable readings that provide a good indication of the casks shielding performance. Therefore, the information in the Prairie Island ISFSI Site-Specific license renewal supports reliance upon utilizing the TLDs at the ISFSI fence to monitor the shielding intended safety function.

References:

1. Letter L-PI-15-082, from Scott Sharp (NSPM), to Document Control Desk (NRC), "Supplement to Prairie Island Independent Spent Fuel Storage Installation License Renewal Application – Revised Aging Management Plan (TAC No. L24592)," October 12, 2015, (ADAMS Accession Number ML15285A007).

Impact:

CoC Renewal Application Sections 4.3.4.2*, 4.3.5,* and 4.3.6.2 have been revised as described in the response.

Enclosure 3 to E-58432

Application for Renewal of Certificate of Compliance No. 1027 for the TN-68 Dry Storage Cask, (Docket No. 72-1027), Revision 2
(Proprietary Version)

Withheld Pursuant to 10 CFR 2.390

Enclosure 4 to E-58432

Application for Renewal of Certificate of Compliance No. 1027 for the TN-68 Dry Storage Cask, (Docket No. 72-1027), Revision 2 (Public Version)

PUBLIC

Certificate of Compliance Renewal Application for the TN-68 Dry Storage Cask

Certificate of Compliance No. 1027

(Docket No. 72-1027)

Prepared by:

TN Americas LLC

Columbia, Maryland

Revision *2*

March 2021

Table of Contents

Acronyms and Abbreviations

ACI	American Concrete Institute
AMA	Aging Management Activity
AMID	Aging Management INPO Database
AMP	Aging Management Program
AMR	Aging Management Review
ANSI/ASCE	American National Standards Institute / American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASR	Alkali-Silica Reaction
B&PV	Boiler and Pressure Vessel
BWR	Boiling Water Reactor
CAR	Corrective Action Report
CFR	Code of Federal Regulations
CISCC	Chloride Induced Stress Corrosion Cracking
CO	Confinement Intended Function
CoC	Certificate of Compliance
CR	Sub-Criticality Control Intended Function
DEF	Delayed Ettringite Formation
DHC	Delayed Hydride Cracking
DOE	Department of Energy
DSC	Dry Shielded Canister
DSS	Dry Storage System
EPRI	Electric Power Research Institute
GALL	Generic Aging Lessons Learned
HAZ	Heat Affected Zone
HBU	High Burnup
HDRP	High Burnup Fuel Dry Storage Cask Research and Development Project
INPO	Institute of Nuclear Power Operations
ISFSI	Independent Spent Fuel Storage Installation
ISG	Interim Staff Guidance
ITS	Important-to-Safety
LRA	License Renewal Application
MAPS	Managing Aging Processes in Storage
MIC	Microbiologically Influenced Corrosion
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NITS	Not Important-to-Safety

OE	Operating Experience
PCMI	Pellet-to-Cladding Mechanical Interaction
PEO	Period of Extended Operation
QA	Quality Assurance
RH	Relative Humidity
RT	Retrievability Intended Function
SAR	Safety Analysis Report
SCC	Stress Corrosion Cracking
SFA	Spent Fuel Assembly
SH	Radiation Shielding Intended Function
SR	Structural Integrity Intended Function
SS	Stainless Steel
SSCs	Structures, Systems and Components
TH	Heat Removal Capability Intended Function
TLAA	Time-Limited Aging Analysis
TLD	Thermoluminescent Dosimeter
TN	TN Americas LLC
TNUG	TN User's Group
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report

CHAPTER 1
GENERAL INFORMATION

CONTENTS

LIST OF TABLES

1.1 Introduction

The TN-68 dry storage cask system, Certificate of Compliance (CoC) No. 1027, Revision 0 [1-3] was approved by the U.S. Nuclear Regulatory Commission (NRC) effective May 30, 2000, for storage of spent nuclear fuel by general licensees. The expiration date for CoC 1027 is May 28, 2020. As the certificate holder of CoC 1027, TN Americas LLC (TN)[1] is applying for renewal of CoC 1027 for a term of 40 years, in accordance with the 10 CFR 72.240(a) [1-1].

This application for CoC 1027 renewal includes the safety analysis report (SAR) information required by 10 CFR 72.240(c). The SAR content of this application is based on the guidance provided in NUREG-1927 [1-2].

In accordance with NUREG-1927 [1-2], this renewal application is based "on the continuation of the approved design basis throughout the period of extended operation."

The identification and management of potential aging degradation mechanisms for different material/environment combinations is based on the guidance of NUREG-2214 [1-7] in support of this renewal application.

[1] TN Americas LLC, formerly AREVA TN, and Transnuclear, Inc.

1.2 TN-68 Dry Storage Cask System Description

1.2.1 General System Description

The TN-68 dry storage cask system is a cask-based system for the dry storage of irradiated spent fuel assemblies (SFAs) and consists of a containment vessel, bolted lid, vent and drain port covers, a thick gamma shield, and radial neutron shielding. Additional structures, systems, and components (SSCs) include a concrete storage pad, and other fuel transfer and auxiliary equipment used to support cask loading and transfer operations.

The following paragraphs provide an overview of the TN-68 dry storage cask system. A more complete system description, including supporting design basis, is contained in the UFSAR, Revision 9 [1-6].

1.2.2 Principal Components of the TN-68 Dry Storage Cask System

1.2.2.1 TN-68 Cask

The TN-68 containment vessel is comprised of an inner shell, which is a welded steel cylinder with an integrally-welded steel bottom; a welded flange forging (with stainless steel weld overlay); a bolted steel lid; and vent and drain port covers, cover seals and cover bolts. The inner shell is surrounded by a thick gamma shield. Radial neutron shielding around the gamma shield wall is provided by borated polyester resin encased in aluminum. A steel outer shell surrounds the radial neutron shielding. Additional neutron shielding is provided by a top neutron shield, which consists of a disc of polypropylene encased in steel that is bolted to the cask lid. Trunnions are provided at the cask upper and lower ends to permit cask movement and transport. A protective weather cover fits over the top of the cask. The cover is sealed with an elastomer O-ring.

The internal basket assembly is made of stainless steel cells joined by a fusion welding process to stainless steel plates. Above and below the plates are slotted borated aluminum or boron carbide/aluminum metal matrix composite plates, which form an egg-crate structure. The basket is supported laterally by aluminum rails. The fuel basket structure is designed to hold 68 fuel assemblies.

1.2.2.2 Spent Fuel Assemblies

The TN-68 dry storage cask system is designed to store boiling water reactor (BWR) spent fuel assembly types as authorized contents per the associated technical specifications (TS) and applicable amendments.

1.2.2.3 Concrete Storage Pad

The TN-68 cask is installed on a load-bearing foundation, which consists of a reinforced concrete pad on a subgrade suitable to support the loads. There are no structural connections or means to transfer shear between the TN-68 cask and the concrete pad.

1.2.2.4 Auxiliary Equipment

Auxiliary equipment used to facilitate cask loading, draining, drying, inerting and sealing operations include, but are not limited to, special lifting devices, vertical transporter, and vacuum drying/helium leak test equipment.

1.2.2.5 Miscellaneous Equipment

Miscellaneous independent spent fuel storage installation (ISFSI) equipment (e.g., ISFSI security fences and gates, lighting, lightning protection, communications, and monitoring equipment) are not part of the CoC 1027 storage system.

1.3 Background

The TN-68 dry storage cask system, originally approved in May 2000, has evolved with time via approval of amendments to the original CoC 1027. Each of these amendments was designed to accommodate the evolving needs of the industry to store spent BWR fuel.

A listing of the currently approved CoC 1027 amendments is provided in Table 1-1. This table provides: (a) the effective date of the amendment, and (b) a brief description of the scope of each amendment.

1.4 Application Format and Content

This application includes SAR information required by 10 CFR 72.240 (c) [1-1]. The format and content of this SAR information is consistent with the guidance contained in NUREG-1927 [1-2].

Chapter 1, General Information: Chapter 1 provides (1) a general description of the TN-68 dry storage cask system, (2) a discussion of CoC 1027 amendments, and (3) information on the format and content of this application.

Chapter 2, Scoping Evaluation: Chapter 2 provides a description of the methodology used to identify the SSCs of the TN-68 dry storage cask system that are within the scope of the renewal. This methodology is based on the process described in NUREG-1927 [1-2]. Chapter 2 also provides a summary of the results of the scoping evaluation based on Revision 9 of the UFSAR [1-6].

Chapter 3, Aging Management Review: Chapter 3 provides the methodology used for the aging management review (AMR) of the TN-68 dry storage cask system, based on the guidance provided in NUREG-1927 [1-2]. The AMR documented in Chapter 3 identifies the materials and environment for those SSCs and associated subcomponents determined to be within the renewal scope in Chapter 2. This is accomplished by reviewing the drawings and the design basis included in the current UFSAR [1-6], CoC Amendment 1 [1-4], and associated TS [1-5]. Once the component material/environment combinations are determined, a review is performed to identify credible aging degradation mechanisms for the different material/environment combinations. NUREG-2214 [1-7] (based on technical literature, related research, industry information, and existing operating experience) is used as a guide for the review. After the credible aging mechanisms and effects are identified, it is determined whether the effects can be managed via a time-limited aging analysis (TLAA) or will require an aging management program (AMP).

Appendix 3A, Time-Limited Aging Analyses: Appendix 3A identifies the calculations or analyses used to demonstrate that in-scope SSCs will maintain their intended safety function throughout an explicitly stated period of operation, i.e., TLAAs. For those TLAAs that would not remain valid through the period of extended operation, Appendix 3A provides a summary of the revised or updated analysis.

Appendix 3B, Supplemental Evaluations: Appendix 3B provides a summary of supplemental evaluations and calculations performed to support the AMR and/or an element in an AMP.

Appendix 3C, Operating Experience Review: Appendix 3C provides a summary of the operating experience review performed to support the AMR and AMPs.

Chapter 4, Aging Management Programs: Chapter 4 provides the AMPs credited for managing each of the identified aging effects for the in-scope SSCs of the TN-68 dry storage cask system. The purpose of an AMP is to ensure that no aging effects result in a loss of intended safety function of the SSCs that are within the scope of renewal, for

the term of the renewal. Each of the AMPs consists of the ten elements called for in NUREG-1927 [1-2].

Attachment A: This attachment provides the recommended changes to the UFSAR for CoC 1027 renewal.

Attachment B: This attachment provides the recommended changes to the CoC and TS for CoC 1027 renewal.

1.5 References (Chapter 1, General Information)

1-1 Title 10 Code of Federal Regulations Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."

1-2 U.S. Nuclear Regulatory Commission, NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," Revision 1, June 2016.

1-3 U.S. Nuclear Regulatory Commission, "Certificate of Compliance for Spent Fuel Storage Casks," Certificate No. 1027, Revision 0, May 30, 2000, Docket No. 72-1027.

1-4 U.S. Nuclear Regulatory Commission, "Certificate of Compliance for Spent Fuel Storage Casks," Certificate No. 1027, Amendment No. 1, Effective October 30, 2007, Docket No. 72-1027.

1-5 U.S. Nuclear Regulatory Commission, CoC 1027 Appendix A, "TN-68 Generic Technical Specifications," Amendment 1, October 30, 2007, Docket No. 72-1027.

1-6 TN-68 Dry Storage Cask Updated Final Safety Analysis Report, Revision 9, May 2018.

1-7 U.S. Nuclear Regulatory Commission, NUREG-2214, "Managing Aging Processes in Storage (MAPS) Report", July 2019.

Table 1-1
Listing of CoC 1027 Amendments

Amendment No.	Amendment Effective Date	Description
0	05/30/00	Initial approval to store spent fuel in the TN-68 dry storage cask system.
1	10/30/07	Approved several changes, included the allowable fuel burnup, minimum cooling times, decay heat, and fuel enrichment. The amendment also approved damaged fuel as authorized contents of the cask and to reduce the cask spacing on the storage pad.

CHAPTER 2
SCOPING EVALUATION

CONTENTS

LIST OF TABLES

2.1 Introduction

Chapter 2 describes the evaluation process and methodology used to identify the structures, systems, and components (SSCs) of the TN-68 dry storage cask system that are within the scope of renewal.

In accordance with the guidance contained in NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel" [2-2], the first step of the renewal process is the performance of a scoping evaluation. The objective of the scoping evaluation is to identify the SSCs of the TN-68 dry storage cask system that are within the scope of renewal.

A description of the scoping process and methodology is provided in Section 2.2. The results of the scoping evaluation are provided in Section 2.3. The bases for excluding certain subcomponents are provided in Section 2.4.

2.2 Scoping Evaluation Process and Methodology

The scoping evaluation of the Certificate of Compliance (CoC) No. 1027 storage system is performed based on the process described in NUREG-1927 [2-2]. SSCs (and associated subcomponents) are considered to be within the scope of the renewal if they satisfy either of the following criteria:

Criterion 1:

The SSC (and associated subcomponents) is classified as important-to-safety (ITS) as it is relied on to perform one of the following functions (Code of Federal Regulations (CFR) 10 CFR 72.3):

a) Maintain the conditions required by the regulations or the CoC to store spent fuel safely.

b) Prevent damage to the spent fuel during handling and storage.

c) 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.

These SSCs ensure that ITS functions are met for (1) confinement, (2) radiation shielding, (3) sub-criticality control, (4) heat-removal capability, (5) structural integrity, and (6) retrievability.

Criterion 2:

The SSC (and associated subcomponents) is classified as not important-to-safety (NITS), but according to the design basis, its failure could prevent fulfillment of a function that is ITS.

The retrievability safety function is based on the ability to repackage/reconfigure a TN-68 cask into a transportation cask; i.e., Revision 0 of Interim Staff Guidance (ISG) 2 [2-7].

In accordance with NUREG-1927 [2-2], the renewal is based "on the continuation of the approved design bases throughout the period of extended operation." Accordingly, the sources of information reviewed for this scoping evaluation that describe the approved design basis and the intended safety functions of the SSCs (and associated subcomponents) are the following:

a) TN-68 Updated Final Safety Analysis Report (UFSAR) [2-1].

b) CoC 1027 Certificate and Technical Specifications (TS) for each amendment [2-3 and 2-5].

c) Safety Evaluation Reports (SERs) for each amendment [2-4 and 2-6].

These documents were reviewed to determine those SSCs (and associated subcomponents) with safety functions that meet either Scoping Criterion 1 or Criterion 2, as defined above. Based on this review, those subcomponents that perform or support any of the identified intended safety functions are determined to be in-scope and require an aging management review (AMR). Those subcomponents that do not perform or support a safety function are excluded from further evaluation in the AMR.

The scope of the CoC 1027 renewal encompasses the initial approved application (Amendment 0 [2-3]) and subsequently approved Amendment 1 [2-5].

Amendment 0 of CoC 1027 [2-3] approved the design of the TN-68 cask to store up to 68 intact, unconsolidated boiling water reactor (BWR) fuel assemblies. Section 1.2 of the associated U.S. Nuclear Regulatory Commission (NRC) Safety Evaluation Report [2-4] states: “The drawings for the TN-68 associated with structures, systems, and components (SSCs) important to safety are contained in Section 1.5 of the SAR. A specific list of these components is noted on the parts list shown on drawing 972-70-2. The applicant provided sufficiently detailed drawings regarding dimension, materials, and specifications to allow for a thorough evaluation of the entire system.” These drawings and the intended safety functions of the TN-68 casks were not affected by the subsequent amendment. Therefore, the drawings in Section 1.5 (along with the design information in Sections 2 through 14) of Revision 9 of the UFSAR [2-1] represent the approved design bases for the TN-68. The drawings in Section 1.5 of the UFSAR were used as the source of the TN-68 subcomponents (along with their material specifications and safety classification) for the scoping evaluation.

Amendment 1 of CoC 1027 [2-5] approved several changes, which included the allowable fuel burnup, minimum cooling times, decay heat, and fuel enrichment. The amendment also approved damaged fuel as authorized contents of the cask and to reduce the cask spacing on the storage pad. Other than the additions of drawings to accommodate damaged fuel, these changes do not affect or change the subcomponents (including materials and safety classification) of the TN-68 casks.

In summary, the drawings and design information in UFSAR Revision 9 [2-1] represent the approved design bases of the SSCs used for the scoping evaluation.

2.3 Results of Scoping Evaluation

The results of the scoping evaluation are summarized in Table 2-1. The following subsections provide a discussion of the scoping evaluation performed for each SSC.

2.3.1 TN-68 Cask

The TN-68 cask includes, but is not limited to, the internal basket assembly, confinement vessel, gamma shielding surrounding the confinement vessel, radial neutron shielding, overpressure system, and protective cover. The TN-68 cask is classified as ITS. Therefore, it satisfies Criterion 1 and is in-scope. The scoping evaluations of the subcomponents of the TN-68 are summarized in Table 2-2, which also lists the UFSAR drawings used to identify the TN-68 subcomponents, their material of construction, and their safety classification.

2.3.2 Spent Fuel Assemblies

The subcomponents of the spent fuel assemblies have intended safety functions required to maintain the conditions required by regulations to store the spent fuel safely. Therefore, the spent fuel assemblies satisfy Criterion 1 and are in-scope. The scoping evaluations of the subcomponents of the spent fuel assemblies are summarized in Table 2-3.

2.3.3 ISFSI Storage Pad

The reinforced concrete storage pad is not classified as ITS in Section 2.3 of [2-1] and is, thus, considered a NITS structure. It is designed and constructed to plant-specific site conditions, and is not part of the CoC 1027 certification. However, a portion of the pad is included in the thermal models. Therefore, failure of the pad could affect the heat removal capability of the system, i.e., prevent fulfillment of an intended safety function. Therefore, the reinforced concrete storage pad meets Criterion 2 and is in-scope for renewal.

2.3.4 Auxiliary Equipment

Auxiliary equipment used to facilitate cask loading, draining, drying, inerting and sealing operations include, but are not limited to, special lifting devices, vertical cask transporter, and vacuum drying/helium leak test equipment.

Part 1(b) of the CoC 1027 [2-3 and 2-5] states; "The auxiliary equipment necessary for ISFSI operation is not included as part of the TN-68 cask system reviewed for a Certificate of Compliance under 10 CFR Part 72, Subpart L. Such equipment may include, but is not limited to, special lifting devices, transfer trailers or equipment, and vacuum drying/helium leak test equipment." Therefore, this equipment is not within the scope of CoC 1027 renewal.

2.3.5 Miscellaneous Equipment

Miscellaneous independent spent fuel storage installation (ISFSI) equipment (e.g., ISFSI security fences and gates, lighting, lightning protection, communications, and monitoring equipment) are not part of the CoC 1027 storage system approved in accordance with 10 CFR Part 72, Subpart L. They are not classified as ITS, nor would their failure prevent the fulfillment of a function that is ITS. Therefore, they are not within the scope of the CoC 1027 license renewal.

2.4 Subcomponents Not Within Scope of CoC 1027 License Renewal

This section provides a summary SSC subcomponents that were identified as not within the scope of renewal and the basis for exclusion.

2.4.1 Overpressure System

Pressure monitoring equipment includes pressure switches or transducers and electrical cables. If the monitoring system was not to function, no safety functions of the cask would be impaired. There would be no leakage in or out of the cask. The overpressure system (including monitoring instrumentation) is designated as NITS (Table 2.3-1 of [2-1]) since the failure of the system will not result in a release of radioactive material. The monitoring system has not been designed to prevent failure during accident loadings. If an accident were to occur, measures would be taken to replace or repair the system soon after the accident. Therefore, the subcomponents of the overpressure system are not within the scope of CoC 1027 renewal.

2.4.2 Drain Tube and Hansen Couplings

Drain tube (with all associated hardware including drain tube clamp, drain tube adapter, attachment screws, and O-ring seals) is for operational convenience only and does not perform any safety functions. The Hansen coupling on the vent and drain ports are for operational purposes only and do not perform any safety functions. Failure of these components while in storage will not affect any safety functions. Therefore, they are not within the scope of CoC 1027 renewal.

2.4.3 External Paint

While a suitable primer and white topcoat paint is applied to the exterior of the cask to protect the cask from rusting, no credit for the coating is taken in the aging management review. Therefore, the external paint is not within the scope of CoC 1027 renewal.

2.4.4 Protective Cover Bolts, Seal, Washers, and Threaded Inserts

The protective cover protects the overpressure tank, top neutron shield and lid from debris and wildlife nesting and allows rain water to drain more easily from the top of the cask. It has no structural integrity safety function. Operating experience has shown moisture can penetrate the protective cover. For the purpose of renewal, therefore, no credit is taken for the cover as a form of weather protection. The protective cover bolts, seal, flat washers, and threaded inserts are classified as NITS and since the protective cover is not being credited as a form of weather protection, failure of these subcomponents will not impact any intended safety functions. Therefore, the protective cover bolts, seal, flat washers, and threaded inserts are not within the scope of CoC 1027 renewal.

2.4.5 Top Neutron Shield Bolts and Washers

Per Section 2.3.1 of the UFSAR [2-1], the top neutron shield is used for supplemental shielding, but the accident condition dose limits are met without installing the top neutron shield. Thus, there is no structural integrity safety function for the top neutron shield, bolts or washers. Failure of the top neutron shield bolts or washers will not impact any intended safety functions. Therefore, the top neutron shield bolts and washers are not within the scope of CoC 1027 renewal.

2.4.6 Lid Alignment Pin and Shear Key

Per Section 2.3.1 of the UFSAR [2-1], the lid alignment pin and shear key are used to ease operation and no structural credit is taken for these subcomponents. Failure of these NITS items will not affect any intended safety functions of the cask. Therefore, the lid alignment pin and shear key are not within the scope of CoC 1027 renewal.

2.4.7 Security Wires and Seals

Per Section 2.3.1 of the UFSAR [2-1], the security wires and seals are used to provide evidence that the cask has not been tampered with. Therefore, failure of these NITS items will not impact any intended safety functions. Therefore, the security wires and seals are not within the scope of CoC 1027 renewal.

2.4.8 Pressure Relief Valve

A pressure relief valve is provided in the outer shell to assure any pressure buildup due to heating of the resin and entrapped air is small. Per Table 3.3-6 of the UFSAR [2-1], the primary function of the pressure relief valve is operations support. Failure of this NITS item will not impact any intended safety functions. Therefore, the pressure relief valve is not within the scope of CoC 1027 renewal.

2.4.9 Fuel Pellets

The fuel pellets are stacked within the fuel cladding and do not perform a renewal intended function. The pellets may have sufficient cracking to have degraded into numerous pieces, rather than their original cylindrical shape. The pieces of pellets will be contained within the fuel cladding. It is the fuel cladding, not the fuel pellets that ensures the fuel remains in a subcritical and coolable geometry. Therefore, the fuel pellets are not within the scope of CoC 1027 renewal.

2.5 References

2-1 TN-68 Dry Storage Cask Updated Final Safety Analysis Report, Revision 9, May 2018.

2-2 U.S. Nuclear Regulatory Commission, NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," Revision 1, June 2016.

2-3 U.S. Nuclear Regulatory Commission, "Certificate of Compliance for Spent Fuel Storage Casks, Certificate No. 1027, Amendment 0," Effective Date May 30, 2000, Docket No. 72-1027, (ADAMS Accession Number ML003714545).

2-4 U.S. Nuclear Regulatory Commission, "Safety Evaluation Report for Transnuclear, Inc. TN-68 Dry Storage Cask System, Amendment 0," (ADAMS Accession Number ML003713477).

2-5 U.S. Nuclear Regulatory Commission, "Certificate of Compliance for Spent Fuel Storage Casks, Certificate No. 1027, Amendment 1," Amendment Effective Date October 30, 2017, Docket No. 72-1027, (ADAMS Accession Number ML073050254).

2-6 U.S. Nuclear Regulatory Commission, "Safety Evaluation Report for TN-68 Dry Storage Cask, CoC 1027 Amendment 1," Docket No. 72-1027, (ADAMS Accession Number ML073050286).

2-7 U.S. Nuclear Regulatory Commission, Spent Fuel Project Office, Interim Staff Guidance 2, "Fuel Retrievability," Revision 0, October 6, 1998.

Table 2-1
Scoping Evaluation of TN-68 Dry Storage Cask System SSCs

SSC	Intended Safety Function						Criterion		In-Scope
	Confinement	Radiation Shielding	Sub-Criticality Control	Heat-Removal Capability	Structural Integrity	Retrievability	No. 1	No. 2	
TN-68 Cask	Yes	Yes	Yes	Yes	Yes	Yes	Yes	N/A	Yes
Spent Fuel Assemblies	Yes	No	Yes	Yes	Yes	No	Yes	N/A	Yes
ISFSI Concrete Storage Pad	No	No	No	Yes(1)	No	No	No	Yes	Yes
Auxiliary Equipment	No	No	No	No	No	No	No	No	No
Miscellaneous Equipment	No	No	No	No	No	No	No	No	No

1. Failure of the storage pad could prevent fulfillment of this intended safety function.

Table 2-2
Scoping Evaluation for TN-68 Cask
(3 Pages)

UFSAR Drawing No.	Rev	Component	Item No.	Subcomponent Parts		Safety Classification	Intended Safety Function						Criterion	
							Confinement	Radiation Shielding	Sub-Criticality Control	Heat-Removal Capability	Structural Integrity	Retrievability	No. 1	No. 2
972-70-2	14	TN-68	1	Gamma Shield		ITS	No	Yes	No	Yes	Yes	Yes	Yes	N/A
972-70-2	14	TN-68	2	Lid (including stainless steel weld overlay at seal location)		ITS	Yes	Yes	No	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	3	Inner Containment		ITS	Yes	Yes	No	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	4	Bottom		ITS	No	Yes	No	Yes	Yes	Yes	Yes	N/A
972-70-2	14	TN-68	5	Bottom Containment		ITS	Yes	Yes	No	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	6	Upper Trunnion		ITS	No	Yes	No	No	Yes	Yes	Yes	N/A
972-70-2	14	TN-68	7	Lower Trunnion		ITS	No	Yes	No	No	Yes	Yes	Yes	N/A
972-70-2	14	TN-68	8	Shield Plate		ITS	No	Yes	No	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	9	Radial Neutron Shield		ITS	No	Yes	No	Yes	No	No	Yes	N/A
972-70-2	14	TN-68	10	Outer Shell		ITS	No	Yes	No	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	11	Protective Cover		NITS	No	Yes(2)	No	Yes(2)	No	No	No	Yes
972-70-2	14	TN-68	12	Top Neutron Shield		NITS	No	Yes(2)	No	Yes(2)	No	No	No	Yes
972-70-2	14	TN-68	13	Radial N-Shield Box		ITS	No	Yes	No	Yes	No	No	Yes	N/A
972-70-2	14	TN-68	14	Lid Bolt		ITS	Yes	Yes	No	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	15	Protective Cover Bolt		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	16	Lid Seal		ITS	Yes	No	No	No	No	No	Yes	N/A
972-70-2	14	TN-68	17	Protective Cover Seal		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	18	Overpressure Port Cover		ITS	Yes(3)	No	No	No	No	No	Yes	N/A
972-70-2	14	TN-68	19	Overpressure Port Cover Seal		ITS	Yes(3)	No	No	No	No	No	Yes	N/A
972-70-2	14	TN-68	20	Top Neutron Shield Bolt		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	21	Pressure Monitoring System		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	22	Drain Port Cover		ITS	Yes	Yes	No	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	23	Vent Port Cover		ITS	Yes	Yes	No	Yes	Yes	No	Yes	N/A

Table 2-2
Scoping Evaluation for TN-68 Cask
(3 Pages)

UFSAR Drawing No.	Rev	Component	Item No.	Subcomponent Parts		Safety Classification	Intended Safety Function						Criterion	
							Confinement	Radiation Shielding	Sub-Criticality Control	Heat-Removal Capability	Structural Integrity	Retrievability	No. 1	No. 2
972-70-2	14	TN-68	24	Vent & Drain Port Cover Seal		ITS	Yes	No	No	No	No	No	Yes	N/A
972-70-2	14	TN-68	25	Vent & Drain Port Cover Bolts (SOC HD Cap)		ITS	Yes	No	No	No	Yes	No	Yes	N/A
972-70-2	14	TN-68	26	Overpressure Port Cover Bolts (SOC HD Cap)		ITS	Yes(3)	No	No	No	Yes	No	Yes	N/A
972-70-2	14	TN-68	27	Lid Alignment Pin		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	28	Basket Rail, Type 1		ITS	No	No	Yes	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	29	Basket Rail, Type 2		ITS	No	No	Yes	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	30	Basket Shim		ITS	No	No	Yes	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	31	Basket Rail Shim		NITS	No	No	No	Yes(2)	No	No	No	Yes
972-70-2	14	TN-68	32	Fuel Compartment		ITS	No	Yes	Yes	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	33a(4)	Poison Plate		ITS	No	Yes	Yes	Yes	No	No	Yes	N/A
972-70-2	14	TN-68	33b(4)	Aluminum Plate		ITS	No	Yes	Yes	Yes	No	No	Yes	N/A
972-70-2	14	TN-68	34	Strucural Plates		ITS	No	No	Yes	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	35	Flange (including stainless steel weld overlay)		ITS	Yes	Yes	No	Yes	Yes	No	Yes	N/A
972-70-2	14	TN-68	36	Shim		ITS	No	Yes	No	No	No	No	Yes	N/A
972-70-2	14	TN-68	37	Trunnion Bolt		ITS	No	No	No	No	Yes	Yes	Yes	N/A
972-70-2	14	TN-68	38	Deleted		-	-	-	-	-	-	-	-	-
972-70-2	14	TN-68	39	Basket Hold Down		ITS	No	Yes	Yes	No	No	No	Yes	N/A
972-70-2	14	TN-68	40	Shear Key		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	41	Pressure Relief Valve		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	42	Security Wire		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	43	Security Seal		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	44	Threaded Insert		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	45	Threaded Insert		ITS	Yes	No	No	No	Yes	No	Yes	N/A
972-70-2	14	TN-68	46	Flat Washer		NITS	No	No	No	No	No	No	No	No
972-70-2	14	TN-68	47	SOC Head Cap Screw		ITS	No	Yes	No	No	Yes	No	Yes	N/A

Table 2-2
Scoping Evaluation for TN-68 Cask
(3 Pages)

UFSAR Drawing No.	Rev	Component	Item No.	Subcomponent Parts	[ ]	Safety Classification	Intended Safety Function						Criterion	
							Confinement	Radiation Shielding	Sub-Criticality Control	Heat-Removal Capability	Structural Integrity	Retrievability	No. 1	No. 2
972-70-2	14	TN-68	48	Shield Ring		NITS	No	Yes(2)	No	No	No	No	No	Yes
972-70-2	14	TN-68	49	Flat Washer		NITS	No	No	No	No	No	No	No	No
N/A	N/A	TN-68	50	Not Used		-	-	-	-	-	-	-	-	-
N/A	N/A	TN-68	51	Not Used		-	-	-	-	-	-	-	-	-
972-70-7	1	TN-68	52	Fuel Compartment Extension		ITS	No	No	Yes	No	No	No	Yes	No
972-70-7	1	TN-68	53	End Cap, Top		ITS	No	No	Yes	No	No	No	Yes	No
972-70-7	1	TN-68	54	End Cap Bottom		ITS	No	No	Yes	No	No	No	Yes	No

1. The metallic seals consist of an inner spring, a lining, and a jacket. The spring is [ ] (a nickel alloy) or an equivalent material. The lining is stainless steel or nickel alloy. The jacket is made of aluminum.
2. Failure of this subcomponent could prevent fulfillment of this intended safety function.
3. The overpressure port cover, bolts, and seal are a barrier so that the pressure monitoring system is able to monitor the interseal pressure. Failure of the cover, bolts, or seal would not result in failure of the confinement boundary. However, they are classified as ITS and, thus, assigned a confinement function.
4. While Drawing 972-70-2 shows that Item #33 as "Poison Plates", Note #7 on Drawing 972-70-5 says "At the basket perimeter, Item 33 may be non-borated aluminum 1000 or 6000 series, as long as the pieces are retained in position by a positive means other than friction." To clearly identify that Item #33 could be two separate materials, Table 2-2 identified the poison plates as Item 33a and the optional aluminum plates as Item 33b.

Table 2-3
Scoping Evaluation for Spent Fuel Assembly Subcomponents

Subcomponent	Material of Construction	Intended Safety Function						Criterion	
		Confinement	Radiation Shielding	Sub-Criticality Control	Structural Integrity	Heat-Removal Capability	Retrievability	No. 1	No. 2
Fuel Pellets	Uranium oxide	No	No	No	No	No	No	No	No
Fuel Cladding and End Plugs	Zircaloy	Yes	No	Yes	Yes	Yes	No	Yes	N/A
Spacer Grid Assemblies	Inconel Zircaloy	No	No	Yes	Yes	No	No	Yes	N/A
Upper End Fitting/Nozzle (and related subcomponents)	Stainless Steel Inconel	No	No	No	Yes	No	No	Yes	N/A
Lower End Fitting/Nozzle (and related subcomponents)	Stainless Steel Inconel	No	No	No	Yes	No	No	Yes	N/A
Water Channels	Zircaloy	No	No	Yes	Yes	No	No	Yes	N/A
Fuel Channel	Zircaloy	No	No	Yes	No	Yes	No	Yes	N/A

CHAPTER 3
AGING MANAGEMENT REVIEW

CONTENTS

LIST OF TABLES

3.1 Introduction

This chapter describes the aging management review (AMR) of Certificate of Compliance (CoC) No. 1027 TN-68 dry storage cask system. The purpose of the AMR is to assess the need for aging management activities (AMA) for the structures, systems, and components (SSCs) determined to be within the scope of the CoC 1027 renewal. The AMR addresses aging mechanisms and effects that could adversely affect the ability of the SSCs to perform their intended safety functions during the period of extended operation (PEO).

Section 3.2 describes the AMR methodology, which follows the guidance and the processes of NUREG-1927 [3-1]. This section addresses each of the major steps of the AMR: Section 3.2.1 – Identification of Materials and Environments; Section 3.2.2 – Identification of Aging Mechanisms and Aging Effects, and Section 3.2.3 – Determination of the Activities Required to Manage the Effects of Aging.

Section 3.3 – Description of TN-68 Dry Storage Cask System, provides a brief description of the in-scope CoC 1027 SSCs.

Section 3.4 – Aging Management Review of Material/Environment, lists the materials of construction and the environments for the in-scope SSCs. The section also evaluates the potential aging degradation mechanisms/effects based on the material/environment combination. For each mechanism, a determination is made of whether the mechanism is considered “credible” in each environment that the material is exposed to during the PEO.

Sections 3.5, 3.6, and 3.7 provide the AMR results of the TN-68 cask, storage pad, and spent fuel assemblies (SFAs), respectively.

Section 3.8 provides a summary of the conclusions of the operating experience (OE) review contained in Appendix 3C.

Appendix 3A, Time-Limited Aging Analyses, identifies the calculations or analyses used to demonstrate that in-scope SSCs will maintain their intended safety function throughout an explicitly stated period of operation, i.e., time-limited aging analyses (TLAAs).

Appendix 3B, Supplemental Evaluations, provides a summary of supplemental evaluations and calculations performed to support the AMR and/or an element in an AMP.

Appendix 3C, Operating Experience Review, provides a summary of the operating experience review performed to support the AMR and aging management programs (AMPs).

3.2 Aging Management Review Methodology

The AMR follows the methodology recommended in NUREG-1927 [3-1]. The AMR provides an assessment of the aging effects that could adversely affect the ability of the SSCs to perform their intended safety functions during the PEO.

The AMR process involves the following major steps:

- Identification of materials and environments
- Identification of aging effects and mechanisms requiring management
- Determination of the activities required to manage the effects and mechanisms of aging; this involves the identification of TLAAs or AMPs for managing the effects of aging.

The scoping and screening evaluation in Chapter 2 identifies the in-scope SSCs for which potential aging effects must be identified and evaluated. For each SSC, the material of construction and the environment to which each SSC is exposed are determined. The component environments are determined based on the location of the component within the storage system. Once the component material/environment combinations are determined, a review of NUREG-2214 [3-2] is performed to identify credible aging degradation mechanisms for the different material/environment combinations. NUREG-2214 is based on technical literature, related research, industry information, and existing OE. After the credible aging mechanisms and effects are identified, it is determined whether the effects can be managed via a TLAA, or will require an AMP.

3.2.1 Identification of Materials and Environments

The first step in the AMR process is to identify the materials of construction for each subcomponent of the in-scope SSCs and the environments to which those materials are exposed during normal storage conditions. The combinations of materials and environments are used to identify the potential aging effects that require management during the PEO.

Materials

The TN-68 dry storage cask system SSCs and associated subcomponent materials of construction are summarized in:

- Table 3-5 – Aging Management Review for TN-68 Cask
- Table 3-7 – Aging Management Review for Storage pad
- Table 3-8 – Aging Management Review for Spent Fuel Assemblies

The materials of construction were identified through a review of the drawings provided in the Updated Final Safety Analysis Report (UFSAR) [3-3], along with other pertinent design information.

Environments

The environments to which SSCs and associated subcomponents are exposed play a critical role in the determination of potential aging mechanisms and effects. A review of the information presented in Chapter 2 of the UFSAR [3-3] was performed to assess the environmental conditions to which the SSCs are normally exposed. The configuration of a TN-68 dry storage cask system at an independent spent fuel storage installation (ISFSI) consisting of an array of individual TN-68 casks designed to provide an effective means of protection against extreme seasonal weather conditions as described in Reference [3-3].

The TN-68 dry storage cask system has been designed and qualified for a wide range of environmental conditions. The cask temperature response to changes in ambient conditions will be relatively slow because the cask thermal inertia is large. A daily average minimum temperature of -20 °F and a daily average maximum temperature of 100 °F were used in the thermal evaluation of the cask.

The environments to which the TN-68 dry storage cask system is exposed are affected by the characteristics of the ISFSI site environment, as well as by the component location within the storage system. Six basic environments that apply for the TN-68 dry storage cask system SSCs and subcomponents:

- Air–outdoor – In this environment, components are directly exposed to weather, including precipitation and wind. During storage, the exterior surfaces of the cask are exposed to all outdoor weather conditions, including insolation, wind, rain, snow, and site-specific ambient air conditions, including moist, possibly salt-laden atmospheric air, ambient temperatures, and humidity. Operating experience has shown moisture can penetrate the protective cover. Therefore, for the purpose of CoC renewal, no credit is taken for the protective cover as a form of weather protection.
- Embedded-in-concrete – In this environment, one or more surfaces of a component are in contact with concrete, e.g., rebar. This may prevent ingress of water and contaminants to the embedded surface, depending on the permeability of the embedding environment.
- Embedded-in-metal – In this environment, one or more surfaces of a component are in contact with another component or material. This may prevent ingress of water and contaminants to the embedded surface, depending on the permeability of the embedding environment. However, for the purposes of this AMR, the materials in this environment are treated as though they are exposed to the surrounding environment. For example, the surfaces of the poison plates are in contact with the fuel compartment and structural plates, but are treated as though they are exposed to the helium environment.

- Fully encased – In this environment the component is fully enclosed inside another component, or the surface between two components is sealed, or fully lined by another material (e.g., steel), which prevents ingress of water and contaminants. An example is the radial neutron shield encased between the gamma shield and the outer shell.
- Helium – In this environment, the component surfaces are exposed to the helium fill gas inside the cask and trace quantities of other gases, such as nitrogen, oxygen, argon, and fission product gases. This environment applies to the fuel pin, cladding, and other internal components inside the cask.
- Groundwater/Soil – In this environment, the component surface is exposed to a soil environment, i.e., below grade. Groundwater is subsurface water found in wells, tunnels, or drainage galleries, or water that flows naturally to the earth's surface via seeps or springs. Soil is a mixture of organic and inorganic materials produced by the weathering of rock and clay minerals or the decomposition of vegetation. Below-grade concrete structures are assumed to be partially exposed to a groundwater or soil environment.

The environments considered in the AMR are the environments that the TN-68 dry storage cask system SSCs and associated subcomponents normally experience. Environmental stressors that are conditions not normally experienced (such as extreme cold), or that may be caused by a design or fabrication condition, are considered event-driven and are not aging-related. Such event-driven situations are evaluated and corrective actions, if any, implemented at the time of the event.

3.2.2 Identification of Aging Mechanisms and Aging Effects

After the component material/environment combinations are identified, potential aging mechanisms are determined. NUREG-2214 [3-2] is reviewed to identify potential aging degradation mechanisms for different materials and environments.

Aging effects are the manifestation of aging mechanisms. In order to effectively manage an aging effect, it is necessary to determine the aging mechanisms that are potentially at work for a given material and environment combination. Therefore, the AMR process identifies both the aging effects and the associated aging mechanisms that cause them. Some aging mechanisms are only applicable at certain conditions such as high temperature or moisture. Each identified aging mechanism is characterized by a set of applicable conditions that must be met for the mechanism to occur. Given this evaluation process, each subcomponent that is subjected to AMR is evaluated to determine if the potential aging effects and mechanisms are credible considering the various material/environment combinations.

3.2.3 Determination of the Activities Required to Manage the Effects of Aging

For each subcomponent with a credible aging mechanism and effect, a determination is made to ascertain whether the effect can be managed via a TLAA or if an AMP is necessary.

TLAAs are calculations or analyses used to demonstrate that an in-scope SSC will maintain its intended safety function throughout the PEO. TLAAs have a time-dependent operating life such as fatigue life (cycles), change in a mechanical property such as fracture toughness or strength of materials due to irradiation. The TLAAs associated with CoC 1027 are identified in Appendix 3A.

AMPs are developed for managing the effects of aging. As appropriate, an AMP is created to summarize the activities to monitor and manage the aging effects. The AMPs credited for managing the effects of aging degradation are presented in Chapter 4.

3.3 Description of TN-68 Dry Storage Cask System

Chapter 2 lists the following in-scope SSCs for the TN-68 dry storage cask system.

- TN-68 cask
- ISFSI storage pad
- Spent fuel assemblies

3.3.1 Description of TN-68 Cask Subcomponents

The following major in-scope subcomponents of the TN-68 dry storage cask are described in this section.

- Cask body (and associated subcomponents)
- Fuel basket
- Cask lid
- Cask seals

Cask Body (and associated subcomponents)

The TN-68 cask containment vessel is comprised of an inner shell, which is a welded steel cylinder with an integrally-welded steel bottom; a welded flange forging (with stainless steel weld overlay); a bolted steel lid; and vent and drain port covers, cover seals and cover bolts. The confinement vessel is designed to the maximum practical extent as a Class I component in accordance with the provisions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Subsection NB, with certain code exceptions as described in the UFSAR [3-3]. The inner shell is surrounded by a thick gamma shield. Radial neutron shielding around the gamma shield is provided by borated polyester resin encased in aluminum. A steel outer shell surrounds the radial neutron shielding. A relief valve at the top of the radial neutron shield provides a vent path for the buildup of gases.

Additional neutron shielding is provided by a polypropylene top neutron shield.

The cask external surfaces are painted for ease of decontamination. The cask sealing surfaces are clad with stainless steel.

Trunnions are provided at the cask upper and lower ends to permit cask movement and transport. A protective weather cover fits over the top of the cask. The cover is sealed with an elastomer O-ring.

Fuel Basket

The fuel basket structure is made of an assembly of stainless steel cells joined by a fusion welding process to stainless steel plates. Above and below the plates are slotted borated aluminum or boron carbide/aluminum metal matrix composite plates (neutron poison plates) which form an egg-crate structure. The basket is supported laterally by aluminum rails which are attached to the periphery of the basket by welded studs. The fuel basket structure is designed to hold 68 fuel assemblies.

Cask Lid

The cask lid is fabricated of steel and secured to the cask body with bolts. A steel shield plate is attached to the bottom of the lid to provide additional radiation shielding. The cask lid sealing arrangement consists of double metallic O-ring seals. The cask lid and cask flange sealing surfaces are clad with a stainless steel weld overlay. Two penetrations are provided in the cask lid for cask venting and draining evolutions.

Cask Seals

The cask lid, drain port, and vent port are equipped with Helicoflex double metallic O-ring seals. The metal seals consist of a stainless steel or nickel alloy liner, a nickel-based alloy spring, and an outer aluminum jacket.

3.3.2 Description of Storage Pad Subcomponents

The storage pad is a reinforced concrete structure designed and constructed in accordance with codes and standards set by the general licensee. It is subject to site-specific foundation analyses and design considerations, including licensee-specific TN-68 cask loading configurations.

3.3.3 Description of Spent Fuel Assemblies Subcomponents

The TN-68 dry storage cask system is designed to store 68 boiling water reactor (BWR) spent fuel assemblies with or without fuel channels. Table 3-1 summarizes the fuel types, cladding material, and burnup limits approved for storage in the TN-68 dry storage cask system.

Fuel Cladding and End Plugs

The fuel rods consist of enriched UO_2 pellets inserted into the cladding tubes. End plugs are seal welded to each end. The cladding and end plugs confine the fuel pellets and fission gases. Each rod is pressurized with helium during fabrication.

Fuel Channel

The fuel channels of BWR fuel assemblies are mechanically attached and secured to the upper tie plate.

Spacer Grid Assemblies

The grid assemblies provide support for the fuel rods, positioning them in a square array and maintaining the designed rod pitch.

Lower Tie Plate (and related subcomponents)

The lower tie plate functions as the bottom structural element of a fuel assembly and positions the fuel rods laterally.

Upper Tie Plate (and related subcomponents)

The upper tie plate functions as the top structural element of a fuel assembly. It also interfaces with the fuel assembly grapple as the lifting point for the fuel assembly.

3.4 Aging Management Review of Material/Environment

3.4.1 Materials Evaluated

The materials of construction for the TN-68 dry storage cask system subcomponents, along with a material grouping, are listed in Table 3-5 through Table 3-8. The material groups represent a collection of individual material specifications that are susceptible to similar aging mechanism/effects and may, thus, be evaluated collectively. Table 3-2 provides a summary of the material groups.

3.4.2 Environments for the TN-68 Dry Storage Cask System SSCs

The environments to which the TN-68 dry storage cask system subcomponents are exposed are listed in Table 3-5 through Table 3-8. These environments are those that are normally (continuously) experienced by the subcomponents while in storage, and are described below. Note that some of the subcomponents are exposed to two or more environments. In Table 3-5 through Table 3-8, the symbol (I) denotes environments seen by surfaces that are internal to the subcomponent, or surfaces that face towards the interior of the cask. The symbol (E) denotes environments seen by surfaces that are external to the subcomponent, or surfaces that face towards the exterior of the cask.

Internal to the TN-68 Cask

Most of the internal subcomponents of the TN-68 cask are exposed to the inert gas (i.e., helium) environment inside the TN-68 cask cavity. The surfaces of some subcomponents are in contact with the surface of another subcomponent or material (i.e., an embedded-in-metal environment), e.g., the surfaces of the poison plate are in contact with the fuel compartment and structural plates. However, for the purpose of this AMR, if the space is not sealed, it is treated as though it is exposed to the helium environment. There are also other components that are in a fully encased environment, e.g., the exterior surface of the inner containment shell. The maximum fuel cladding temperature, cavity gas temperature, and cavity pressure during normal storage conditions were determined to be 649 °F, 422 °F, and 18.7 psig, respectively (Table 4B.3-1 and Table 4B.3-8 of Reference [3-3]). Table 3-3 summarizes the maximum component temperatures for the TN-68 cask during normal storage conditions.

The TN-68 cask internal components are exposed to significant neutron and gamma radiation.

External to the TN-68 Cask

Each TN-68 cask is positioned for long-term storage on a reinforced concrete storage pad located outdoors; thus, the exterior surfaces of the TN-68 cask are exposed to all weather conditions, including insolation, wind, rain, snow, and site-specific ambient air conditions, including moist, possibly salt-laden atmospheric air, ambient temperatures, and humidity, i.e., an air-outdoor environment.

Storage pad

The top surface of the storage pad is exposed to the air-outdoor environment. For the purpose of CoC renewal, no credit is taken for the cask sheltering the portion of the storage pad directly beneath the cask, i.e., it is assumed to be exposed to the air-outdoor environment. The below grade portion of the storage pad is exposed to a groundwater/soil environment.

Spent Fuel Assemblies

Since the spent fuel assemblies are located within the sealed TN-68 cask, their subcomponents are exposed to the internal TN-68 cask environment, i.e., helium environment. During fabrication, each fuel pin is pressurized with helium. Hence, the internal environment of a fuel pin will consist of helium and fission gases, i.e., a helium environment.

3.4.3 Aging Effects Requiring Management

This section evaluates known aging degradation mechanisms/effects for the TN-68 dry storage cask system materials of construction. For each mechanism, a determination is made of whether the mechanism is considered "credible" in each environment that the material is exposed to during the PEO. A credible aging mechanism is one that could manifest into an aging effect that affects an important-to-safety function during the PEO. The evaluation relied upon the information in NUREG-2214 [3-2] for identifying known aging degradation mechanisms and for determining if they are credible for the material/environment combination.

3.4.3.1 Aging Mechanism of Steel Material

A review of Table 3-5 through Table 3-8 shows that the environments that the steel subcomponents are exposed to are:

- Air-outdoor
- Embedded-in-concrete
- Helium
- Fully encased

The following aging mechanisms for steel material were evaluated to determine if they are credible in the environments that steel is exposed to:

- General corrosion
- Pitting and crevice corrosion
- Galvanic corrosion
- Microbiologically influenced corrosion (MIC)
- Stress corrosion cracking (SCC)

- Creep
- Fatigue
- Thermal aging
- Radiation embrittlement
- Stress relaxation
- Wear

3.4.3.1.1 General Corrosion of Steel Material

General corrosion, also known as uniform corrosion, proceeds at approximately the same rate over a metal surface. Freely exposed steel surfaces in contact with moist air or water are subject to general corrosion.

In an air-outdoor environment, rain, fog, snow, and dew condensation can generate moisture layers on the steel surface that cause general corrosion.

In the embedded-in-concrete environment, the concrete provides an alkaline solution that passivates the steel. However, if the concrete degrades, the embedded steel could be exposed to water containing dissolved carbonates and chlorides and general corrosion is possible.

In the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, the corrosion reaction with steel will be limited. Similarly, there is very little moisture in a fully encased environment and, thus, the corrosion reaction with steel will be limited.

Therefore, general corrosion of the steel material is considered credible in air-outdoor or embedded-in-concrete environments, but not credible in helium or fully encased environments.

3.4.3.1.2 Pitting and Crevice Corrosion of Steel Material

Pitting corrosion is a localized form of corrosion that is confined to a point or small area of a metal surface. It takes the form of cavities called pits. Crevice corrosion is another localized form of corrosion that occurs in a wetted environment when a crevice exists. It occurs more frequently in connections, lap joints, splice plates, bolt threads, under bolt heads, or at points of contact between metals and nonmetals. Crevice corrosion is associated with stagnant or low-flow solutions. Steel is susceptible to pitting and crevice corrosion in an oxidizing and alkaline environment, especially in the presence of chlorides.

The potential to form aqueous electrolytes on the steel surfaces exposed to an air-outdoor environment is present via direct exposure to precipitation or through deliquescence of deposited salts. These electrolytes could be conducive to pitting and crevice corrosion of steel. As such, there is the potential for pitting and crevice corrosion of steel subcomponents exposed to an air-outdoor environment.

In an embedded-in-concrete environment, if the concrete degrades with time, steel can be exposed to water containing dissolved carbonates and chlorides, which could be conducive to pitting and crevice corrosion as well.

In the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, the corrosion reaction with steel will be limited. Similarly, there is very little moisture in a fully encased environment and, thus, the corrosion reaction with steel will be limited.

Therefore, pitting and crevice corrosion of the steel material is considered credible in air-outdoor or embedded-in-concrete environments, but not credible in helium or fully encased environments.

3.4.3.1.3 Galvanic Corrosion of Steel Material

Galvanic corrosion occurs when two dissimilar metals or conductive materials are in physical contact in the presence of a conducting solution. Under these conditions, an electrolytic cell is formed, transmitting an electrical current between an anode (i.e., less noble material) and a cathode (i.e., more noble material). Oxidation occurs at the anode, and reduction occurs at the cathode. In a TN-68 cask, galvanic coupling exists between steel and other more noble materials such as stainless steel, graphite, and nickel.

The potential to form aqueous electrolytes on the steel surfaces exposed to an air-outdoor environment is present via direct exposure to precipitation or through deliquescence of deposited salts. Because these electrolytes could initiate steel corrosion, and corrosion of steel is expected to be enhanced under galvanic coupling, loss of material due to galvanic corrosion of steel is considered credible in dissimilar metal couples.

In an embedded-in-concrete environment, the steel will not be in contact with a dissimilar metal or conductive material.

In the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, very few aqueous electrolytes. Similarly, there is very little moisture in a fully encased environment and, thus, very few aqueous electrolytes.

Therefore, galvanic corrosion of the steel material is considered credible in an air-outdoor environment, but not credible in embedded-in-concrete, helium or fully encased environments.

3.4.3.1.4 Microbiologically Influenced Corrosion of Steel Material

Microbiologically influenced corrosion is corrosion caused or promoted by the metabolic activity of microorganisms. Active microbial metabolism requires water in the form of water vapor, condensation, or deliquescence, and available nutrients to support microbial activity. Biofilms can form even under radiation environments. MIC is limited where relative humidity is below 90 percent and negligible for relative humidity below 60 percent. Although most of the evidence of MIC for metallic components is from conditions under which the metal surface is kept continuously wet, microorganisms can live in many environments, such as water, soil, and air, where aerobic bacteria (e.g., iron-manganese oxidizing bacteria, sulfur/sulfide oxidizing bacteria, methane producers, organic acid-producing bacteria), fungi, and algae can develop.

The potential to form aqueous electrolytes on the steel surfaces exposed to an air-outdoor environment is present via direct exposure to precipitation or via deliquescence of deposited salts. These electrolytes have the potential to support microbial activity. However, there is no OE of MIC degradation of steel engineering components that are exposed to environments similar to those of dry storage cask systems, where continuous exposure to a relative humidity above 90 percent is not expected. The OE of MIC for metallic components is largely from instances in which the metal surface remained continuously wet. Because there is no applicable OE of MIC damage of steel under relevant atmospheric conditions, MIC is not considered credible in an air-outdoor environment.

In an embedded-in-concrete environment, if the concrete is exposed to a groundwater/soil environment and is degraded, the steel could be exposed to groundwater or soil. Under these conditions, the steel could be susceptible to MIC.

In the helium environment, there is very little residual water and nutrients in internal environments of a TN-68 cask following drying and refilling with inert helium gas. Similarly, there is very little moisture and nutrients in a fully encased environment.

Therefore, MIC of the steel material is considered credible in an embedded-in-concrete environment if the concrete is exposed to a ground/soil environment, but is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.1.5 Stress Corrosion Cracking of Steel Material

Stress corrosion cracking is the cracking of a metal produced by the combined action of corrosion and tensile stress (applied or residual). SCC is highly chemical-specific in that certain alloys are likely to undergo SCC only when exposed to a small number of chemical environments. SCC is the result of a combination of three factors: (1) a susceptible material, (2) exposure to a corrosive environment, and (3) tensile stresses. High-strength steels with yield strengths greater than or equal to 150,000 pounds per square inch (150 ksi) have been found to be susceptible to SCC under exposure to aqueous electrolytes.

Since the steels used in the construction of the TN-68 dry storage cask system are not high-strength steels, i.e., they have a yield strength less than 150 ksi and, they are not susceptible to SCC.

Therefore, SCC of the steel material is not credible in air-outdoor, embedded-in-concrete, helium, or fully encased environments.

3.4.3.1.6 Creep of Steel Material

Creep is the time-dependent, inelastic deformation that takes place at an elevated temperature and a constant stress. Because the deformation processes that produce creep are thermally activated, the rate of this time-dependent deformation is a strong function of the temperature. The creep rate also depends on the applied stress, but does not normally vary with the environment. As a general rule of thumb, at temperatures below $0.4T_m$, where T_m is the melting point of the metal in Kelvin (K), thermal activation is insufficient to produce creep that could compromise the intended functions of SSCs. With a melting point of 1,789 °K (2,760 °F), temperatures of at least 716 °K (829 °F) are required to initiate creep in steels. However, the $0.4T_m$ rule of thumb underestimates the minimum creep temperature for steels, because it has been found that temperatures above 932 °F are required for creep in steels.

The highest temperatures within the TN-68 dry storage cask system are at locations close to the fuel rods. As shown in Table 3-3, the maximum fuel cladding temperature for fuel in a TN-68 cask was determined to be 649 °F at the beginning of storage. Because the fuel rods are the only heat source within the system, they provide upper temperature limits for all subcomponents regardless of their environment. It is apparent from the temperatures in Table 3-3 that the steel subcomponents will not approach the minimum 932 °F required for creep to occur in steels.

Therefore, creep of the steel material is not credible in air-outdoor, embedded-in-concrete, helium, or fully encased environments.

3.4.3.1.7 Fatigue of Steel Material

Fatigue is the progressive and localized structural damage that occurs when a material is subjected to cyclic loading. The only cyclic loading experienced by the TN-68 cask steel material is associated with thermal cycling. The only source of potential thermal fatigue of the TN-68 cask is ambient seasonal and daily temperature fluctuation. Due to the large thermal inertia of the TN-68 cask, it does not experience the full amplitude of ambient temperature cycles, and a gradual, long-term temperature decrease occurs during the course of storage.

Section 3.2.1.7 of NUREG-2214 [3-2] calls for a review of all fatigue analyses contained in the design basis documents to determine whether the renewal application adequately addresses the implications of extending the operating period to 60 years. It also says that if no fatigue analysis was performed in support of the component design, no action is required. The review of the design basis documents for fatigue analyses is in Appendix 3A. The review did not identify any fatigue analyses/evaluations for the TN-68 dry storage cask system components. Therefore, per NUREG-2214 [3-2], no action is required.

Therefore, fatigue of the steel material is not credible in air-outdoor, embedded-in-concrete, helium, or fully encased environments.

3.4.3.1.8 Thermal Aging of Steel Material

The microstructures of most steels will change, given sufficient time at temperature, and this can affect mechanical properties. The degree to which thermal aging occurs depends on the steel grade and the exposure time and temperature.

The effects of elevated storage temperatures on material properties were evaluated during the initial license application. Carbon steels in the normalized condition (ferrite/pearlite microstructures) are commonly used in the petroleum and chemical industry with exposure temperatures of approximately 400 °C (752 °F). Per Table 3-3 the maximum temperature of a steel subcomponent (i.e., the inner shell) is only 340 °F at the beginning of the initial storage period. Therefore, it can be concluded that thermal aging is not expected to produce degradation of the mechanical properties of steels in the PEO.

Therefore, thermal aging of the steel material is not credible in air-outdoor, embedded-in-concrete, helium, or fully encased environments.

3.4.3.1.9 Radiation Embrittlement of Steel Material

Embrittlement of metals may occur under exposure to neutron radiation. Depending on the neutron fluence, radiation can cause changes in mechanical properties, such as loss of ductility, reduced fracture toughness, and decreased resistance to cracking.

Section 3.2.1.9 of NUREG-2214 [3-2] states that neutron fluence levels greater than 10^{19} neutrons per square centimeter (n/cm^2) are required to produce a measureable degradation of the mechanical properties. NUREG-2214 [3-2] then describes a bounding calculation that shows estimated fluence level within a dry storage cask system is three orders of magnitude below the levels reported to degrade the fracture resistance of carbon and alloy steels.

Therefore, radiation embrittlement of the steel material is not credible in air-outdoor, embedded-in-concrete, helium, or fully encased environments.

3.4.3.1.10 Stress Relaxation of Steel Material

Stress relaxation of bolting or other tightening subcomponents is the steady loss of elastic stress in a loaded part due to atomic movement at elevated temperature. It results in a loss of clamping forces or preload in a heavily loaded joint. Stress relaxation is a strong function of temperature and bolt material. It also depends on geometry of the bolt and thread quality. It decreases with time, as the tensile stress in the bolt decreases.

Research has demonstrated that the residual stress of carbon steel bolts due to relaxation is about 85 percent of the initial applied stress at temperatures greater than about 212 °F. Table 3-3 shows that the initial temperature of the lid is greater than the 212 °F threshold. Therefore, bolts in contact with the lid (i.e., in an air-outdoor environment) may also be above the stress relaxation temperature threshold. A review of Table 3-5 through Table 3-8 shows that there are no bolts in an embedded-in-concrete, helium, or fully encased environment.

Therefore, stress relaxation of the steel bolt material is considered credible in an air-outdoor environment, but not credible in embedded-in-concrete, helium, or fully encased environments.

3.4.3.1.11 Wear of Steel Material

Rolling contact wear results from the repeated mechanical stressing of the surface of a body rolling on another body.

There are no steel bodies rolling on another body while the TN-68 casks are in storage.

Therefore, wear of the steel material is not credible in air-outdoor, embedded-in-concrete, helium, or fully encased environments.

3.4.3.2 Aging Mechanism of Stainless Steel Material

A review of Table 3-5 through Table 3-8 shows that the environments that the stainless steel subcomponents are exposed to are:

- Air-outdoor
- Helium
- Fully encased

The following aging mechanisms for stainless steel material were evaluated to determine if they are credible in the environments that stainless steel is exposed to:

- General corrosion
- Pitting and crevice corrosion
- Galvanic corrosion

- MIC
- SCC
- Creep
- Fatigue
- Thermal aging
- Radiation embrittlement
- Stress relaxation
- Wear

3.4.3.2.1 General Corrosion of Stainless Steel Material

Stainless steels exhibit passive behavior in all environments, resulting in negligible general corrosion rates.

Therefore, general corrosion of the stainless steel material is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.2.2 Pitting and Crevice Corrosion of Stainless Steel Material

Pitting corrosion is a localized form of corrosion that is confined to a point or small area of a metal surface. It takes the form of cavities called pits. Crevice corrosion is another localized form of corrosion that occurs in a wetted environment when a crevice exists that allows a corrosive environment to develop. It occurs more frequently in connections, lap joints, splice plates, bolt threads, under bolt heads, or at points of contact between metals and nonmetals. Crevice corrosion is associated with stagnant or low-flow solutions. Stainless steel is known to be susceptible to pitting and crevice corrosion.

The potential to form aqueous electrolytes on the steel surfaces exposed to an air-outdoor environment is present via direct exposure to precipitation or through deliquescence of deposited salts. These electrolytes could be conducive to pitting and crevice corrosion of stainless steel. As such, there is the potential for pitting and crevice corrosion of stainless steel subcomponents exposed to an air-outdoor environment.

In the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, the corrosion reaction with stainless steel will be limited. Similarly, there is very little moisture in a fully encased environment and, thus, the corrosion reaction with stainless steel will be limited.

Therefore, pitting and crevice corrosion of the stainless steel material is considered credible in air-outdoor environment, but not credible in helium, or fully encased environments.

3.4.3.2.3 Galvanic Corrosion of Stainless Steel Material

Galvanic corrosion occurs when two dissimilar metals or conductive materials are in physical contact in the presence of a conducting solution. Under these conditions, an electrolytic cell is formed, transmitting an electrical current between an anode (i.e., less noble material) and a cathode (i.e., more noble material). Oxidation occurs at the anode, and reduction occurs at the cathode. Galvanic coupling exists between stainless steel and other more noble materials such as graphite.

The potential to form aqueous electrolytes on the stainless steel surfaces exposed to an air-outdoor environment is present via direct exposure to precipitation or through deliquescence of deposited salts. Because electrolytes conducive to galvanic corrosion may exist, galvanic corrosion of stainless steel in contact with graphite lubricants is considered credible.

In the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, very few aqueous electrolytes. Similarly, there is very little moisture in a fully encased environment and, thus, very few aqueous electrolytes.

Therefore, galvanic corrosion of the stainless steel material is considered credible in an air-outdoor environment when in contact with a graphite lubricant, but not credible in helium, or fully encased environments.

3.4.3.2.4 Microbiologically Influenced Corrosion of Stainless Steel Material

As discussed in Section 3.4.3.1.4, MIC is caused or promoted by the metabolic activity of microorganisms. Microorganisms can live in many environments, such as water, soil, and air, where aerobic bacteria (e.g., iron-manganese oxidizing bacteria, sulfur/sulfide oxidizing bacteria, methane producers, and organic acid-producing bacteria), fungi, and algae can develop.

The potential to form aqueous electrolytes on the stainless steel surfaces exposed to an air-outdoor environment is present via direct exposure to precipitation or through deliquescence of deposited salts. These electrolytes have the potential to support microbial activity. However, there has not yet been any OE of MIC in atmospheric environments where stainless steel surfaces are only intermittently wetted. Because there is no applicable OE of MIC damage of stainless steel under relevant atmospheric conditions, MIC is not considered credible in an air-outdoor environment.

In the helium environment, there is very little residual water and nutrients in internal environments of a TN-68 cask following drying and refilling with inert helium gas. Similarly, there is very little moisture and nutrients in a fully encased environment.

Therefore, MIC of the stainless steel material is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.2.5 Stress Corrosion Cracking of Stainless Steel Material

Stress corrosion cracking is the cracking of a metal produced by the combined action of corrosion and tensile stress and is highly chemical-specific. Most ferritic and duplex stainless steels are either immune or highly resistant to SCC; however, all austenitic grades, especially Types 304, 304L, 304LN, 316, 316L, and 316LN, are susceptible to chloride-induced SCC in the normal wrought condition. This susceptibility increases when the material is sensitized. In the welded condition, the heat affected zone (HAZ), which is a thin band located adjacent to the weld, can be sensitized by the precipitation of carbides that extract chromium out of the metal matrix.

In the helium environment, there is a lack of halides and very little residual water in internal environments of a TN-68 following drying and refilling with inert helium gas and, thus, SCC is not considered credible. Similarly, there is a lack of halides and very little moisture in a fully encased environment and, thus, SCC is not considered credible.

The construction of the TN-68 dry storage cask system does not involve welding of stainless steel material together in an air-outdoor environment.

Therefore, SCC of the stainless steel material is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.2.6 Creep of Stainless Steel Material

As discussed in Section 3.4.3.1.6, as a general rule of thumb, at temperatures below $0.4T_m$, where T_m is the melting point of the metal in Kelvin, thermal activation is insufficient to produce creep that could compromise the intended functions of SSCs. The term “stainless steel” covers a wide range of compositions and microstructures, including austenitic, ferritic, martensitic, duplex, and precipitation hardening stainless steels. Using the melting point temperature for austenitic or 300 series stainless steels (because they are most commonly used in the dry storage cask systems and have the lowest melting point) of 1,698 °K (2,597 °F), temperatures of at least 679 °K (763 °F) are required to initiate creep in the austenitic stainless steels.

The highest temperatures within the TN-68 dry storage cask system are at locations close to the fuel rods. As shown in Table 3-3, the maximum fuel cladding temperature for fuel in a TN-68 cask was determined to be 649 °F at the beginning of storage. Because the fuel rods are the only heat source within the system, they provide upper temperature limits for all subcomponents regardless of their environment. It is apparent from the temperatures in Table 3-3 that internal stainless steel subcomponents will not approach the minimum 763 °F temperature that has been found to be required for creep to occur in stainless steels.

Therefore, creep of the stainless steel material is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.2.7 Fatigue of Stainless Steel Material

Fatigue is the progressive and localized structural damage that occurs when a material is subjected to cyclic loading. The only cyclic loading experienced by the TN-68 cask stainless steel material is associated with thermal cycling. The only source of potential thermal fatigue of the TN-68 cask is ambient seasonal and daily temperature fluctuation. Due to the large thermal inertia of the TN-68 cask, it does not experience the full amplitude of ambient temperature cycles, and a gradual, long-term temperature decrease occurs during the course of storage.

Section 3.2.1.7 of NUREG-2214 [3-2] calls for a review of all fatigue analyses contained in the design basis documents to determine whether the renewal application adequately addresses the implications of extending the operating period to 60 years. It also says that if no fatigue analysis was performed in support of the component design, no action is required. The review of the design basis documents for fatigue analyses is in Appendix 3A. The review did not identify any fatigue analyses/evaluations for the TN-68 dry storage cask system components. Therefore, per NUREG-2214 [3-2], no action is required.

Therefore, fatigue of the stainless steel material is not credible in air-outdoor, embedded-in-concrete, helium, or fully encased environments.

3.4.3.2.8 Thermal Aging of Stainless Steel Material

The microstructures of most stainless steels will change, given sufficient time at temperature, and these changes may alter the material's strength and fracture toughness. For stainless steel subcomponents, the thermal aging process differs for welded and non-welded subcomponents.

Welded Austenitic Stainless Steel Subcomponents in Helium

The ferrite present in austenitic stainless steel welds can transform by spinodal decomposition to form iron-rich alpha and chromium-rich alpha prime phases, and further aging can produce an intermetallic G-phase. The spinodal decomposition and the formation of the intermetallic G-phase takes place during extended exposure to temperatures between 300 and 400 °C (572 and 752 °F). The highest temperatures within the TN-68 dry storage cask system are at locations close to the fuel rods. As shown in Table 3-3, the maximum fuel cladding temperature for fuel in a TN-68 cask was determined to be 649 °F at the beginning of storage. Because the fuel rods are the only heat source within the system, they provide upper temperature limits for all subcomponents regardless of their environment. It is apparent from the temperatures in Table 3-3 that internal stainless steel subcomponents may be above the 300 °C (572 °F) minimum temperature required for these phase changes.

Based on Charpy impact toughness testing of cast duplex stainless steels, it is concluded that ferrite levels above 15 percent are required for significant embrittlement, because ferrite resides in discrete islands below this level and does not provide a continuous low-toughness fracture path. Because most welds contain around 4 to 15 percent ferrite, substantial embrittlement of austenitic stainless steel welds is not expected. NUREG/CR-6428 [3-7] concluded that thermal aging produced moderate decreases (no more than 25 percent) in the upper shelf Charpy impact energy and relatively small decreases in the fracture toughness of a wide range of austenitic welds. Although the phase changes associated with thermal embrittlement of austenitic stainless steel welds could take place in subcomponents near the fuel within the 60-year timeframe, the minor reductions in fracture toughness that would be produced in the weld indicate that this is not a credible aging mechanism for subcomponents in proximity to the fuel rods.

Non-Welded Austenitic Stainless Steel Subcomponents in Helium

Because the phase changes described previously occur only within the ferrite-containing, HAZ of a weld, embrittlement will not occur in austenitic stainless steel subcomponents that do not contain a weld.

Stainless Steel Subcomponents in Air-Outdoor, and Fully Encased Environments

Because the peak temperatures for stainless steel subcomponents exposed to air-outdoor, and fully encased environments are below the temperature required for the phase changes associated with thermal embrittlement of stainless steels, thermal aging is not considered credible for these subcomponents.

Summary

Therefore, thermal aging of the stainless steel material is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.2.9 Radiation Embrittlement of Stainless Steel Material

Embrittlement of metals may occur under exposure to neutron radiation. Depending on the neutron fluence, radiation can cause changes in mechanical properties, such as loss of ductility, reduced fracture toughness, and decreased resistance to cracking.

Section 3.2.2.9 of NUREG-2214 [3-2] states that neutron fluence levels greater than 10^{20} n/cm^2 are required to produce a measureable degradation of the mechanical properties. NUREG-2214 [3-2] then describes a bounding calculation that showed estimated fluence level within a dry storage cask system is four orders of magnitude below the levels that would degrade the mechanical properties of stainless steel.

Therefore, radiation embrittlement of the stainless steel material is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.2.10 Stress Relaxation of Stainless Steel Material

Stress relaxation of bolting or other tightening subcomponents is the steady loss of elastic stress in a loaded part due to atomic movement at elevated temperature. It results in a loss of clamping forces or preload in a heavily loaded joint. Stress relaxation is a strong function of temperature and bolt material. It also depends on geometry of the bolt and thread quality. It decreases with time, as the tensile stress in the bolt decreases.

Research has demonstrated that the loss of initial applied stress in austenitic stainless steel bolting due to stress relaxation is negligible at temperatures below 572 °F. Table 3-3 shows that the initial maximum temperature of the lid is less than this threshold. Therefore, stainless steel bolts in contact with the lid (i.e., in an air-outdoor environment) are below the stress relaxation temperature threshold. A review of Table 3-5 through Table 3-8 shows that there are no bolts in a helium or fully encased environment.

Therefore, stress relaxation of the stainless steel bolt material is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.2.11 Wear of Stainless Steel Material

Rolling contact wear results from the repeated mechanical stressing of the surface of a body rolling on another body.

There are no stainless steel bodies rolling on another body while TN-68s are in storage. While the upper trunnions are bolted to the TN-68 cask, Section 1.2.1 of the UFSAR [3-3] states that they are not intended to be removed during storage and, thus, are not intended to be reused.

Therefore, wear of other stainless steel material is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.3 Aging Mechanisms of Aluminum Material

A review of Table 3-5 through Table 3-8 shows that the only environment that the aluminum subcomponents are exposed to is:

- Helium
- Fully encased
- Air-outdoor

The following aging mechanisms for aluminum material were evaluated to determine if they are credible in the environments that aluminum is exposed to:

- General corrosion
- Pitting and crevice corrosion

- Galvanic corrosion
- MIC
- Creep
- Fatigue
- Thermal aging
- Radiation embrittlement

3.4.3.3.1 General Corrosion of Aluminum Material

General corrosion, also known as uniform corrosion, proceeds at approximately the same rate over a metal surface. Freely exposed aluminum surfaces in contact with moist air or water are subject to general corrosion. The corrosion rate depends on solution composition, pH, and temperature. The corrosion rate of aluminum is normally controlled by the formation of a passive film of Al_2O_3 at the metal and water interface.

In an air-outdoor environment, rain, fog, snow, and dew condensation can generate moisture layers on the aluminum surface that cause general corrosion.

In the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, the corrosion reaction with aluminum will be limited. Similarly, there is very little moisture in a fully encased environment and, thus, the corrosion reaction with aluminum will be limited.

Therefore, general corrosion of the aluminum material is credible in an air-outdoor environment but not credible in helium, or fully encased environments.

3.4.3.3.2 Pitting and Crevice Corrosion of Aluminum Material

As discussed in Section 3.4.3.1.2, pitting corrosion is a localized form of corrosion that is confined to a point or small area of a metal surface, and crevice corrosion occurs in a wetted environment when a crevice exists that allows a corrosive environment to develop in a component. Aluminum and its alloys form a passive film on the surface. However, localized corrosion in the form of pitting or crevice corrosion could occur on aluminum subcomponents, especially in the presence of halides.

In an air-outdoor environment, rain, fog, snow, and dew condensation can generate moisture layers on the aluminum surface that cause corrosion reaction.

In the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, the corrosion reaction with aluminum will be limited. Similarly, there is very little moisture in a fully encased environment and, thus, the corrosion reaction with aluminum will be limited.

Therefore, pitting and crevice corrosion of the aluminum material is credible in an air-outdoor environment but not credible in helium, or fully encased environments.

3.4.3.3.3 Galvanic Corrosion of Aluminum Material

As discussed in Section 3.4.3.1.3, galvanic corrosion occurs when two dissimilar metals or conductive materials are in physical contact in the presence of a conducting solution. In dry storage cask systems, galvanic coupling exists between aluminum and steel, and stainless steel (where aluminum is less noble in each case).

The potential to form aqueous electrolytes on surfaces exposed to an air-outdoor environment is present via direct exposure to precipitation or through deliquescence of deposited salts. Because these electrolytes could initiate aluminum corrosion, and corrosion of aluminum is expected to be enhanced under galvanic coupling, loss of material due to galvanic corrosion of aluminum is considered credible in dissimilar metal couples.

In the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, very few aqueous electrolytes. Similarly, there is very little moisture in a fully encased environment and, thus, very few aqueous electrolytes.

Therefore, galvanic corrosion of the aluminum material is considered credible in an air-outdoor environment, but not credible in helium or fully encased environments.

3.4.3.3.4 Microbiologically Influenced Corrosion of Aluminum Material

Microbiologically influenced corrosion (MIC) is corrosion caused or promoted by the metabolic activity of microorganisms. Active microbial metabolism requires water in the form of water vapor, condensation, or deliquescence, and available nutrients to support microbial activity. Biofilms can form even under radiation environments. Microbiologically influenced corrosion is limited where relative humidity is below 90 percent and negligible for relative humidity below 60 percent. Although most of the evidence of MIC for metallic components is from conditions under which the metal surface is kept continuously wet, microorganisms can live in many environments, such as water, soil, and air, where aerobic bacteria (e.g., iron-manganese oxidizing bacteria, sulfur/sulfide oxidizing bacteria, methane producers, and organic acid-producing bacteria), fungi, and algae can develop.

However, there is no operating experience of MIC degradation of aluminum engineering components that operate in environments similar to those of dry cask storage systems. All of the OE of MIC for metallic components is from conditions in which the metal surface is kept continuously wet. Due to the absence of any applicable experience of MIC damage of aluminum components under atmospheric conditions, MIC is not considered to be significant in an air-outdoor environment.

In the helium environment, there is very little residual water and nutrients in internal environments of a TN-68 cask following drying and refilling with inert helium gas. Similarly, there is very little moisture and nutrients in a fully encased environment.

Therefore, MIC of the aluminum material is not credible in air-out door, helium or fully encased environments.

3.4.3.3.5 Creep of Aluminum Material

Creep is the time-dependent, inelastic deformation that takes place at an elevated temperature and a constant stress. Because the deformation processes that produce creep are thermally activated, the rate of this time-dependent deformation is a strong function of the temperature. The creep rate also depends on the applied stress but does not generally vary with the environment. As a general rule of thumb, at temperatures below $0.4T_m$, where T_m is the melting point of the metal in Kelvin, thermal activation is insufficient to produce creep that could compromise the intended functions of SSCs. With a melting point of 911 to 930 °K (1,180 to 1,215 °F), temperatures of at least 364 to 372 °K (196 to 210 °F) are required to initiate creep in aluminum. These temperatures are consistent with other research that which indicates that creep in aluminum is possible at temperatures greater than 212 °F. Microstructure also plays a significant role in the resistance to creep of a metal. Hence, while this 212 °F minimum temperature for creep is representative for pure aluminum, creep in precipitation-hardened aluminum alloys does not become significant until about 392 °F. Additionally, at temperatures near these threshold values, high stresses are required to produce creep.

Subcomponents that do not serve a structural function are not under loads other than their own weight, and in many instances, their weight is also supported by adjacent structures. Due to the minimal applied loads, creep of nonstructural subcomponents will not produce significant damage during the 60-year timeframe.

A review of Table 3-5 through Table 3-8 shows that the only aluminum sub-components with a structural function are the basket rails. These basket rails are constructed out of precipitation-hardened 6061-T6 aluminum. Table 3-3 shows that the maximum temperature for the basket rails is 402 °F. While this value is slightly above the creep threshold temperature of 392 °F for precipitation-hardened aluminum alloys, it is expected to drop below the threshold value by the end of the initial storage period. In addition, the basket rails are under no loads other than their own weight and, thus, are not expected to experience creep.

Therefore, creep of the aluminum material is not credible in air-outdoor, helium or fully encased environments.

3.4.3.3.6 Fatigue of Aluminum Material

Fatigue is the progressive and localized structural damage that occurs when a material is subjected to cyclic loading. The only cyclic loading experienced by the TN-68 cask aluminum material is associated with thermal cycling. The only source of potential thermal fatigue of the TN-68 cask is ambient seasonal and daily temperature fluctuation. The TN-68 cask does not experience the full amplitude of ambient temperature cycles, and a gradual, long-term temperature decrease occurs during the course of storage. The seasonal and daily variations in ambient conditions are ameliorated by the thermal mass of the TN-68 cask.

Section 3.2.3.6 of NUREG-2214 [3-2] calls for a review of all fatigue analyses contained in the design basis documents to determine whether the renewal application adequately addresses the implications of extending the operating period to 60 years. It also says that if no fatigue analysis was performed in support of the component design, no action is required. The review of the design basis documents for fatigue analyses is in Appendix 3A. The review did not identify any fatigue analyses/evaluations for the TN-68 dry storage cask system components. Therefore, per NUREG-2214 [3-2], no action is required.

Therefore, fatigue of the aluminum material is not credible in air-outdoor, helium, or fully encased environments.

3.4.3.3.7 Thermal Aging of Aluminum Material

The microstructures of most aluminum alloys will change, given sufficient time at temperature, and this can affect mechanical properties. The effect of thermal aging (i.e., loss of strength) will depend on the time at temperature and the microstructure and chemical composition of the aluminum subcomponents. This loss of strength could be an issue for any subcomponents that perform a structural function.

A review of Table 3-5 through Table 3-8 shows that the only aluminum sub-components that have a structural function are the basket rails. These subcomponents are constructed out of precipitation-hardened 6061-T6 aluminum. Research has shown that when alloy 6061-T6 is held at 392 °F its yield strength drops from approximately 18 ksi at 10,000 hours (1.14 years) to approximately 11.5 ksi at 100,000 hours (11.4 years). Because of this sensitivity to exposure time, ASME B&PV Code Section II requires that time-dependent properties be used for exposures above 350 °F for this alloy. UFSAR [3-3] Section 3.3.2 states that the material properties of the Aluminum alloy (6061-T6) were taken from the ASME Code.

Table 3-3 shows that when the cask is initially placed in storage, the maximum temperature for the basket rails is 402 °F. As the temperatures inside the cask decrease during the initial storage period (e.g., Section 3.2.3.7 of NUREG-2214 [3-2] describes a 242 °F decrease in the maximum fuel cladding temperature) it is expected that the basket rails temperatures will drop below 350 °F by the end of the initial storage period. Therefore, thermal aging of aluminum is not expected to be a concern during the PEO. In addition, the summary of the basket rail structural analysis in UFSAR [3-3] Section 3B.6 shows that the limiting stress in the basket aluminum rail meets the allowable stress limit for a 98 g side drop load which is much higher than the maximum calculated g-load of 77 g. Note that these g-loadings are at the top of the basket while the area of the higher temperature is in the middle of the basket where the g-loads are much lower. Due to high margin of safety compared against the ASME code allowable limit, it is reasonable to conclude that any prolonged elevated temperature exposure resulting in reduced strength of aluminum alloy will not affect the TN-68 basket rails. Therefore, any thermal aging of the basket rails is not expected to result in the loss of their structural intended safety function during the PEO.

Therefore, thermal aging of the aluminum material is not credible in air-outdoor, helium or fully encased environments.

3.4.3.3.8 Radiation Embrittlement of Aluminum Material

Embrittlement of metals may occur under exposure to neutron radiation. Depending on the neutron fluence, radiation can cause changes in mechanical properties, such as loss of ductility, reduced fracture toughness, and decreased resistance to cracking.

Section 3.2.3.8 of NUREG-2214 [3-2] describes data showing that the mechanical properties of aluminum are not degraded at neutron fluence levels on the order of 10^{20} n/cm^2. Section 3.2.1.9 of NUREG-2214 [3-2] describes a bounding calculation that showed estimated fluence level within a dry storage cask system is four orders of magnitude below the levels reported to degrade the mechanical properties of aluminum.

Therefore, radiation embrittlement of the aluminum material is not credible in air-out door, helium, and fully encased environments.

3.4.3.4 Aging Mechanism of Nickel Alloy Material

A review of Table 3-5 through Table 3-8 shows that the only subcomponent with material in the nickel alloy material group is a portion of the metallic O-rings (i.e., the liner) exposed to:

- Helium

The following aging mechanisms for nickel alloy material were evaluated to determine if they are credible in the environment that the O-ring liner, i.e., nickel alloy material, is exposed to:

- General corrosion

- Pitting and crevice corrosion
- MIC
- SCC
- Fatigue
- Radiation embrittlement
- Wear

3.4.3.4.1 General Corrosion of Nickel Alloy Material

There is very little residual water in the interspace of the metallic O-rings and in the over pressure monitoring gas.

Therefore, general corrosion of the nickel alloy material is not credible in a helium environment.

3.4.3.4.2 Pitting and Crevice Corrosion of Nickel Alloy Material

Pitting and crevice corrosion require the presence of aqueous electrolytes to form a corrosive environment. There is very little residual water in the interspace of the metallic O-rings and in the over pressure monitoring gas to generate aqueous electrolytes.

Therefore, pitting and crevice corrosion of the nickel alloy material is not credible in a helium environment.

3.4.3.4.3 Microbiologically Influenced Corrosion of Nickel Alloy Material

Active microbial metabolism requires water in the form of water vapor, condensation, or deliquescence, and available nutrients to support microbial activity. There is very little residual water in the interspace of the metallic O-rings and in the over pressure monitoring gas to support MIC.

Therefore, MIC of the nickel alloy material is not credible in a helium environment.

3.4.3.4.4 Stress Corrosion Cracking of Nickel Alloy Material

SCC is the cracking of a metal produced by the combined action of corrosion and tensile stress and is highly chemical-specific. There is very little residual water in the interspace of the metallic O-rings and in the over pressure monitoring gas to support a corrosive environment. In addition, the metallic seals are crushed during installation and, as such, there are no tensile stresses.

Therefore, SCC of the nickel alloy material is not credible in a helium environment.

3.4.3.4.5 Fatigue of Nickel Alloy Material

Fatigue is the progressive and localized structural damage that occurs when a material is subjected to cyclic loading. The only cyclic loading experienced by the TN-68 cask nickel alloy material is associated with thermal cycling due to ambient seasonal and daily temperature fluctuation. However, the metallic seals are crushed during installation and thermal cycling of the liner will not affect the ability of the aluminum jacket to fulfill the confinement function.

Therefore, fatigue of the nickel alloy material is not credible in a helium environment.

3.4.3.4.6 Radiation Embrittlement of Nickel Alloy Material

Embrittlement of metals may occur under exposure to neutron radiation. Depending on the neutron fluence, radiation can cause changes in mechanical properties, such as loss of ductility, reduced fracture toughness, and decreased resistance to cracking. The function of the seal nickel alloy liner and spring is to maintain the shape of the seal prior to installation. During installation, the seals are crushed including the liner and spring. Therefore, radiation embrittlement of the seal liner and spring will not affect the ability of the aluminum jacket to fulfill the confinement function.

Therefore, radiation embrittlement of the nickel alloy material is not credible in a helium environment.

3.4.3.4.7 Wear of Nickel Alloy Material

Rolling contact wear results from the repeated mechanical stressing of the surface of a body rolling on another body.

There are no nickel alloy bodies rolling on another body while TN-68 casks are in storage.

Therefore, wear of the nickel alloy material is not credible in a helium environment.

3.4.3.5 Aging Mechanism of Polymer Material

A review of Table 3-5 through Table 3-8 shows that the environment that the polymer subcomponents are exposed to is:

- Fully Encased

The following aging mechanisms for polymer material were evaluated to determine if they are credible in the environment that polymer material is exposed to:

- Boron depletion
- Thermal aging
- Radiation embrittlement

3.4.3.5.1 Boron Depletion of Polymer Material

Table 5.1-1 of the UFSAR [3-3], lists the thickness of the radial neutron shield as 6 inches (15.24 cm) and Table 5.3-5 of the UFSAR [3-3] list the atom fractions of the materials used in the shielding models for the radial neutron shield along with the density of the radial neutron shield. The areal atom density of each element may be determined using the density of the neutron shield, the atom fractions, the molar mass and the thickness of the shield as follows:

- The relative mass of each element is determined by multiplying the atom fraction and the molar mass of the element then dividing by Avogadro's number ($6.0221*10^{23}$) particles per mole.
- The weight fraction of each element is determined by dividing the relative mass by the sum of all relative masses.
- The density of the element is determined by multiplying the weight fraction and the density of the radial neutron shield.
- Finally the areal density is determined by multiplying the density, Avogadro's number and the thickness of the radial neutron shield then dividing by the molar mass.

Element	Atom Fraction	Molar Mass (g/mole)	Relative Mass (grams)	Weight Fraction	Density	Areal Density (atoms/cm^2)
^{27}Al	0.10331	26.982	4.629	$2.851\text{x}10^{-1}$	$4.810\text{x}10^{-1}$	$1.636\text{x}10^{23}$
^{12}C	0.24658	12.000	4.914	$3.027\text{x}10^{-1}$	$5.106\text{x}10^{-1}$	$3.905\text{x}10^{23}$
^{16}O	0.21985	15.995	5.839	$3.596\text{x}10^{-1}$	$6.066\text{x}10^{-1}$	$3.480\text{x}10^{23}$
^{1}H	0.42207	1.008	$7.065\text{x}10^{-1}$	$4.352\text{x}10^{-3}$	$7.342\text{x}10^{-3}$	$6.555\text{x}10^{23}$
^{10}B	0.00164	10.013	$2.727\text{x}10^{-2}$	$1.680\text{x}10^{-3}$	$2.834\text{x}10^{-3}$	$2.598\text{x}10^{21}$
^{11}B	0.00655	11.009	$1.197\text{x}10^{-1}$	$7.373\text{x}10^{-3}$	$1.244\text{x}10^{-2}$	$1.037\text{x}10^{22}$
		Total	16.235			

The areal density of boron-10 ($2.598\text{x}10^{21}$ atom/cm^2) is on the order of the boron-10 areal density used in the evaluation of boron depletion performed in Section 3.4.1.1 of NUREG-2214 [3-2]. That section of NUREG-2214 concluded that with a typical neutron flux within a dry storage cask system, boron depletion would only be 0.0002 percent of the available boron-10 atoms.

Therefore, boron depletion of the polymer material is not credible in a fully encased environment.

3.4.3.5.2 Thermal Aging of Polymer Material

Polymers may be susceptible to heat-induced changes to material properties and configuration due to a number of mechanisms. At elevated temperatures, the long chain backbone of a polymer can undergo molecular scission (breaking) and cross linking. Also, gaseous products may be formed, including H_2, CH_4, and CO_2. These reactions may cause embrittlement, shrinkage/cracking, decomposition, and changes in physical configuration (e.g., loss of hydrogen or water).

In the TN-68 casks, the polymer materials are fully encased and have no structural integrity function. Therefore, embrittlement of the material is not an aging effect that needs to be managed.

Appendix 9A of the UFSAR [3-3], describes the thermal stability of the TN-68 cask radial neutron shield material. Interpolation of test data indicates an exponential weight loss that rapidly approaches a maximum weight loss value of 1.3%. This maximum weight loss, i.e., off gassing or change in physical configuration, would occur during the initial license period and, thus, not be affected by the PEO. Therefore, decomposition, changes in physical configuration, and the buildup of flammable gas due to thermal aging in the polymers are not aging effects that must be managed during the PEO.

Shrinkage/cracking due to thermal aging could locally displace shielding material and potentially diminish shielding effectiveness. Although this effect would most likely occur during the initial license period, it should be managed during the PEO.

Therefore, thermal aging of the polymer material is considered credible in a fully encased environment.

3.4.3.5.3 Radiation Embrittlement of Polymer Material

Radiation can alter polymer structures by molecular scission and cross-linking to reduce ductility, fracture toughness, and resistance to cracking.

The aging management review of radiation embrittlement of the polymer material is the same as for thermal aging of the polymer material. Therefore, embrittlement of the material, decomposition, changes in physical configuration, and the buildup of flammable gas due to radiation of the polymers are not aging effects that must be managed during the PEO. However, shrinkage/cracking is an aging effect that should be managed during the PEOs.

Therefore, radiation embrittlement of the polymer material is considered credible in a fully encased environment.

3.4.3.6 Aging Mechanism of Borated Aluminum Material

A review of Table 3-5 through Table 3-8 shows that the environment that the borated aluminum subcomponents are exposed to is:

- Helium

The following aging mechanisms for borated aluminum material were evaluated to determine if they are credible in the environment that borated aluminum is exposed to:

- General corrosion
- Galvanic corrosion
- Boron depletion
- Creep
- Thermal aging
- Radiation embrittlement
- Wet corrosion and blistering

3.4.3.6.1 General Corrosion of Borated Aluminum Material

Because aluminum is present as a continuous matrix (borated aluminum and aluminum metal-matrix composites) or used as an outer cladding (Boral®), the degree of general corrosion is considered to be largely governed by the corrosion of aluminum.

As discussed in Section 3.4.3.3.1, in the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, the corrosion reaction with aluminum will be limited.

Therefore, general corrosion of the borated aluminum material is not credible in a helium environment.

3.4.3.6.2 Galvanic Corrosion of Borated Aluminum Material

As discussed in Section 3.4.3.1.3, galvanic corrosion occurs when two dissimilar metals or conductive materials are in physical contact in the presence of a conducting solution. The aluminum-based neutron poison materials used inside dry storage cask systems can be in galvanic contact with the stainless steel fuel compartments, where aluminum is less noble.

In the helium environment, there is very little residual water in internal environments of a TN-68 cask following drying and refilling with inert helium gas and, thus, very few aqueous electrolytes.

Therefore, galvanic corrosion of the borated aluminum material is not credible in a helium environment.

3.4.3.6.3 Boron Depletion of Borated Aluminum Material

Boron depletion refers to the loss of the capability of a material to absorb neutrons when the neutron fluence significantly consumes boron-10 atoms. At typical levels of neutron flux and boron-10 concentration, the neutron dose after 60 years would deplete at most 0.0002 percent of the available boron-10 atoms. Using the highest expected neutron flux and the lowest boron-10 concentration as a worst-case scenario, only 0.02 percent of the available boron-10 atoms would be depleted after 60 years, which is too small to challenge the criticality control function of the neutron poisons.

Therefore, boron depletion of the borated aluminum material is not credible in a helium environment.

3.4.3.6.4 Creep of Borated Aluminum Material

Creep is the time-dependent inelastic deformation that takes place at an elevated temperature and a constant stress. Because the deformation processes that produce creep are thermally-activated, the rate of this time-dependent deformation is a strong function of the temperature. Because aluminum is present as a continuous matrix or as an external cladding in the neutron poison plates, and aluminum has a lower melting point than the other portions of the material microstructures, the creep behavior of poison materials is considered to be governed by the behavior of aluminum.

As discussed in Section 3.4.3.3.5, as a general rule of thumb, at temperatures below $0.4T_m$, where T_m is the melting point of the metal in Kelvin, thermal activation is insufficient to produce creep that could compromise the intended functions of SSCs. Applying the $0.4T_m$ rule, the critical creep temperature for aluminum is 212 °F. As shown in Table 3-3, the maximum basket temperature was determined to be 624 °F at the beginning of storage. It is apparent from this temperature that the borated aluminum plates could be exposed to temperatures above the minimum creep temperatures for aluminum.

Because temperatures within the TN-68 cask have the potential to exceed the minimum creep temperature of aluminum, it is necessary to consider the load applied to the subcomponent to determine whether significant creep deformation will occur, as well as the specific application, to determine whether the creep affects safety. As shown in Table 3-5 through Table 3-8, the borated aluminum subcomponents do not serve a structural function and are not expected to be under loads other than their own weight. In addition, the borated aluminum plates are supported by adjacent subcomponents. Because of the minimal applied loads and presence of adjacent supporting structures, the impact of creep on the intended safety functions function of the borated aluminum subcomponents is not considered credible.

Therefore, creep of the borated aluminum material is not credible in a helium environment.

3.4.3.6.5 Thermal Aging of Borated Aluminum Material

Prolonged exposure to elevated temperatures can lead to a loss of fracture toughness and ductility in some materials as a result of changes to their microstructure. Testing of aluminum-based neutron poison plates, however, has shown that these materials typically increase in ductility when they are aged at high temperatures. Qualification tests performed on neutron poisons demonstrate that microstructural changes induced by aging typically make the aluminum softer and more ductile as it is annealed, while the boron and carbide particulates are thermally stable at TN-68 cask internal temperatures.

Also, as discussed in Section 3.4.3.6.4 above, the borated aluminum subcomponents do not serve a structural function and are supported by adjacent subcomponents. Therefore, decreases in strength due to thermal aging are not expected to affect the intended safety functions of the borated aluminum subcomponents.

Therefore, thermal aging of the borated aluminum material is not credible in a helium environment.

3.4.3.6.6 Radiation Embrittlement of Borated Aluminum Material

Embrittlement of metals may occur under exposure to neutron radiation. Depending on the neutron fluence, radiation can cause changes in mechanical properties, such as loss of ductility, reduced fracture toughness, and decreased resistance to cracking.

Section 3.4.2.7 of NUREG-2214 [3-2] describes data showing that the mechanical properties of aluminum are not degraded at neutron fluence levels on the order of 10^{20} neutrons/square centimeter (n/cm^2). Section 3.2.1.9 of NUREG-2214 describes a bounding calculation that showed estimated fluence level within a dry storage cask system is four orders of magnitude below the levels reported to degrade the mechanical properties of aluminum-based neutron poison.

Therefore, radiation embrittlement of the borated aluminum material is not credible in a helium environment.

3.4.3.6.7 Wet Corrosion and Blistering of Borated Aluminum Material

The core of aluminum-boron carbide laminate composites is not fully sintered and, as a result, can have a porosity of 1 to 8 percent with varying degrees of interconnectivity among pores. This may allow water ingress into the core, where the water can react with the aluminum to form aluminum oxide and hydrogen gas. Blistering has been experimentally observed in the Boral® cladding during repeated wetting and drying representative of dry storage operations. Tests simulating the wetting and vacuum drying cycles during canister closure operations show that Boral® can form blisters in the aluminum cladding because of water ingress through its exposed edges. The blisters are characterized by a local area where the aluminum cladding separates from the underlying boron carbide-aluminum core, and the cladding is physically deformed outward.

There is no OE recorded in the United States, however, on the blistering of aluminum-boron carbide laminate composites used in loaded dry storage systems (DSSs). Although indications of wet corrosion have been observed during DSS loadings (via indirect measurements of generated hydrogen), this aging mechanism has not been observed to result in blistering or reduce the neutron-absorbing capability of Boral® in loaded DSSs. Further, because only a trace amount of water will be left in a dry storage cask after dehydration and helium backfill, wet corrosion is not expected to compromise the intended function of aluminum-boron carbide laminate composites during the PEO.

Therefore, wet corrosion and blistering of the borated aluminum material is not credible in a helium environment.

3.4.3.7 Aging Mechanism of Concrete Material

A review of Table 3-5 through Table 3-8 shows that the environments that the concrete subcomponents are exposed to are:

- Air-outdoor
- Groundwater/soil

The following aging mechanisms for concrete material were evaluated to determine if they are credible in the environments that concrete is exposed to:

- Freeze-thaw
- Creep
- Reaction with aggregates
- Differential settlement
- Aggressive chemical attack
- Corrosion of reinforcing steel
- Shrinkage
- Leaching of calcium hydroxide
- Radiation damage
- Fatigue
- Dehydration at high temperature
- Microbiological degradation
- Delayed ettringite formation
- Salt scaling
- Hardening

3.4.3.7.1 Freeze-Thaw of Concrete Material

Concretes that are nearly or fully saturated with water can be damaged by repeated freezing and thawing cycles. The degradation mode would initiate at the outer concrete surface exposed to outdoor environments, primarily at horizontal surfaces where water ponding can occur. For below-grade concrete structures, water that resides in soil can also be subject to freezing conditions, potentially promoting freeze-thaw damage. Because water expands when freezing, fully or mostly saturated concrete will experience internal stresses from the expanding ice, which can cause concrete cracking or scaling when pressures exceed the concrete tensile strength.

Therefore, freeze-thaw of concrete material is considered credible in air-outdoor or groundwater/soil environments.

3.4.3.7.2 Creep of Concrete Material

Creep in concrete is the time-dependent deformation resulting from sustained loads. While there are several factors that affect creep in concrete, the most important parameter controlling creep is sustained loading. The creep rate in concrete decreases exponentially with time, i.e., creep would be more of a concern during the initial license period rather than the PEO. In addition, the creep in concrete is largely mitigated by proper design practices. Furthermore, creep-induced concrete cracks are not generally large enough to reduce the compressive strength of concrete, cause deterioration of concrete, or cause exposure of reinforcing steel to the environment.

Therefore, creep of concrete material is not credible in air-outdoor or groundwater/soil environments.

3.4.3.7.3 Reaction with Aggregates of Concrete Material

The two most common alkali-aggregate reactions are alkali-silica reaction (ASR) and alkali-carbonate reaction, with ASR being the most common and damaging. ASR is a chemical reaction between hydroxyl ions (present in the alkaline cement pore solution) and reactive forms of silica present in some aggregates. ASR damage in the concrete manifests itself as a characteristic map cracking on the concrete surface. The internal damage results in the degradation of concrete mechanical properties and, in severe cases, the expansion can result in undesirable dimensional changes and popouts. In general, ASR is a slow degradation mechanism that can cause serviceability issues and may exacerbate other deterioration mechanisms. The requisite conditions for initiation and propagation of ASR include 1) a sufficiently high alkali content of the cement (or alkali from other sources, such as deicing salts, seawater, and groundwater), 2) a reactive aggregate, and 3) available moisture, generally accepted to be relative humidity greater than 80 percent.

ASR may take from three to more than 25 years to develop in concrete structures, depending on the nature (reactivity level) of the aggregates, the moisture and temperature conditions to which the structures are exposed, and the concrete alkali content. The delay in exhibiting deterioration indicates that there may be less reactive forms of silica that can eventually cause deterioration. Recent OE has revealed degradation of the concrete at a nuclear power plant as a result of ASR even though the concrete used at the plant passed all industry standard ASR screening tests at the time of construction. In addition, ASR screening tests are not conducted on each aggregate source but rather in select batches.

Because of the uncertainties in screening tests that can effectively be used to eliminate the potential for ASR and previous ASR OE at a nuclear facility, the aging mechanism is considered credible in concrete exposed to any environment with available moisture.

Therefore, reaction with aggregates of concrete material is considered credible in air-outdoor or groundwater/soil environments.

3.4.3.7.4 Differential Settlement of Concrete Material

Differential settlement is a result of the uneven deformation of the supporting foundation soil. Differential settlement, which causes distortion (loss of form) and damage (cracking) to concrete structures, is a function of the uniformity of the soil, stiffness of the structure, stiffness of the soil, and distribution of loads within the structure. The settlement of saturated cohesive soil consists of three components: 1) immediate settlement occurring due to the applied load, 2) consolidation settlement occurring gradually due to dissipation of the excess pore pressures generated by the applied load, and 3) secondary compression that depends on the composition and structure of the soil skeleton. The settlement of course-grained granular soils subject to applied load occurs immediately, primarily from the compression of the soil skeleton due to rearrangement of particles. However, most settlement issues involving a combination of immediate settlement and progressing long-term settlement are typically discovered in less than one year of construction. However, OE has shown that it can occur. Therefore, differential settlement of concrete exposed to groundwater or soil (below-grade) environment is considered credible.

Although portions of concrete structures in contact with air can exhibit the effects of settlement, the direct interaction with the underlying soil is the mechanism that causes the aging. As a result, differential settlement of concrete exposed to an outdoor air environment is not considered credible.

Therefore, differential settlement of concrete material is considered credible in a groundwater/soil environment, but not credible in an air-outdoor environment.

3.4.3.7.5 Aggressive Chemical Attack of Concrete Material

The intrusion of aggressive ions or acids into the pore network of the concrete can cause various degradation phenomena. The aggressive chemical attack typically originates from an external source of sulfate or magnesium ions as well as acidic environmental conditions.

In an air-outdoor or groundwater/soil environment, groundwater, seawater, and rainwater may contain sulfate species that penetrate the concrete and chemically react with alkali and calcium ions to form a precipitate of calcium sulfate in addition to other forms of calcium and sulfate-based compounds. The manifestation of sulfate attack is cracking, increase in concrete porosity and permeability, loss of strength, and surface scaling generated by the expansion associated with the formation of ettringite within the concrete and the pressure generated by the precipitated calcium and sulfate-base compounds inside the concrete pore network. Acids from groundwater and acid rain can dissolve both hydrated and unhydrated cement compounds (e.g., calcium hydroxide, calcium silicate hydrates, and calcium aluminate hydrates) as well as calcareous aggregate in concrete without any significant expansion reaction. In most cases, the chemical reaction forms water-soluble calcium compounds, which are then leached away by aqueous solutions. The dissolution of concrete commences at the surface and propagates inward as the concrete degrades. The signs of acidic attack are loss of alkalinity (also disturbing of electrochemical passive conditions for the embedded steel reinforcement), loss of material (i.e., concrete cover), and loss of strength.

NUREG-1801, [3-4], states that continued or frequent cyclic exposure to the following aggressive chemical environments is necessary to cause significant aggressive chemical attack degradation:

- Acidic solutions with pH < 5.5
- Chloride solutions > 500 ppm
- Sulfate solutions > 1500 ppm

Since the groundwater/soil and air-outdoor may contain solutions that exceed these criteria, aggressive chemical attack is considered possible.

Therefore, aggressive chemical attack of concrete material is considered credible in air-outdoor or groundwater/soil environments.

3.4.3.7.6 Corrosion of Reinforcing Steel of Concrete Material

Corrosion of the reinforcing steel embedded in the concrete is mainly caused by the presence of chloride ions in the concrete pore solution and carbonation of the concrete. The highly alkaline environment provided by the concrete (normally with porewater pH > 13.0) results in the formation of a metal-adherent oxide film on the reinforcement steel bar surface, which passivates the steel. However, chloride ions may penetrate the concrete matrix and break down the steel passive layer, once the chloride concentration at the reinforcing steel surface exceeds a threshold value, triggering corrosion of the reinforcing steel and shortening the service life of a concrete structure. Concrete durability is directly related to the quality of the concrete, the external concentration of chlorides on the concrete surface, and the reinforcement material. The service life of concretes exposed to chloride attack depends on the concrete cover, the surface chloride concentration, the chloride diffusion coefficient, the type of cementitious material, and the reinforcing steel material.

In an air-outdoor or groundwater/soil environment, chloride ions may penetrate the concrete from the outside environment, such as when using deicing salts, aggressive groundwater, or marine environments. Although no cases of corrosion-induced damage have been reported, the results of a durability model show that corrosion of the reinforcing steel in concrete can potentially initiate and propagate within the 60-year timeframe for concretes of moderate to low quality.

Therefore, corrosion of reinforcing steel of concrete material is considered credible in air-outdoor or groundwater/soil environments.

3.4.3.7.7 Shrinkage of Concrete Material

Shrinkage occurs when hardened concrete dries from a saturated condition to a state of equilibrium in about 50 percent relative humidity. As excess concrete water evaporates, tensile stresses are induced in the concrete due to internal pressure from the capillary action of water movement, which results in cracking. The factors affecting shrinkage are cement content, water-to-cement ratio, degree of hydration, elastic modulus of aggregates, amount and characteristics of concrete admixtures, temperature and humidity during curing, and size and shape of concrete.

Shrinkage of concrete occurs initially during curing, which can be controlled through concrete formulation and the density and distribution of internal reinforcement. Over 90 percent of the shrinkage occurs during the first year, reaching 98 percent by the end of the first five years. Thus, shrinkage is not expected to influence concrete performance during the PEO, because most of the shrinkage will take place early on in the life of the concrete. As a result, shrinkage of concretes is not considered credible.

Therefore, shrinkage of concrete material is not credible in air-outdoor or groundwater/soil environments.

3.4.3.7.8 Leaching of Calcium Hydroxide of Concrete Material

A constant or intermittent flux of water over a concrete surface can result in the removal or leaching of calcium hydroxide. Calcium hydroxide leaching is observed in the form of white leachate deposits (calcium carbonate) on the concrete surface. The extent of the leaching depends on the environmental salt content and temperature, and it can take place above and below ground.

In an air-outdoor or groundwater/soil environment, the concrete surface may be exposed to intermittent fluxes of water, thus making leaching of calcium hydroxide credible.

Therefore, leaching of calcium hydroxide of concrete material is considered credible in air-outdoor or groundwater/soil environments.

3.4.3.7.9 Radiation Damage of Concrete Material

Radiation effects on concrete properties will depend on the gamma and neutron radiation doses, temperature, and exposure period. Gamma radiation can decompose and evaporate water in concrete. Because most of the water is contained in the cement paste, the effect of gamma radiation on cement paste is more significant than on the aggregates. Gamma radiation can also decompose the Si-O bond within calcium silicate hydrate. Neutron radiation deteriorates concrete by reducing stiffness, forming cracks by swelling, and changing the microstructure of the aggregates.

Section 3.5.1.9 of NUREG-2214 [3-2] describes data showing that the mechanical properties of concrete are not degraded at neutron fluence levels on the order of 10^{19} neutrons/square centimeter (n/cm^2) and 10^{10} rad for gamma rays. Section 3.2.1.9 of NUREG-2214 describes a bounding calculation that showed estimated neutron fluence level within a dry storage cask system is three orders of magnitude below the levels reported to degrade the mechanical properties of concrete. The gamma dose is also expected to be several orders of magnitude less than the 10^{10} rad limit.

Therefore, radiation damage of concrete material is not credible in air-outdoor or groundwater/soil environment.

3.4.3.7.10 Fatigue of Concrete Material

Concrete fatigue strength is defined as the maximum stress that the concrete can sustain without failure under a given number of stress cycles. Because the storage pad is a static application, mechanical cyclic loading is not expected. However, restraint of the concrete from expanding and contracting as it is exposed to rapid changes in temperature will lead to internal stresses in the structure. If the changes in temperature are severe and the resulting strains are sufficient, local plastic deformation can occur. Repeated application of this thermal loading can lead to crack initiation and propagation in low-cycle fatigue. Concrete fatigue in the reinforced concrete storage pad may be caused by diurnal and seasonal temperature gradients.

Section 3.5.1.10 of NUREG-2214 [3-2] discusses a generic evaluation of concrete fatigue over 60 years of storage considering an extreme seasonal ambient temperature variation from -40 °F to 125 °F. This range bounds the design temperature range for the TN-68 dry storage cask system of -20 °F to 100 °F. The evaluation concluded that fatigue of concrete exposed to sheltered, outdoor, groundwater or soil (below-grade), and fully encased environments is not considered credible.

Therefore, fatigue of concrete material is not credible in air-outdoor or groundwater/soil environments.

3.4.3.7.11 Dehydration at High Temperature of Concrete Material

Exposure of concrete to elevated temperatures can affect its mechanical and physical properties. It is well known that concretes can degrade at high temperatures due to dehydration of the hydrated cement paste, thermal incompatibility between the cement and aggregates, and physicochemical deterioration of the aggregates.

The effects of thermal dehydration were addressed during the initial licensing of the TN-68 dry storage cask system. Since the fuel temperature decreases over time, the concrete temperature will also decrease over time. Therefore, the design temperature considerations during the initial licensing are expected to remain adequate during the PEO.

Therefore, dehydration at high temperature of concrete material is not credible in air-outdoor or groundwater/soil environments.

3.4.3.7.12 Microbiological Degradation of Concrete Material

Biodeterioration is caused by colonization of microbes and microorganisms that grow on concrete surfaces that offer favorable environmental conditions (e.g., available moisture, near neutral pH, presence of nutrients). Conducive environments may have elevated relative humidity (i.e., greater than about 60 percent), long cycles of humidification and drying, freezing and thawing, high carbon dioxide concentrations, high concentrations of chloride ions or other salts, or high concentrations of sulfates and small amounts of acids. Biodeterioration may lead to reduction of the protective cover depth and increase both concrete porosity and the transport of aggressive chemicals. In addition, this degradation mode can promote a reduction in concrete pH, loss of concrete strength, and spalling/scaling. Evidence shows that a wide variety of organisms can cause concrete deterioration in polluted soils and groundwater. The biodeterioration of concrete typically is confined to the surface. The rate of deterioration is slow, but the degradation mode has been observed within 40 years of exposure.

Although no cases of microbiological degradation of concrete have been reported in nuclear applications, the degradation mode is considered credible, as below-grade environments may be conducive to microbe and bacteria growth.

In air-outdoor environments, favorable conditions for microbiological degradation mechanisms may exist because of the potential presence of moisture. However, the conditions would be intermittent, and there is no evidence that actual concrete subcomponents in these environments microbiologically degrade.

Therefore, microbiological degradation of concrete material is considered credible in a groundwater/soil environment, but is not credible in an air-outdoor environment.

3.4.3.7.13 Delayed Ettringite Formation of Concrete Material

At the initial stage of fresh concrete curing, ettringite, commonly referred to as "naturally occurring ettringite," is formed by the reaction of tricalcium aluminate and gypsum in the presence of water. The formation of naturally occurring ettringite in fresh concrete is not detrimental to the overall concrete performance. At the still-early stage of concrete curing, the naturally occurring ettringite may convert to monosulfoaluminate if curing temperatures are greater than about 70 °C (158 °F). After concrete hardens, if the temperature decreases below this value, the monosulfoaluminate becomes unstable and, in the presence of sulfates released by the calcium-silicate-hydrate gel, ettringite will reform. This mechanism is called "delayed ettringite formation" (DEF), which results in volume expansion and increased internal pressures in the concrete.

The conditions necessary for the occurrence of DEF are excessive temperatures during concrete placement and curing, the presence of internal sulfates, and a moist environment. American Concrete Institute (ACI) 318-05 indicates that inspection reports shall document concrete temperature and protection during placement when the ambient temperature is above 35 °C (95 °F). Protection measures during concrete placement include lowering the temperature of the batch water, cement, and aggregates as referenced in ACI 305R-10. As such, following the ACI 318-05, ACI 305R-10, and ACI 308R-01 guidelines during concrete placement and curing can effectively limit the concrete temperature to below 70 °C (158 °F), therefore preventing the development of DEF.

As noted in Section 3.3.2, the storage pad is designed and constructed in accordance with codes and standards set by the general licensee. Construction practices for the concrete storage pad may vary from ISFSI to ISFSI. Therefore, DEF may be an applicable concrete aging mechanism for the storage pad, unless ruled out by the general licensee based on an ISFSI-specific evaluation.

Therefore, delayed ettringite formation of the storage pad concrete material is considered credible in air-outdoor or groundwater/soil environments.

3.4.3.7.14 Salt Scaling of Concrete Material

Salt scaling is defined as superficial damage caused by freezing a saline solution on the surface of a concrete body. The damage is progressive and consists of the removal of small chips or flakes of material. Similar to freeze-thaw damage, salt scaling takes place when concrete is exposed to freezing temperatures, moisture, and dissolved salts. The degradation is maximized at a moderate concentration of salt (e.g., from deicing salts).

Similar to freeze-thaw damage, the degradation initiates at the outer concrete surface exposed to outdoor environments, primarily at horizontal surfaces where water ponding can occur. For below-grade concrete structures, water that resides in soil may contain salt and, thus, the concrete may be susceptible to salt scaling.

Therefore, salt scaling of concrete material is considered credible in air-outdoor or groundwater/soil environments.

3.4.3.7.15 Hardening of Concrete Material

Concrete compressive strength is known to increase with age. As the compressive strength increases, the concrete hardens also increases causing the applicable deceleration g-loads during cask drops and tip-over accidents to increase. This could potentially affect the structural capability of the TN-68 casks.

UFSAR [3-3] Section 3D.3.2 describes the inputs used for the concrete material (i.e., material law 16 in LS-DYNA) in the cask tip over analysis, per the design basis calculation, these inputs represent 4,200 psi compressive strength concrete. UFSAR Section 3D.7.2 lists the concrete strength used in the cask end drop analysis as 6,000 psi.

The data in Table 2.13 of NUREG/CR-6424 [3-8], shows the time-dependent compressive strength values for various Portland cement concretes at 28 days, 20 years, and 30 years. The table shows that concrete with a higher initial compressive strength will realize a smaller percentage increase in strength over time, versus concrete with a lower initial compressive strength. For example, concrete with a 28-day compressive strength 19.2 MPa will increase 87% during the first 20 years and additional 10% during the next 10 years. While concrete with a 28-day compressive strength of 33.6MPa will increase only 41% during the first 20 years and a negligible amount beyond 20 years. Since the minimum design basis concrete compressive strength for the CoC 1027 concrete storage pads is 4,200 psi (28.9 Mpa), it is reasonable to conclude that any increase in the concrete hardness will occur during the first 20 years of storage and not during the PEO.

Therefore, hardening of concrete material is not credible in air-outdoor or groundwater/soil environments.

3.4.3.8 Aging Mechanism of Spent Fuel Assembly Cladding

A review of Table 3-5 through Table 3-8 shows that the environment that the spent fuel assembly (SFA) cladding is exposed to is:

- Helium

The following aging mechanisms for SFA cladding were evaluated to determine if they are credible in the environment that SFA cladding is exposed to:

- Hydride-induced embrittlement
- Delayed hydride cracking
- Thermal creep
- Low temperature creep
- Mechanical overload
- Oxidation
- Pitting corrosion
- Galvanic corrosion
- SCC
- Radiation embrittlement
- Fatigue

3.4.3.8.1 Hydride-Induced Embrittlement of Spent Fuel Assembly Cladding

In reactor service, the zirconium-based fuel cladding absorbs hydrogen, which leads to the precipitation of hydride platelets as the dissolved hydrogen exceeds the solubility limit of the cladding. The primary source of the hydrogen is water-side corrosion (oxidation) of the cladding. The total concentration of hydrogen absorbed by the cladding (i.e., dissolved in the zirconium matrix and in precipitated hydrides) increases with burnup and varies axially across the fuel rods. When discharged from the reactor and during wet storage, the hydride platelets are mostly oriented in the circumferential-axial direction, with a smaller fraction oriented in the radial-axial direction.

During vacuum drying of the TN-68 casks, the temperature of the SFAs and the temperature-dependent solubility limit of hydrogen in the cladding will also increase. As a result, some of the hydrides present in the cladding will redissolve as hydrogen. Once the loaded cask is dried and backfilled, the cladding temperature will decrease over time and upon a sufficient temperature drop, some of the hydrogen in solution will reprecipitate as new hydrides. During this process, the orientation of these precipitated hydrides may change from the circumferential-axial to the radial-axial direction. The degree of reorientation is primarily driven by the metallurgical microstructure of the cladding alloy and the cladding hoop stresses during drying operations and subsequent cooling, which are determined by the rod internal pressure at a given gas temperature. Cladding with a high concentration of radial hydrides (determined by the dry storage cask system drying conditions) has been shown to have reduced ductility under pinch-load stresses at sufficiently low temperatures. Section 3.6.1.1 of NUREG-2214 [3-2] contains more discussion of the hydride reorientation phenomenon, including expected ranges of dissolved hydrogen, drying temperatures, solubility limits, hoop stresses, and experimental studies.

Considering the hydrogen content, peak drying temperatures, and corresponding hoop stresses, hydride reorientation in zirconium-based cladding is only credible for high burnup (HBU) fuel during the 60-year service timeframe. Furthermore, depending on the specific fuel contents, it is possible for some of the cladding to reach temperatures near or below the ductile-to-brittle transition temperatures reported in the literature.

Therefore, degradation of mechanical properties during pinch-type stresses due to hydride-induced embrittlement is considered a credible aging mechanism for zirconium-based HBU (i.e., > 45 GWd/MTU) fuel claddings. Hydride-induced embrittlement is not credible for low burnup (i.e., ≤ 45 GWd/MTU) fuel.

3.4.3.8.2 Delayed Hydride Cracking of Spent Fuel Assembly Cladding

Delayed hydride cracking (DHC) is a time-dependent mechanism traditionally thought to occur by the diffusion of hydrogen to an incipient crack tip (notch, flaw) in the cladding, followed by nucleation, growth, and subsequent fracture of the precipitated hydrides at the crack tip. Hydrogen dissolved in the cladding (see Section 3.4.3.8.1) can diffuse up a stress gradient in the crystalline lattice, or into the stress field at the core of an edge dislocation. The concentration gradient established by the stress gradient may lead to hydrogen supersaturation (i.e., solubility limit being exceeded) leading to the precipitation of hydrides at the crack tip. The precipitated hydride will continue to grow by the dissolution of hydrides in the low-stress regions of the material and by the continued diffusion of hydrogen up the stress gradient. Once the hydride reaches a critical size, it will crack and propagate to the end of the hydride, where it will blunt. The cycle could then repeat, until the crack propagates through the thickness of the material.

Requisite conditions for DHC are the presence of: 1) hydrides, 2) existing crack tips (notch, flaws) that act as initiating sites, and 3) sufficient cladding hoop stresses. Section 3.6.1.2 of NUREG-2214 [3-2] discusses the availability of hydrides, initial depth of existing crack tips, and hoop stresses in zirconium-based fuel claddings. NUREG-2214 goes on to determine that the critical flaw size needed to initiate DHC is larger than the initial depth of potentially existing cracks.

Therefore, DHC is not credible for zirconium-based alloy SFA cladding.

3.4.3.8.3 Thermal Creep of Spent Fuel Assembly Cladding

Creep is the time-dependent deformation of a material under stress. Creep in zirconium-based cladding is caused by the hoop stresses from the rod internal pressure at a given fuel temperature. Therefore, the mechanism is expected to be self-limiting, due to the decreasing temperatures and creep-induced volume expansion, which results in lower internal rod pressures with time. Excessive creep of the cladding during dry storage could lead to thinning, hairline cracks, or gross ruptures. Section 3.6.1.3 of NUREG-2214 [3-2] contains an evaluation of cladding hoop stresses during dry storage for HBU fuel. The evaluation estimated the maximum cladding strain to be near 2.1 percent. The elastic strain limit for various zirconium-based cladding alloys with circumferential hydrides is less than 1 percent and is expected to be lower for cladding containing both circumferential and radial hydrides. Therefore, zirconium-based cladding in HBU fuel is expected to undergo creep strains during the 60-year timeframe.

Interim Staff Guidance (ISG) -11, Revision 3 [3-6], provides the basis for which thermal creep of low burnup (i.e., ≤ 45GWd/MTU) zirconium-based cladding is not a concern provided the peak normal cladding temperature is below 752 °F. Table 3-3 shows the maximum cladding temperature for normal condition for fuel stored in a TN-68 cask is 649°F. Therefore, creep of low burnup fuel stored in a TN-68 cask is not a concern.

Therefore, thermal creep is considered a credible aging mechanism for zirconium-based HBU (i.e., > 45 GWd/MTU) fuel claddings. Thermal creep is not credible for low burnup (i.e., ≤ 45 GWd/MTU) fuel with stainless steel or zirconium-based alloy cladding.

3.4.3.8.4 Low Temperature Creep of Spent Fuel Assembly Cladding

Low-temperature creep (also called “athermal creep”) may occur when sustained hoop stresses operate on the cladding material at or near ambient temperature. Various athermal creep mechanisms have been proposed at low stresses, although there is no evidence or literature information to support that these will be operational on zirconium-based alloys. However, the literature shows that low-temperature creep has been shown to occur in titanium and its alloys, which leads to deformation twinning. Since both titanium and zirconium have the same crystalline structure (hexagonal close packed crystalline), Section 3.6.1.4 of NUREG-2214 [3-2] performed an evaluation using the titanium threshold stress for low-temperature creep of 25 percent of the yield strength as the threshold stress for zirconium-based cladding.

The evaluation concluded that the room temperature hoop stresses on the zirconium-based cladding are expected to be less than 25 percent of the yield strength during the 60-year storage timeframe.

Therefore, low temperature creep is not credible for zirconium-based alloy SFA cladding.

3.4.3.8.5 Mechanical Overload of Spent Fuel Assembly Cladding

Mechanical overload is generally associated with pellet-to-cladding mechanical interaction (PCMI), which could compromise the cladding integrity during storage. PCMI is likely during reactor operations when the reactivity transient during a reactivity-initiated accident results in a rapid increase in a fuel rod power, leading to a nearly adiabatic heating of the fuel pellets and potential failure of the fuel cladding. Data generated in experimental reactors conducting ramp testing of heavily hydrided fuel claddings (i.e., associated with HBU fuel) indicate that hydride rims with large hydride number density at the cladding outer surface may lead to crack initiation. The cracks could propagate from the outside toward the inner cladding surface, potentially resulting in failures. During dry storage, PCMI stresses could develop due to pellet swelling and release of fission gases to the gap between the fuel and cladding.

Section 3.6.1.5 of NUREG-2214 [3-2] contains a generic evaluation showing that the strain rates during storage of HBU zirconium-based cladding are expected to be five to seven orders of magnitude lower than the threshold strain rate needed for PCMI-induced failures.

Due to the expected lower hydride concentration in low burnup fuel (≤ 45 GWd/MTU), PCMI-induced failures are not considered credible.

Therefore, mechanical overload is not credible for zirconium-based alloy SFA cladding.

3.4.3.8.6 Oxidation of Spent Fuel Assembly Cladding

There is very little residual water in the internal environments of a TN-68 cask following drying and refilling with inert helium gas, the oxidation of the external surface of the cladding will be limited. Similarly, there is no moisture internal to the fuel pin that could contribute to the oxidation of the cladding.

Therefore, oxidation is not credible for zirconium-based alloy SFA cladding.

3.4.3.8.7 Pitting Corrosion of Spent Fuel Assembly Cladding

Because there is very little residual water in the internal environments of a TN-68 cask following drying and refilling with inert helium gas, pitting corrosion of the external surface of the cladding will be limited. Similarly, there is no moisture internal to the fuel pin that could contribute to pitting corrosion of the cladding.

Therefore, pitting corrosion is not credible for zirconium-based alloy SFA cladding.

3.4.3.8.8 Galvanic Corrosion of Spent Fuel Assembly Cladding

Galvanic corrosion can occur due to a mismatch in corrosion potentials between two metals in an aqueous solution. In fuel assemblies, the mismatch can occur when the cladding is in contact with other metallic components, which could result in the formation of a galvanic cell, provided there is an aqueous solution between the two subcomponents.

Because there is very little residual water in the internal environments of a TN-68 cask following drying and refilling with inert helium gas, the oxidation of the external surface of the cladding will be limited. Similarly, there is no moisture internal to the fuel pin that could contribute to the corrosion of the cladding.

Therefore, galvanic corrosion is not credible for zirconium-based alloy SFA cladding.

3.4.3.8.9 Stress Corrosion Cracking of Spent Fuel Assembly Cladding

Stress corrosion cracking occurs as a result of a synergistic combination of a susceptible material, an aggressive environment, and sufficiently high tensile stress. The corrosive environment associated with SCC of fuel rods has been attributed to specific fission products, such as iodine, cesium, and cadmium, generated during reactor irradiation. Stress corrosion cracking of the cladding can occur at the inner surface of the rod where the fuel pellet and cladding mechanically interact, and is related to PCMI hoop stresses on the cladding.

Section 3.6.1.9 of NUREG-2214 [3-2] contains an evaluation showing that hoop stresses during a 60-year period of storage, including those due to PCMI, will remain below that needed for inducing SCC in zirconium-based cladding.

Therefore, SCC is not credible for zirconium-based alloy SFA cladding.

3.4.3.8.10 Radiation Embrittlement of Spent Fuel Assembly Cladding

Radiation embrittlement of cladding can result in degradation of the mechanical properties of the cladding, such as ductility and strength. Embrittlement is largely observed during reactor operation due to cumulative fast neutron fluence on the order of 10^{22} n/cm^2.

Section 3.2.1.9 of NUREG-2214 [3-2] describes a bounding calculation that showed estimated fluence after 100 years of storage, which is five to six orders of magnitude below the fluence seen during reactor operations. Therefore, 100 years of storage has a negligible contribution to the overall fast neutron fluence.

Therefore, radiation embrittlement is not credible for zirconium-based alloy SFA cladding.

3.4.3.8.11 Fatigue of Spent Fuel Assembly Cladding

Fatigue is the progressive and localized structural damage that occurs when a material is subjected to cyclic loading. The only cyclic loading experienced by the spent fuel cladding is associated with thermal cycling. The only source of potential thermal fatigue of the spent fuel cladding is ambient seasonal and daily temperature fluctuation. The spent fuel cladding does not experience the full amplitude of ambient temperature cycles, and a gradual, long-term temperature decrease occurs during the course of storage. The seasonal and daily variations in ambient conditions are ameliorated by the thermal mass of the TN-68 cask.

Section 3.6.1.11 of NUREG-2214 [3-2] contains a very conservative evaluation of the cyclic stresses in zirconium-based spent fuel cladding due to thermal cycles over a 60-year storage period. It concludes that the cumulative cyclic stresses are well below the threshold needed for fatigue-induced failure of the zirconium-based cladding.

Therefore, fatigue is not credible for zirconium-based alloy SFA cladding.

3.4.3.9 Aging Mechanism of Spent Fuel Assembly Hardware Materials

A review of Table 3-5 through Table 3-8 shows that the environment that the SFA hardware materials are exposed to is:

- Helium

The following aging mechanisms for SFA hardware materials were evaluated to determine if they are credible in the environment that SFA hardware materials are exposed to:

- Creep
- Hydriding
- General corrosion

- Stress corrosion cracking
- Radiation embrittlement
- Fatigue

3.4.3.9.1 Creep of Spent Fuel Assembly Hardware Materials

As discussed in Section 3.4.3.1.6, as a general rule of thumb, at temperatures below $0.4T_m$, where T_m is the melting point of the metal in Kelvin, thermal activation is insufficient to produce creep that could compromise the intended functions of SSCs. The melting temperature of various zirconium alloys, Inconel, and stainless steel alloys is above 3,272 °F, 2,300 °F, and 2,597 °F, respectively. Appling the $0.4T_m$ criterion yields a creep threshold of 1,033 °F for zirconium alloys, 644 °F for Inconel alloys, and 763 °F for stainless steel.

The highest temperatures within the TN-68 dry storage cask system are at locations close to the fuel rods. As shown in Table 3-3 the maximum fuel cladding temperature was determined to be 649 °F. Because the fuel rods are the only heat source within the system, they provide upper temperature limits for all subcomponents regardless of their environment. It is apparent from the temperatures in Table 3-3 that SFA hardware made of zirconium alloys and stainless steel will not approach the minimum 1,033 °F and 763 °F creep threshold temperatures, respectively. While the maximum cladding temperature of 649 °F is above the 644 °F Inconel alloy threshold temperature, it will decrease during the initial storage period such that it will be below the Inconel threshold temperature during the PEO.

Therefore, creep is not credible for the SFA hardware materials.

3.4.3.9.2 Hydriding of Spent Fuel Assembly Hardware Materials

Hydriding may occur in zirconium alloys that experience hydrogen pickup in reactor service. As the temperature of the assembly hardware decreases, zirconium hydrides precipitate due to the decreasing hydrogen solubility in the zirconium matrix. The hydride precipitation will occur when the hardware cools in the spent fuel pools after reactor discharge. Some of the hydride will dissolve during the drying process and will reprecipitate due to subsequent cooling during storage. Unlike the fuel rod cladding, there is no hoop stress for the zirconium-based assembly hardware to cause hydride reorientation. Any load on the assembly hardware is predominantly due to its own weight, which is not sufficient to cause hydride reorientation. Because there is limited load during storage on assembly hardware, it is unlikely that hydriding will affect the ability of the assembly hardware to ensure that the spent fuel remains in the as-analyzed configuration.

Hydride formation is not a concern in Inconel and stainless steel materials, thus hydriding is not credible in stainless steel spent fuel assembly hardware.

Therefore, hydriding is not credible for the SFA hardware materials.

3.4.3.9.3 General Corrosion of Spent Fuel Assembly Hardware Materials

There is very little residual water in the internal environments of a TN-68 cask following drying and refilling with inert helium gas, thus the general corrosion of the SFA hardware will be limited.

Therefore, general corrosion is not credible for the SFA hardware materials.

3.4.3.9.4 Stress Corrosion Cracking of Spent Fuel Assembly Hardware Materials

SCC of SFA hardware made of zirconium alloys is not considered credible based on the discussion in Section 3.4.3.8.9.

Various stainless steel and Inconel assembly hardware components could be susceptible to SCC in the presence of an aggressive environment and sufficient residual tensile stresses. Residual tensile stresses are expected to be present in the assembly hardware, primarily in welded areas. Regarding the chemical environment, the SFA hardware is exposed to the inert helium environment within the TN-68 cask. In addition there is very little residual water in the internal environments of a TN-68 cask following drying and refilling with inert helium gas. Because of the lack of halides and the small amount of water in the helium environments, SCC of stainless steel and Inconel hardware components is not considered credible.

Therefore, SCC is not credible for the SFA hardware materials.

3.4.3.9.5 Radiation Embrittlement of Spent Fuel Assembly Hardware Materials

Radiation embrittlement of SFA hardware made from zirconium alloys is excluded using the basis provided in Section 3.4.3.8.10. This basis (i.e., the fluence seen during dry storage is several orders of magnitude below that seen during reactor operation) is also applicable to SFA hardware made of Inconel. Similarly, radiation embrittlement of assembly hardware made of stainless steel is not considered credible per the technical bases provided in Section 3.4.3.2.9.

Therefore, radiation embrittlement is not credible for the SFA hardware materials.

3.4.3.9.6 Fatigue of Spent Fuel Assembly Hardware Materials

Fatigue of SFA hardware made from zirconium alloys is excluded using the basis provided in Section 3.4.3.8.11. Fatigue of assembly hardware made of stainless steel is not considered credible per the technical basis provided in Section 3.4.3.2.7. Similarly rational (as used for zirconium and stainless steel materials) may also be used to show that fatigue of assembly hardware made of Inconel is not considered credible.

Therefore, fatigue is not credible for the SFA hardware materials.

3.4.3.10 Summary of Aging Mechanism of Materials

Table 3-4 provides a summary of the aging mechanisms for materials used in the TN-68 dry storage cask system. The table also lists the environments where the aging mechanism could manifest into an aging effect that affects an important-to-safety function, i.e., is considered credible. Table 3-4 also identifies the aging effect into which each credible aging mechanism could manifest, (based on Table 2-4 of NUREG-2214 [3-2]).

3.5 Aging Management Review for TN-68 Cask

Table 3-5 provides the detailed results of the AMR for the subcomponent parts of the TN-68 cask. For each material group/environment combination and each part, the table lists the credible aging mechanisms and effects based on the information in Table 3-4. A summary of the aging effects that require management for the TN-68 cask is provided in Table 3-6.

The following aging effects/mechanisms will be managed via the TN-68 AMP:

- Steel
  - Loss of material due to general, pitting, crevice, and galvanic corrosion
  - Loss of preload due to stress relaxation of bolts
- Stainless steel
  - Loss of material due to pitting, crevice, and galvanic corrosion
- Polymers
  - Shrinkage/cracking due to thermal aging and radiation embrittlement
- Aluminum
  - Loss of material due to general, pitting, crevice, and galvanic corrosion

3.6 Aging Management Review for Storage Pad

Table 3-7 provides the detailed results of the AMR for the subcomponent parts of the storage pad. For each material group/environment combination and each part, the table lists the credible aging mechanisms and effects based on the information in Table 3-4.

The following aging effects/mechanisms will be managed via the storage pad AMP:

- Steel
  - Loss of material due to general, pitting, and crevice corrosion
  - Loss of material due to microbiological influenced corrosion
- Concrete
  - Loss of material due to freeze-thaw, aggressive chemical attack, corrosion of reinforcing steel, microbiological degradation, delayed ettringite formation [1], and salt scaling
  - Cracking due to freeze-thaw, reaction with aggregates, differential settlement, aggressive chemical attack, corrosion of reinforcing steel, and delayed ettringite formation [1]
  - Loss of strength due to reaction with aggregates, aggressive chemical attack, corrosion of reinforcing steel, leaching of calcium hydroxide, microbiological degradation, and delayed ettringite formation [1]
  - Reduction of concrete pH due to aggressive chemical attack, leaching of calcium hydroxide, and microbiological degradation
  - Loss of concrete/steel bond due to corrosion of reinforcing steel
  - Increase in porosity and permeability due to leaching of calcium hydroxide and microbiological degradation

[1] Delayed ettringite formation may be ruled out as a credible aging mechanism by the general licensee based on an ISFSI-specific evaluation.

3.7 Aging Management Review for Spent Fuel Assemblies

Table 3-8 provides the detailed results of the AMR for the subcomponent parts of the SFAs. For each material group/environment combination and each part, the table lists the credible aging mechanisms and effects based on the information in Table 3-4.

The following aging effects/mechanisms for HBU fuel will be managed via the HBU Fuel AMP:

- Zirconium-Based Alloy
  - Loss of ductility due to hydride-induced embrittlement
  - Changes in dimensions due to thermal creep

3.8 Operating Experience Review Results – Aging Effects Identification

The review of various sources of OE discussed in Appendix 3C did not identify any aging mechanisms and/or effects that were not already identified in NUREG-2214 [3-2]. In addition, no incidents were identified where aging effects lead to the loss of intended safety functions of TN-68 dry storage cask system SSCs. Therefore, it is concluded that the effects of aging will be managed adequately so that the SSC intended safety functions will be maintained during the PEO.

3.9 References

3-1 Nuclear Regulatory Commission, NUREG-1927, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,” Revision 1, U.S. June 2016 (ADAMS Accession Number ML16179A148).

3-2 U.S. Nuclear Regulatory Commission, NUREG-2214, “Managing Aging Processes in Storage (MAPS) Report,” July 2019.

3-3 TN-68 Dry Storage Cask Updated Final Safety Analysis Report, Revision 9, May 2018.

3-4 U.S. Nuclear Regulatory Commission, NUREG-1801, “Generic Aging Lessons Learned (GALL) Report—Final Report,” Revision 2, December 2010.

3-5 U.S. Nuclear Regulatory Commission, CoC 1027 Appendix A, “TN-68 Generic Technical Specifications,” Amendment 1, October 30, 2007, Docket No. 72-1027.

3-6 U.S. Nuclear Regulatory Commission, Spent Fuel Project Office, Interim Staff Guidance 11,“Cladding Considerations for the Transportation and Storage of Spent Fuel,” Revision 3, November 17, 2003.

3-7 U.S. Nuclear Regulatory Commission, NUREG/CR-6428, “Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds,” May 1996.

3-8 U.S. Nuclear Regulatory Commission, NUREG/CR-6424, “Report on Aging of Nuclear Power Plant Reinforced Concrete Structures,” March 1996.

Table 3-1
Approved Fuel Designs

Fuel Type	Cladding Material	Burnup Limit	Source
GE 7x7	Zircaloy	40 GWd/MTU	[3-5]
GE 8x8	Zircaloy	60 GWd/MTU	[3-5]
GE 9x9	Zircaloy	60 GWd/MTU	[3-5]
GE 10x10	Zircaloy	60 GWd/MTU	[3-5]

Table 3-2
Material Groupings

Material Group	Description
Steel	Various carbon steels, alloy steels, and high-strength, low-alloy steels. Galvanized steel, aluminum-coated steel, and electroless nickel-plated steel, and are also included in the category of steel.
Stainless Steel	Stainless steel includes austenitic stainless steels and martensitic stainless steel. Chrome-plated stainless steel is also included in the category of stainless steel.
Aluminum	Includes commercially pure aluminum 1100 and precipitation-hardened alloys 6061 and 6063.
Concrete	A mixture of hydraulic cement, aggregates, and water, with or without admixtures, fibers, or other cementitious materials.
Spent Fuel Assembly Cladding	This grouping includes zircaloy cladding.
Spent Fuel Assembly Hardware	This group includes stainless steel, zirconium-based alloys, and nickel alloys.
Borated Aluminum	An aluminum alloy typically containing up to 4.5 weight percent boron. It is used as a neutron poison material. The boron is incorporated in the aluminum matrix as discrete particles of AlB_2 or TiB_2 (for alloys also containing titanium). Aluminum alloys 1100, 6063, and 6351 have been used as base materials for boron additions. This group also includes laminate composite neutron poison material consisting of a core of aluminum and boron-carbine powder sandwiched between sheets of aluminum, e.g., Boral®.
Nickel Alloy	This group is limited to the nickel alloy option for the liner of the metallic seals.
Polymers	This group includes the polypropylene used in the top neutron shield and the borated polyester used in the radial neutron shield.

Table 3-3
Maximum TN-68 Cask Temperatures During Normal Storage

Component	Maximum Temperature Table 4B.3-1 of [3-3] (°F)
Fuel Cladding	649
Basket	624
Basket Rails	402
Basket Shim	369
Inner Shell	340
Gamma Shield Shell	334
Radial Neutron Shield T_{avg}	300
Top Neutron Shield	227
Cask Outer Surface	277
Cask Lid Seal	227
Vent & Port Seals	227

Table 3-4
Summary of Potential Aging Mechanisms
(4 Pages)

Material Grouping	Aging Mechanism	Credible Environments	Section	Aging Effect
Steel	General Corrosion	Air-Outdoor Embedded-in-Concrete	3.4.3.1.1	Loss of Material
	Pitting and Crevice Corrosion	Air-Outdoor Embedded-in-Concrete	3.4.3.1.2	Loss of Material
	Galvanic Corrosion	Air-Outdoor (5)	3.4.3.1.3	Loss of Material
	Microbiologically Influenced Corrosion	Embedded-in-Concrete (3)	3.4.3.1.4	Loss of Material
	Stress Corrosion Cracking	None	3.4.3.1.5	None
	Creep	None	3.4.3.1.6	None
	Fatigue	None	3.4.3.1.7	None
	Thermal Aging	None	3.4.3.1.8	None
	Radiation Embrittlement	None	3.4.3.1.9	None
	Stress Relaxation	Air-Outdoor(2)	3.4.3.1.10	Loss of Preload
	Wear	None	3.4.3.1.11	None
Stainless Steel	General Corrosion	None	3.4.3.2.1	None
	Pitting and Crevice Corrosion	Air-Outdoor	3.4.3.2.2	Loss of Material
	Galvanic Corrosion	Air-Outdoor (1)	3.4.3.2.3	Loss of Material
	Microbiologically Influenced Corrosion	None	3.4.3.2.4	None
	Stress Corrosion Cracking	None	3.4.3.2.5	None
	Creep	None	3.4.3.2.6	None
	Fatigue	None	3.4.3.2.7	None
	Thermal Aging	None	3.4.3.2.8	None
	Radiation Embrittlement	None	3.4.3.2.9	None
	Stress Relaxation	None	3.4.3.2.10	None
	Wear	None	3.4.3.2.11	None
Aluminum	General Corrosion	Air-outdoor	3.4.3.3.1	Loss of Material
	Pitting and Crevice Corrosion	Air-outdoor	3.4.3.3.2	Loss of Material
	Galvanic Corrosion	Air-outdoor	3.4.3.3.3	Loss of Material
	Microbiologically Influenced Corrosion	None	3.4.3.3.4	None
	Creep	None	3.4.3.3.5	None
	Fatigue	None	3.4.3.3.6	None

Table 3-4
Summary of Potential Aging Mechanisms
(4 Pages)

Material Grouping	Aging Mechanism	Credible Environments	Section	Aging Effect
Aluminum	Thermal Aging	None	3.4.3.3.7	None
	Radiation Embrittlement	None	3.4.3.3.8	None
Nickel Alloy	General Corrosion	None	3.4.3.4.1	None
	Pitting and Crevice Corrosion	None	3.4.3.4.2	None
	Microbiologically Influenced Corrosion	None	3.4.3.4.3	None
	Stress Corrosion Cracking	None	3.4.3.4.4	None
	Fatigue	None	3.4.3.4.5	None
	Radiation Embrittlement	None	3.4.3.4.6	None
	Wear	None	3.4.3.4.7	None
Polymer	Boron Depletion	None	3.4.3.5.1	None
	Thermal Aging	Fully encased	3.4.3.5.2	Shrinkage / Cracking
	Radiation Embrittlement	Fully encased	3.4.3.5.3	Shrinkage / Cracking
Borated Aluminum	General Corrosion	None	3.4.3.6.1	None
	Galvanic Corrosion	None	3.4.3.6.2	None
	Boron Depletion	None	3.4.3.6.3	None
	Creep	None	3.4.3.6.4	None
	Thermal Aging	None	3.4.3.6.5	None
	Radiation Embrittlement	None	3.4.3.6.6	None
	Wet Corrosion and Blistering	None	3.4.3.6.7	None
Concrete	Freeze-Thaw	Air-Outdoor Groundwater/Soil	3.4.3.7.1	Cracking Loss of Material
	Creep	None	3.4.3.7.2	None
	Reaction with Aggregates	Air-Outdoor Groundwater/Soil	3.4.3.7.3	Cracking Loss of Strength
	Differential Settlement	Groundwater/Soil	3.4.3.7.4	Cracking

Table 3-4
Summary of Potential Aging Mechanisms
(4 Pages)

Material Grouping	Aging Mechanism	Credible Environments	Section	Aging Effect
Concrete	Aggressive Chemical Attack	Air-Outdoor Groundwater/Soil	3.4.3.7.5	Cracking
				Loss of Strength
				Loss of Material
				Reduction of Concrete pH
	Corrosion of Reinforcing Steel	Air-Outdoor Groundwater/Soil	3.4.3.7.6	Loss of Concrete/Steel Bond
				Loss of Material
				Cracking
				Loss of Strength
	Shrinkage	None	3.4.3.7.7	None
	Leaching of Calcium Hydroxide	Air-Outdoor Groundwater/Soil	3.4.3.7.8	Loss of Strength
				Increase in Porosity and Permeability
				Reduction of Concrete pH
	Radiation Damage	None	3.4.3.7.9	None
	Fatigue	None	3.4.3.7.10	None
	Dehydration at High Temperature	None	3.4.3.7.11	None
	Microbiological Degradation	Groundwater/Soil	3.4.3.7.12	Loss of Strength
				Loss of Material
				Increase in Porosity and Permeability
				Reduction of Concrete pH
	Delayed Ettringite Formation	Air-Outdoor (4) Groundwater/Soil (4)	3.4.3.7.13	Loss of Material
				Loss of Strength
				Cracking
	Salt Scaling	Air-Outdoor Groundwater/Soil	3.4.3.7.14	Loss of Material
	Hardening	None	3.4.3.7.15	None

Table 3-4
Summary of Potential Aging Mechanisms
(4 Pages)

Material Grouping	Aging Mechanism	Credible Environments	Section	Aging Effect
Spent Fuel Assembly Cladding	Hydride-Induced Embrittlement	Helium(6)	3.4.3.8.1	Loss of Ductility
	Delayed Hydride Cracking	None	3.4.3.8.2	None
	Thermal Creep	Helium(6)	3.4.3.8.3	Change in Dimensions
	Low Temperature Creep	None	3.4.3.8.4	None
	Mechanical Overload	None	3.4.3.8.5	None
	Oxidation	None	3.4.3.8.6	None
	Pitting Corrosion	None	3.4.3.8.7	None
	Galvanic Corrosion	None	3.4.3.8.8	None
	Stress Corrosion Cracking	None	3.4.3.8.9	None
	Radiation Embrittlement	None	3.4.3.8.10	None
	Fatigue	None	3.4.3.8.11	None
Spent Fuel Assembly Hardware	Creep	None	3.4.3.9.1	None
	Hydriding	None	3.4.3.9.2	None
	General Corrosion	None	3.4.3.9.3	None
	Stress Corrosion Cracking	None	3.4.3.9.4	None
	Radiation Embrittlement	None	3.4.3.9.5	None
	Fatigue	None	3.4.3.9.6	None

1. Only for stainless steel in contact with a graphite lubricant.
2. Only for bolt subcomponents.
3. Only if the concrete is exposed to a groundwater/soil environment.
4. DEF is credible for the storage pad only, unless ruled out by the general licensee based on ISFSI-specific evaluation.
5. Only for steel in contact with more noble materials such as stainless steel or a graphite lubricant.
6. Only for HBU fuel.

Table 3-5
Aging Management Review for TN-68 Cask
(4 Pages)

Component	UFSAR Drawing No.	Item No.	Subcomponent Parts	Intended Safety Function (1)		Material Group	Environment (2)	Credible Aging Mechanism	Aging Effect	Aging Management Activity
TN-68	972-70-2	1	Gamma Shield	SH, TH, SR, RT		Steel	Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
							Fully Encased (E)	None	None	None
							Fully Encased (I)	None	None	None
TN-68	972-70-2	2	Lid (including stainless steel weld overlay at seal location)	CO, SH, TH, SR,		Steel	Air-outdoor(E)	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
							Fully Encased (I)	None	None	None
						Stainless Steel	Fully Encased	None	None	None
TN-68	972-70-2	3	Inner Containment	CO, SH, TH, SR,		Steel	Fully Encased (E)	None	None	None
							Helium (I)	None	None	None
TN-68	972-70-2	4	Bottom	SH, TH, SR, RT		Steel	Fully Encased (I)	None	None	None
							Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
TN-68	972-70-2	5	Bottom Containment	CO, SH, TH, SR,		Steel	Fully Encased (E)	None	None	None
							Helium (I)	None	None	None
TN-68	972-70-2	6	Upper Trunnion	SH, SR, RT		Stainless Steel	Fully Encased (I)	None	None	None
							Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP
TN-68	972-70-2	7	Lower Trunnion	SH, SR, RT		Steel	Fully Encased (I)	None	None	None
							Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
TN-68	972-70-2	8	Shield Plate	SH, TH, SR		Steel	Helium (I)	None	None	None
							Fully Encased (E)	None	None	None
TN-68	972-70-2	9	Radial Neutron Shield	SH, TH		Polymer	Fully Encased	Thermal Aging	Shrinkage / Cracking	AMP
								Radiation Embrittlement	Shrinkage / Cracking	AMP
TN-68	972-70-2	10	Outer Shell	SH, TH, SR		Steel	Fully Encased (I)	None	None	None
							Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP

Table 3-5
Aging Management Review for TN-68 Cask
(4 Pages)

Component	UFSAR Drawing No.	Item No.	Subcomponent Parts	Intended Safety Function [1]		Material Group	Environment [2]	Credible Aging Mechanism	Aging Effect	Aging Management Activity
TN-68	972-70-2	11	Protective Cover	SH, TH		Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
TN-68	972-70-2	12	Top Neutron Shield	SH, TH		Polymer	Fully Encased[4]	Thermal Aging	Shrinkage / Cracking	AMP
								Radiation Embrittlement	Shrinkage / Cracking	AMP
TN-68	972-70-2	13	Radial N-Shield Box	SH, TH		Aluminum	Fully Encased	None	None	None
TN-68	972-70-2	14	Lid Bolt	CO, SH, TH, SR		Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
								Stress Relaxation	Loss of Preload	AMP
TN-68	972-70-2	16	Lid Seal	CO		Nickel alloy	Helium	None	None	None
						Stainless Steel	Helium	None	None	None
						Aluminum	Air-Outdoor	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
TN-68	972-70-2	18	Overpressure Port Cover	CO		Stainless Steel	Helium (I)	None	None	None
							Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP
TN-68	972-70-2	19	Overpressure Port Cover Seal	CO		Nickel alloy	Helium	None	None	None
						Stainless Steel	Helium	None	None	None
						Aluminum	Air-Outdoor	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
TN-68	972-70-2	22	Drain Port Cover	CO, SH, TH, SR		Stainless Steel	Helium (I)	None	None	None
							Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP
TN-68	972-70-2	23	Vent Port Cover	CO, SH, TH, SR		Stainless Steel	Helium (I)	None	None	None
							Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP

Table 3-5
Aging Management Review for TN-68 Cask
(4 Pages)

Component	UFSAR Drawing No.	Item No.	Subcomponent Parts	Intended Safety Function [1]		Material Group	Environment [2]	Credible Aging Mechanism	Aging Effect	Aging Management Activity
TN-68	972-70-2	24	Vent & Drain Port Cover Seal	CO		Nickel alloy	Helium	None	None	None
						Stainless Steel	Helium	None	None	None
						Aluminum	Air-Outdoor	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
TN-68	972-70-2	25	Vent & Drain Port Cover Bolts (SOC HD Cap)	CO, SR		Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
								Stress Relaxation	Loss of Preload	AMP
TN-68	972-70-2	26	Overpressure Port Cover Bolts (SOC HD Cap)	CO, SR		Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
								Stress Relaxation	Loss of Preload	AMP
TN-68	972-70-2	28	Basket Rail, Type 1	CR, TH, SR		Aluminum	Helium	None	None	None
TN-68	972-70-2	29	Basket Rail, Type 1	CR, TH, SR		Aluminum	Helium	None	None	None
TN-68	972-70-2	30	Basket Shim	CR, TH, SR		Stainless Steel	Helium	None	None	None
TN-68	972-70-2	31	Basket Shim	TH		Aluminum	Helium	None	None	None
TN-68	972-70-2	32	Fuel Compartment	SH, CR, TH, SR		Stainless Steel	Helium	None	None	None
TN-68	972-70-2	33a	Poison Plate	SH, CR, TH		Borated Aluminum	Helium	None	None	None
TN-68	972-70-2	33b	Aluminum Plate	SH, CR, TH		Aluminum	Helium	None	None	None
TN-68	972-70-2	34	Structural Plates	CR, TH, SR		Stainless Steel	Helium	None	None	None
TN-68	972-70-2	35	Flange (including Stainless Steel Weld Overlay)	CO, SH, TH, SR		Steel	Helium (I)	None	None	None
							Fully Encased (E)	None	None	None
							Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
						Stainless Steel	Fully Encased (I)	None	None	None
							Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP
TN-68	972-70-2	36	Shim	SH		Steel	Fully Encased	None	None	None

Table 3-5
Aging Management Review for TN-68 Cask
(4 Pages)

Component	UFSAR Drawing No.	Item No.	Subcomponent Parts	Intended Safety Function [1]		Material Group	Environment [2]	Credible Aging Mechanism	Aging Effect	Aging Management Activity
TN-68	972-70-2	37	Trunnion Bolt	SR, RT		Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
								Stress Relaxation	Loss of Preload	AMP
TN-68	972-70-2	39	Basket Hold down	SH, CR		Stainless Steel	Helium	None	None	None
TN-68	972-70-2	45	Threaded Insert	CO, SR		Stainless Steel	Air-Outdoor	Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
TN-68	972-70-2	47	SOC Head Cap Screw	SH, SR		Stainless Steel	Air-Outdoor	Pitting and Crevice Corrosion	Loss of Material	AMP
								Galvanic Corrosion	Loss of Material	AMP
TN-68	972-70-2	48	Shield Ring	SH		Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
								Pitting and Crevice Corrosion	Loss of Material	AMP
TN-68	972-70-7	52	Fuel Compartment Extension	CR		Stainless Steel	Helium	None	None	None
TN-68	972-70-7	53	End Cap, Top	CR		Stainless Steel	Helium	None	None	None
TN-68	972-70-7	54	End Cap Bottom	CR		Stainless Steel	Helium	None	None	None

1. The intended safety functions are: Confinement (CO), Radiation Shielding (SH), Sub-Criticality Control (CR), Structural Integrity (SR), Heat Removal Capability (TH), Retrievability (RT)
2. If the subcomponent has an internal and external surface exposed to different environments, (I) refers to an internal (or towards the interior of the TN-68 cask) environment and (E) refers to an external (or towards the exterior of the TN-68 cask) environment.
3. The metallic seals consist of an inner spring, a lining, and a jacket. The spring is [ ] (a nickel alloy) or an equivalent material. The lining is stainless steel or nickel alloy. The jacket is made of aluminum.
4. Although it is not shown on the UFSAR Drawings, the top neutron shield polypropylene material is encased in carbon steel.

Table 3-6
Aging Management Review Results Summary TN-68 Cask
(6 Pages)

Subcomponent Parts	UFSAR Drawing (Part #s)	Intended Safety Function(s) [(1)]	Material Group	Environment [(2)]	Credible Aging Mechanism	Aging Effect	Aging Management Activity
Gamma Shield	972-70-2 (1)	SH, TH, SR, RT	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
Lid (including stainless steel weld overlay at seal location)	972-70-2 (2)	CO, SH, TH, SR,	Steel	Air-outdoor(E)	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
Bottom	972-70-2 (4)	SH, TH, SR, RT	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
Upper Trunnion	972-70-2 (6)	SH, SR, RT	Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP
Lower Trunnion	972-70-2 (7)	SH, SR, RT	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
Radial Neutron Shield	972-70-2 (9)	SH, TH	Polymer	Fully Encased	Thermal Aging	Shrinkage / Cracking	AMP
					Radiation Embrittlement	Shrinkage / Cracking	AMP

Table 3-6
Aging Management Review Results Summary TN-68 Cask
(6 Pages)

Subcomponent Parts	UFSAR Drawing (Part #s)	Intended Safety Function(s) (1)	Material Group	Environment (2)	Credible Aging Mechanism	Aging Effect	Aging Management Activity
Outer Shell	972-70-2 (10)	SH, TH, SR	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
Protective Cover	972-70-2 (11)	SH, TH	Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
Top Neutron Shield	972-70-2 (12)	SH, TH	Polymer	Fully Encased	Thermal Aging	Shrinkage / Cracking	AMP
					Radiation Embrittlement	Shrinkage / Cracking	AMP
Lid Bolt	972-70-2 (14)	CO, SH, TH, SR	Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
					Stress Relaxation	Loss of Preload	AMP

Table 3-6
Aging Management Review Results Summary TN-68 Cask
(6 Pages)

Subcomponent Parts	UFSAR Drawing (Part #s)	Intended Safety Function(s) (1)	Material Group	Environment (2)	Credible Aging Mechanism	Aging Effect	Aging Management Activity
Lid Seal	972-70-2 (16)	CO	Aluminum	Air-Outdoor	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
Overpressure Port Cover	972-70-2 (18)	CO	Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP
Overpressure Port Cover Seal	972-70-2 (19)	CO	Aluminum	Air-Outdoor	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
Drain Port Cover	972-70-2 (22)	CO, SH, TH, SR	Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP
Vent Port Cover	972-70-2 (23)	CO, SH, TH, SR	Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP
Vent & Drain Port Cover Seal	972-70-2 (24)	CO	Aluminum	Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP

Table 3-6
Aging Management Review Results Summary TN-68 Cask
(6 Pages)

Subcomponent Parts	UFSAR Drawing (Part #s)	Intended Safety Function(s) (1)	Material Group	Environment (2)	Credible Aging Mechanism	Aging Effect	Aging Management Activity
Vent & Drain Port Cover Bolts (SOC HD Cap)	972-70-2 (25)	CO, SR	Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
					Stress Relaxation	Loss of Preload	AMP
Overpressure Port Cover Bolts (SOC HD Cap)	972-70-2 (26)	CO, SR	Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
					Stress Relaxation	Loss of Preload	AMP
Flange	972-70-2 (35)	CO, SH, TH, SR	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
			Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material	AMP

Table 3-6
Aging Management Review Results Summary TN-68 Cask
(6 Pages)

Subcomponent Parts	UFSAR Drawing (Part #s)	Intended Safety Function(s) (1)	Material Group	Environment (2)	Credible Aging Mechanism	Aging Effect	Aging Management Activity
Trunnion Bolt	972-70-2 (37)	SR, RT	Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
					Stress Relaxation	Loss of Preload	AMP
Threaded Inserts	972-70-2 (45)	CO, SR	Stainless Steel	Air-Outdoor	Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
SOC Head Cap Screw	972-70-2 (47)	SH, SR	Stainless Steel	Air-Outdoor	Pitting and Crevice Corrosion	Loss of Material	AMP
					Galvanic Corrosion	Loss of Material	AMP
Shield Ring	972-70-2 (48)	SH	Steel	Air-Outdoor	General Corrosion	Loss of Material	AMP
					Pitting and Crevice Corrosion	Loss of Material	AMP

1. The intended safety functions are: Confinement (CO), Radiation Shielding (SH), Sub-Criticality Control (CR), Structural Integrity (SR), Heat Removal Capability (TH), Retrievability (RT)
2. If the subcomponent has an internal and external surface exposed to different environments, (I) refers to an internal (or towards the interior of the TN-68 cask) environment and (E) refers to an external (or towards the exterior of the TN-68 cask) environment.

Table 3-7
Aging Management Review for Storage pad
(3 Pages)

Subcomponent Parts	Intended Safety Function(s) (1)	Material Group	Environment	Credible Aging Mechanism	Aging Effect	Aging Management Activity
Storage Pad	TH	Steel	Embedded-in-Concrete	General Corrosion	Loss of Material	Storage pad AMP
				Pitting and Crevice Corrosion	Loss of Material	Storage pad AMP
				Microbiologically Influenced Corrosion	Loss of Material	Storage Pad AMP
		Concrete	Air-Outdoor	Freeze-Thaw	Cracking	Storage pad AMP
					Loss of Material	Storage pad AMP
				Reaction with Aggregates	Cracking	Storage pad AMP
					Loss of Strength	Storage pad AMP
				Aggressive Chemical Attack	Cracking	Storage pad AMP
					Loss of Strength	Storage pad AMP
					Loss of Material	Storage pad AMP
					Reduction of Concrete pH	Storage pad AMP
				Corrosion of Reinforcing Steel	Loss of Concrete/Steel Bond	Storage pad AMP
					Loss of Material	Storage pad AMP
					Cracking	Storage pad AMP
					Loss of Strength	Storage pad AMP
				Leaching of Calcium Hydroxide	Loss of Strength	Storage pad AMP
					Increase in Porosity and Permeability	Storage pad AMP
					Reduction of Concrete pH	Storage pad AMP

Table 3-7
Aging Management Review for Storage pad
(3 Pages)

Subcomponent Parts	Intended Safety Function(s) [(1)]	Material Group	Environment	Credible Aging Mechanism	Aging Effect	Aging Management Activity
Storage Pad	TH	Concrete	Air-Outdoor	Delayed Ettringite Formation [(2)]	Loss of Material	Storage pad AMP
					Loss of Strength	Storage pad AMP
					Cracking	Storage pad AMP
				Salt Scaling	Loss of Material	Storage pad AMP
			Groundwater/ Soil	Freeze-Thaw	Cracking	Storage pad AMP
					Loss of Material	Storage pad AMP
				Reaction with Aggregates	Cracking	Storage pad AMP
					Loss of Strength	Storage pad AMP
				Differential Settlement	Cracking	Storage pad AMP
				Aggressive Chemical Attack	Cracking	Storage pad AMP
					Loss of Strength	Storage pad AMP
					Loss of Material	Storage pad AMP
					Reduction of Concrete pH	Storage pad AMP
				Corrosion of Reinforcing Steel	Loss of Concrete/Steel Bond	Storage pad AMP
					Loss of Material	Storage pad AMP
					Cracking	Storage pad AMP
					Loss of Strength	Storage pad AMP
				Leaching of Calcium Hydroxide	Loss of Strength	Storage pad AMP
					Increase in Porosity and Permeability	Storage pad AMP
					Reduction of Concrete pH	Storage pad AMP

Table 3-7
Aging Management Review for Storage pad
(3 Pages)

Subcomponent Parts	Intended Safety Function(s) [1]	Material Group	Environment	Credible Aging Mechanism	Aging Effect	Aging Management Activity
Storage Pad	TH	Concrete	Groundwater/ Soil	Microbiological Degradation	Loss of Strength	Storage pad AMP
					Loss of Material	Storage pad AMP
					Increase in Porosity and Permeability	Storage pad AMP
					Reduction of Concrete pH	Storage pad AMP
				Delayed Ettringite Formation [2]	Loss of Material	Storage pad AMP
					Loss of Strength	Storage pad AMP
					Cracking	Storage pad AMP
				Salt Scaling	Loss of Material	Storage pad AMP

1. The intended safety functions are: Confinement (CO), Radiation Shielding (SH), Sub-Criticality Control (CR), Structural Integrity (SR), Heat Removal Capability (TH), Retrievability (RT).
2. Delayed Ettringite Formation may be ruled out as a credible aging mechanism by the general licensee based on an ISFSI-specific evaluation.

Table 3-8
Aging Management Review for Spent Fuel Assemblies
(2 pages)

Subcomponent	Intended Safety Function (1)	Material of Construction	Material Group	Environment (2)	Credible Aging Mechanism	Aging Effect	Aging Management Activity
Fuel Cladding and End Plugs	CO, CR, SR, TH	Zircaloy	Spent Fuel Assembly Cladding	(I) Helium	Hydride-Induced Embrittlement(3)	Loss of Ductility(3)	HBU Fuel AMP
					Thermal Creep(3)	Changes in Dimensions(3)	HBU Fuel AMP
				(E) Helium	Hydride-Induced Embrittlement(3)	Loss of Ductility(3)	HBU Fuel AMP
					Thermal Creep(3)	Changes in Dimensions(3)	HBU Fuel AMP
Spacer Grid Assemblies	CR, SR	Inconel	Spent Fuel Assembly Hardware	Helium	None	None	No
		Zircaloy	Spent Fuel Assembly Hardware	Helium	None	None	No
Upper End Fitting/Nozzle (and Related subcomponents)	SR	Stainless Steel	Spent Fuel Assembly Hardware	Helium	None	None	No
		Inconel	Spent Fuel Assembly Hardware	Helium	None	None	No
Lower End Fitting/Nozzle (and Related subcomponents)	SR	Stainless Steel	Spent Fuel Assembly Hardware	Helium	None	None	No
		Inconel	Spent Fuel Assembly Hardware	Helium	None	None	No
Water Channels	CR, SR	Zircaloy	Spent Fuel Assembly Hardware	Helium	None	None	No
Fuel Channel	CR, TH	Zircaloy	Spent Fuel Assembly Hardware	Helium	None	None	No

1. The intended safety functions are: Confinement (CO), Radiation Shielding (SH), Sub-Criticality Control (CR), Structural Integrity (SR), Heat Removal Capability (TH), Retrievability (RT)

2. If the subcomponent has an internal and external surface exposed to different environments, (I) refers to the internal environment of the fuel pin and (E) refers to an external environment of the fuel pin.

3. For HBU fuel only.

APPENDIX 3A
TLAA IDENTIFICATION AND DISPOSITION

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3A.1 Introduction

Per Code of Federal Regulations (CFR) 10 CFR 72.240(c)(2) [3A-1], a certificate of compliance (CoC) renewal application must include time-limited aging analyses (TLAAs) that demonstrate that structures, systems, and components (SSCs) important to nuclear safety will continue to perform their intended safety functions for the requested period of extended operation (PEO). For an analysis to be considered a TLAA, it must meet the six selection criteria defined in 10 CFR 72.3 [3A-1].

This appendix describes the process used to identify and disposition TLAAs. It also summarizes the TLAAs that were updated to demonstrate that they have been projected to the end of the PEO.

Note that the initial license for CoC No. 1027 was for a 20-year period and that the license renewal application will request an additional 40 years. Therefore, the total PEO will be 60 years.

3A.2 Methodology for Identification and Disposition of TLAAs

3A.2.1 Identification of TLAAs

TLAAs are calculations or analyses used to demonstrate that in-scope SSCs will maintain their intended safety function throughout an explicitly stated PEO. To be considered a TLAA, the calculation/analysis must meet all six of the following criteria as defined in 10 CFR 72.3 [3A-1]:

1. Involves SSCs important to safety within the scope of the spent fuel storage certificate renewal, as delineated in Subpart L of 10 CFR Part 72 [3A-1];
2. Considers the effects of aging;
3. Involves time-limited assumptions defined by the current operating term;
4. Was determined to be relevant by the certificate holder in making a safety determination;
5. Involves conclusions or basis of conclusions related to capability of the SSCs to perform their intended safety functions; and
6. Is contained or incorporated by reference in the design bases.

TLAAs were identified via a review of the updated final safety analysis report (UFSAR) [3A-3] and the TN-68 design calculations.

The review of the UFSAR involved performing word searches for key words (i.e., year, yr, hour, hr, life, cycle, aging, fatigue).

For each hit of a key word, the context of the key word was then reviewed to determine if a time-limited assumption used in an analysis was involved, i.e., TLAA Criterion 3. If it was determined that a time limiting-assumption was not involved, the review stopped at that point. If a time-limited assumption was involved, then the context was reviewed until another criterion was not met. If all criteria were met, then the item involved a TLAA.

The review of the design basis calculations followed a similar process (i.e., first determining if a time-limited assumption was involved and if one was, then continue review until another criterion was not meet.) If all the criteria were met, then the calculation involved a TLAA.

3A.2.2 Disposition of Identified TLAAs

The identified TLAAs were dispositioned using one of the following three methods listed in Section 3.5.1 of NUREG-1927 [3A-2]:

1. Demonstrate the existing analysis remains valid for the PEO, has already considered the requested PEO, and concludes that the SSC will continue to perform its intended safety function through the end of the requested PEO;

2. Revise or update the existing analysis to demonstrate that it has been projected to the end of the requested PEO and concludes that the SSC will continue to perform its intended safety function through the end of the requested PEO; and
3. Manage the effects of aging on the SSC for the requested PEO through an aging management program (AMP).

3A.3 Identified TLAAs

Table 3A-1 provides a summary of the identified TLAAs.

3A.4 Disposition of Identified TLAAs

3A.4.1 End of Life Cavity Pressure

USFAR [3A-3] Section 2.2.5.3.3 contains a design criteria that 1 atmosphere of pressure must exist in the cavity on the coldest day at the end of life. The purpose of pressurizing the cavity above atmospheric pressure is to prevent in-leakage of air. While the design of the overpressure monitoring system ensures there will be no out-leakage past the seals during normal operations, the cavity pressure decreases over time as the decay heat decreases. The end of life cavity pressure analysis described in USFAR [3A-3] Section 7.2.2 assumes a cavity gas temperature corresponding to the end of a 40-year storage period.

Therefore, the end of life cavity pressure analysis must be revised or updated to demonstrate that the cavity pressure will be above one atmosphere at the end of the PEO, i.e., a total of 60 years.

3A.5 Summary of TLAA Identification and Disposition

The review of the approved design basis for CoC 1027 identified the following TLAA:

- ensure cavity pressure remains above one atmosphere on the coldest day at the storage period

The following TLAA must be revised or updated (see Section 3A.6 for updated analysis) to demonstrate that it has been projected to the end of the PEO and concludes that the SSC will continue to perform its intended function through the end of the PEO:

- ensure cavity pressure remains above one atmosphere on the coldest day at the storage period

3A.6 Summary of Updated End of Life Cavity Pressure Analysis

This section summarizes the calculation that demonstrates that the cavity pressure of a TN-68 cask remains above atmospheric pressure on the coldest day at the end of the PEO.

UFSAR Section 2.3.2.1 states that since the level of permeation through the confinement vessel is negligible and leakage past the higher pressure of the monitoring system is physically impossible, a decrease in cavity pressure during the storage period occurs only as a result of a reduction in the cavity gas temperature with time.

The ideal gas law governs the change in cavity pressure as the temperature inside the cavity decreases so that

$$P_2 = P_1 * T_2 / T_1$$

where,

P_1 = is the pressure at the beginning of storage;
T_1 = is the temperature at the beginning of storage;
P_2 = is the pressure at the end of storage; and
T_2 = is the temperature at the end of storage.

The backfill pressure of a TN-68 cavity is controlled by Technical Specification 3.1.2 [3A-4] “Cask Helium Backfill Pressure,” i.e. 2.0 atm (+0/-10%). Therefore, the calculation conservatively assumes an initial pressure of 1.8 atm.

From Section 7.2.2.1 of the UFSAR [3A-3], the cavity gas temperature at the time the backfilling of the cask is complete is 350 °F (810 °R).

The average cavity gas temperature at the end of 60 years of storage was determined to be 47.5 °F (507 °R) based on a decay heat value of 8.42 kW and an ambient temperature of -20 °F.

Therefore, the pressure at the end of storage on the coldest day is

$$P_2 = 1.8 \text{ atm} * (507\ °R / 810\ °R)$$

$$P_2 = 1.127 \text{ atm}$$

Since this pressure is greater than 1 atm, the cavity gas pressure within a TN-68 cask will remain above atmospheric pressure on the coldest day at the end of the storage period.

3A.7 References (Appendix A, TLAA Identification and Disposition)

3A-1 U.S. Nuclear Regulatory Commission, 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," Code of Federal Regulations.

3A-2 U.S. Nuclear Regulatory Commission, NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," Revision 1, June 2016.

3A-3 TN-68 Dry Storage Cask Updated Final Safety Analysis Report, Revision 9, May 2018.

3A-4 U.S. Nuclear Regulatory Commission, CoC 1027 Appendix A, "TN-68 Generic Technical Specifications," Amendment 1, October 30, 2017, Docket No. 72-1027.

Table 3A-1
Identified TLAAs

SSC Involved	Aging Effect Involved	UFSAR Section	Comment
All subcomponents within the confinement barrier	Loss of material due to corrosion[(1)]	UFSAR Sect 7.2.2	Ensures cask cavity remains above 1 atm at end of life

(1) While ensuring that the cask cavity pressure is greater than 1 atmosphere is not a direct aging effect, it is intended to ensure that air does not leak into the cask cavity. Air in-leakage could lead to loss of material due to corrosion, which is an aging effect.

APPENDIX 3B
SUPPLEMENTAL EVALUATIONS

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3B.1 Introduction

No supplemental evaluations were performed to support the AMR and/or an element in an AMP associated with the renewal of CoC 1027.

APPENDIX 3C
Operating Experience Review

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3C.1 Introduction

This Appendix summarizes the operating experience (OE) review conducted for the renewal of Certificate of Compliance (CoC) No. 1027. The OE review is intended to provide objective evidence to support (or refute) the conclusion that the effects of aging will be managed adequately so that the in-scope structures, systems, and components (SSCs) intended safety functions will be maintained during the period of extended operation (PEO).

3C.2 Operating Experience Review Approach

This is a document review of the sources of OE listed below looking for aging-related (versus event-driven) degradation.

The sources of potential OE reviewed are:

- Internal and industry wide condition reports
- Relevant international and non-nuclear OE
- Previous independent spent fuel storage installation (ISFSI) inspection results
- Licensee event reports
- Vendor-issued safety bulletins
- U.S. Nuclear Regulatory Commission (NRC) Generic Communications
- Updated consensus codes, standards, or guides
- Applicable industry-initiatives

3C.3 Internal and Industry Condition Reports

3C.3.1 Review of Internal Condition Reports

Table 3C-1 provides a summary of the TN Americas LLC corrective action reports (CARs) related to bolted casks components (e.g., TN-32, TN-40, and TN-68) since 2002. As shown in Table 3C-1, most conditions were either event-driven or involve subcomponents that are not in-scope for the renewal of CoC 1027. Those conditions that were aging-related involved aging mechanisms/effects already identified as requiring management.

3C.3.2 Review of CoC Users Issue Reports

Currently, there is a single ISFSI storing spent fuel using CoC 1027, the Peach Bottom Nuclear Station. Table 3C-2 provides a summary of the Peach Bottom ISFSI issue reports. As shown in Table 3C-2, most conditions were either event-driven or involve subcomponents that are not in-scope for the renewal of CoC 1027. Those conditions that were aging related involved aging mechanisms/effects already identified as requiring management.

3C.3.3 Review of Industry Condition Reports

A review of the Aging Management INPO Database (AMID) did not identify any reported OE associated with the TN-68 dry storage cask system. Also, AMID did not contain any reported OE for the similar TN-32 and TN-40 storage systems.

3C.4 Relevant International and Non-Nuclear Operating Experience

The aging management review (AMR) conducted for the renewal of CoC 1027 is based on the aging mechanisms/effects described in NUREG-2214 [3C-8]. Since the information in this report is based, in part, on international and non-nuclear OE, the AMR is also based in part on international and non-nuclear OE.

3C.5 Previous ISFSI Inspection Results

3C.5.1 Prairie Island ISFSI Pre-Application Inspection

In the summer of 2011, Northern States Power Company of Minnesota (NSPM) conducted a license renewal pre-application inspection [3C-6] of TN-40 casks at their Prairie Island Nuclear Generating Power plant ISFSI. In addition to the accessible exterior surfaces of a cask (Cask 01), the inspection also included visual inspections on the bottom of a cask and the area underneath the protective cover. Additional inspections were performed of the area underneath the protective cover of a second cask (Cask 13). The results of the pre-application inspection are summarized in [3C-6].

The results of the cask bottom inspection revealed that approximately 25% of the protective coating on the bottom of Cask 01 exhibited loss of adhesion (peeling). In areas with loss of adhesion, the base metal did not exhibit any degradation that would affect the cask's intended function. The majority of the base metal was clean, however some corrosion and corrosion product stains were observed, mainly in areas where the epoxy coating itself was cracking. In those areas, the base metal did not have observable loss of material (no depth). Additionally, the concrete under the cask exhibited no visual signs of degradation.

With the protective cover removed, inspection of the area underneath the cover of Cask 01 was performed. During this inspection, no subcomponents within the scope of License Renewal exhibited any evidence of degradation. The observable area of the lid and lid bolt heads had no indication of corrosion. A coating of rust was found on the cask flange at the protective cover interface. This rust coating originated on the carbon steel protective cover, was deposited on the cask flange, and was easily removed. The removal of this coating revealed no degradation to the stainless steel overlay surface of the cask flange and no corrosion between the lid and flange in the main lid seal area. The neutron shield bolts were removed, inspected, and observed to have no indication of corrosion with the N-5000 lubricant still intact on the threads. The neutron shield protective coating exhibited no signs of corrosion.

The protective cover was found to have uniform corrosion on the flange sealing surface outside (external side) of the O-ring seal. There was minor corrosion around the protective cover bolt holes where the bolt heads had broken the epoxy coating due to friction upon installation. The underside of the protective cover dome had no signs of degradation. The protective cover O-ring seal remained in acceptable condition with the exterior coating on the protective cover exhibiting checking on approximately 15% to 20% of the surface area.

Inspection of the area underneath the protective cover of Cask 13 was also performed with the protective cover removed. During this inspection, no subcomponents within the scope of License Renewal exhibited any evidence of degradation. The observable area of the lid and lid bolt heads had no indication of corrosion. The stainless steel flange overlay had only small stains where rust from the protective cover was deposited. The stains were removed and there was no indication of corrosion on the observable area of the flange and no corrosion was observed between the lid and flange near the main lid seal area. The neutron shield bolts were removed and inspected with no indication of corrosion; they also had the N-5000 lubricant still intact on the threads. The neutron shield had two rust stains on the protective coating directly below the access cover with one stain approximately six inches in diameter and the other approximately two inches in diameter. The protective cover was found to have corrosion on the interior. The corrosion appears to have started at the interior face of the access cover opening where the stainless steel overpressure system piping welded to the access plate made contact with the protective cover. The protective cover dome had evidence of corrosion in the area where it connected to the access plate. The access plate itself had corrosion on the entire interior surface excluding the area that was covered by the rubber gasket. However, none of these subcomponents exhibiting corrosion are within the scope of license renewal.

During the baseline inspections of Casks 01 and 13, the accessible areas of the casks were also inspected. The upper trunnions of Cask 01 exhibited some corrosion product stains on the top of the trunnion shaft. There was no indication of corrosion on all other areas inspected on Cask 01 and Cask 13.

3C.5.2 North Anna ISFSI Pre-Application Inspection

In the fall of 2015, Virginia Electric and Power Company conducted a license renewal pre-application inspection [3C-12] of TN-32 casks at their North Anna Power Station ISFSI. In addition to the accessible exterior surfaces of a cask, the inspection also included visual inspections on the bottom of a cask and beneath the protective cover of another cask. The results of the pre-application inspection are summarized in [3C-12].

The results of the bottom inspection revealed areas where portions of the protective coating exhibited loss of adhesion and areas where the coating adhered to the concrete. Rust stains were also observed. There was no detectable loss of material from the base metal. Visual inspection of the concrete beneath the cask revealed no detectable degradation, other than shrinkage cracking that occurred during concrete curing. The inside and outside of the protective cover and subassembly were found to be acceptable and no blemishes or rust stains were noted. Surface corrosion stains were noted on four of the protective cover bolt holes, but no detectable loss of material. The upper and lower trunnions exhibited surface corrosion on the top of each trunnion. The bottom half of each trunnion appeared to have a narrow ring of surface corrosion as well. There was no detectable loss of material from the trunnions. The visible portions of the lid, lid bolts, top neutron shield enclosure, neutron shield bolts, and overpressure system components were all in acceptable condition, i.e., no detectable loss of material from the base metal. Four of the twelve protective cover bolts and two of the four neutron shield bolts were inspected using VT-1 NDE techniques. Two protective cover bolts were noted to have surface corrosion stains on the bolt shank, but no detectable loss of material. The stains were easily removed with Scotch-Brite™. No corrosion or rust stains were noted on the neutron shield bolts.

3C.6 Licensee Event Reports

A search was performed of the NRC's Licensee Event Report database using the following key words:

- Cask
- Dry Fuel Storage
- ISFSI
- TN-32
- TN-40
- TN-68

No Licensee Event Reports were found associated with aging-related degradation of the TN-32, TN-40 or TN-68 dry storage cask systems.

3C.7 Vendor-Issued Safety Bulletins

Table 3C-3 provides a summary of the TN Americas LLC User's Group (TNUG) Technical Bulletins. As shown in Table 3C-3, most conditions were either event-driven or involve subcomponents that are not in-scope for the renewal of CoC 1027. The few conditions that were aging-related involved aging mechanisms/effects already identified as requiring management, e.g., galvanic corrosion and chloride-induced stress corrosion cracking (CISCC).

3C.8 NRC Generic Communications

3C.8.1 NRC Information Notices

The following NRC information notices (INs) were found to be related to aging of dry fuel storage systems.

NRC Information Notice 2011-20: Concrete Degradation by Alkali-Silica Reaction [3C-2]

This IN addresses the occurrence of alkali-silica reaction (ASR)-induced concrete degradation on Seismic Category I structures at a nuclear power plant. ASR is an aging mechanism that is evaluated in the AMR for concrete.

NRC Information Notice 2012-20: Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters [3C-3]

Several failures in austenitic stainless steels have been attributed to CISCC. The components that have failed at nuclear power plants because of this failure mechanism are made from the same types of austenitic stainless steels typically used in dry cask storage systems. CISCC is a specialized form of a stress corrosion crack, which is an aging mechanism that is evaluated in the AMR for stainless steel.

NRC Information Notice 2013-07: Premature Degradation of Spent Fuel Storage Cask Structures and Components from Environmental Moisture [3C-4]

This IN describes OE on premature degradation of structures and components important-to-safety (ITS) during spent nuclear fuel storage operations due to environmental moisture. The instances described in this IN illustrate how the intrusion of water can potentially decrease the effective life of both the structures and components of a spent fuel storage system. In one instance, the presence of water not only caused chemical degradation through oxidation of one metal, but it also facilitated the formation of a galvanic cell between two dissimilar metals, which contributed to the degradation of the secondary confinement barrier of the storage system. In another instance, water contributed to an accelerated aging process of the concrete structures of the spent fuel storage system through the freeze-thaw aging mechanism. The AMR assumes the components will be exposed to environmental moisture when in air-outdoor and groundwater/soil environments.

3C.8.2 NRC Bulletins

NRC Bulletin 96-04 Chemical, Galvanic or other Reactions in Spent Fuel Storage and Transportation Casks [3C-5]

NRC Bulletin 96-04 addresses the generation of hydrogen due to the reaction internal materials with spent fuel pool water during loading. The specific concern associated with NRC Bulletin is not related to aging of dry fuel storage systems and does not need to be addressed in the AMR.

No other NRC Bulletins were found related to aging of dry fuel storage systems.

3C.9 Updated Consensus Codes, Standards, or Guides

3C.9.1 Aging Management Reviews

The AMR conducted for the renewal of CoC 1027 is based on the aging mechanisms/effects described in NUREG-2214 [3C-8]. Therefore, the AMR is based on the version/edition/revision of the codes, standards, and guides referenced in NUREG-2214.

3C.9.2 Aging Management Programs

The AMPs for the renewal of CoC 1027 will use the most up-to-date (as of the time the AMPs are developed) version/edition/revision of the applicable codes, standards, and guidelines.

3C.10 Applicable Industry Initiatives

3C.10.1 High Burnup Fuel Demonstration Project

The Electrical Power Research Institute (EPRI) and Department of Energy (DOE) are conducting a joint High Burnup Dry Storage Cask Research and Development Project (HDRP) [3C-7] that monitors the performance of high burnup (HBU) fuel in dry storage. The HDRP is a program designed to collect data from an SNF storage system containing HBU fuel in a dry helium environment. The program entails loading and storing a TN-32B bolted lid cask (the "Research Project Cask") at Dominion Virginia Power's North Anna Power Station with intact HBU fuel (of nominal burnups ranging between 53 GWd/MTU and 58 GWd/MTU). The fuel used in the program includes four kinds of cladding (Zircaloy-4, low-tin Zircaloy-4, ZIRLO™, and M5™). The Research Project Cask was loaded and placed in service in November 2017[3C-13].

3C.10.2 Chloride-Induced Stress Corrosion Cracking

EPRI has undertaken various initiatives to improve the understanding of the CISCC cracking phenomena in terms of susceptibility [3C-9], risk [3C-10], and aging management guidance [3C-11]. While this initiative is focused on dry shielded canisters (DSC), it was reviewed and determined not to be applicable (due differences in materials, configurations, and lack of stainless steel to stainless steel welds) to the AMP for the TN-68 casks.

3C.11 Conclusion

The review of various sources of OE did not identify any aging mechanism and/or effects that were not already identified in NUREG-2214 [3C-8]. In addition, no incidents were identified where aging effects lead to the loss of intended safety functions of SSCs of TN bolted casks. Therefore, it is concluded that the effects of aging will be managed adequately so that the SSCs intended safety functions will be maintained during the PEO.

3C.12 References

3C-1 Sandia Report SAND2017-2306, "Analysis of Corrosion Residue collected from the Aluminum Basket Rails of the High-Burnup Demonstration Cask," March, 2017.

3C-2 U.S. Nuclear Regulatory Commission, Information Notice 2011-20, "Concrete Degradation by Alkali-Silica Reaction," Office of Nuclear Material Safety and Safeguards, November 18, 2011.

3C-3 U.S. Nuclear Regulatory Commission, Information Notice 2012-20, "Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canister," November 14, 2012.

3C-4 U.S. Nuclear Regulatory Commission, Information Notice 2013-07, "Premature Degradation of Spent Fuel Storage Cask Structures and Components from Environmental Moisture," April 16, 2013.

3C-5 U.S. Nuclear Regulatory Commission, Bulletin 96-04, "Chemical, Galvanic or Other Reactions in Spent Fuel Storage and Transportation Casks," July 5, 1996.

3C-6 Letter L-PI-15-082, from Scott Sharp (NSPM), to Document Control Desk (NRC), "Supplement to Prairie Island Independent Spent Fuel Storage Installation License Renewal Application – Revised Aging Management Plan (TAC No. L24592)," October 12, 2015, (Adams Accession Number ML15285A007).

3C-7 Electric Power Research Institute, "High Burnup Dry Storage Cask Research and Development Project: Final Test Plan," Rev. 0, DE-NE-0000593, February 27, 2014.

3C-8 U.S. Nuclear Regulatory Commission, NUREG-2214, "Managing Aging Process in Storage (MAPS) Report," July 2019.

3C-9 Electric Power Research Institute, "Susceptibility Assessment Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless Steel Canisters for Dry Storage Cask Systems," EPRI Report 3002005371, September 2015.

3C-10 Electric Power Research Institute, "Dry Cask Storage Welded Stainless Steel Canister Breach Consequence Analysis Scoping Study," EPRI Report 3002008192, November 2017.

3C-11 Electric Power Research Institute, "Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless Steel Canisters," EPRI Report 3002008193, March 2017.

3C-12 North Anna ISFSI – License, No SNM-2507, Renewal Application, (Adams Accession Number ML16153A140).

3C-13 Federal Register Notice for the Final Environmental Assessment for the North Anna ISFSI License Renewal, January 29, 2018, (Adams Accession Number ML17312A606).

Proprietary Information on Pages 3C-16 through 3C-21
Withheld Pursuant to 10 CFR 2.390

CHAPTER 4
AGING MANAGEMENT PROGRAMS

CONTENTS

LIST OF TABLES

4.1 Introduction

The purpose of an aging management program (AMP) is to monitor and control the degradation of structures, systems, and components (SSCs) to ensure that no aging effects result in a loss of intended safety function of the in-scope SSCs for the period of extended operations (PEO). An effective AMP prevents, mitigates, or detects aging effects and provides for the prediction of the extent of the effects of aging and timely implementation of corrective actions before there is a loss of intended safety function. AMPs are based on the results of the aging management review (AMR) for the TN-68 casks, storage pad, and high burnup (HBU) fuel presented in Chapter 3. The tables in Chapter 3 summarize the results of the AMR and identify the aging management activity (AMA) credited for managing each aging effect and aging mechanism for each component or subcomponent evaluated in the AMR.

The AMPs will apply under extended Certificate of Compliance (CoC) No. 1027 terms, that is, implementation of the AMPs will be a CoC condition for the TN-68 dry storage cask system components in service after the initial 20-year period at an independent spent fuel storage installation (ISFSI). The requirement for implementation of the proposed AMPs will be set forth under the terms and conditions of the proposed renewed certificate as described in Attachment B.

The AMPs developed to manage aging effects for the PEO are:

- TN-68 Aging Management Program
- Storage Pad Aging Management Program
- HBU Fuel Aging Management Program

4.2 Aging Management Program Elements

The structure of the AMPs is consistent with the 10 program elements described in NUREG-1927 [4-1], as follows:

1. Scope of the program: The scope of the program includes the specific SSCs and subcomponents subject to the AMP and the intended safety functions to be maintained. In addition, the element states the specific materials, environments, and aging mechanisms and effects to be managed.
2. Preventive actions: Preventive actions used to prevent aging or mitigate the rates of aging for SSCs.
3. Parameters monitored or inspected: This element identifies the specific parameters that will be monitored or inspected and describes how those parameters will be capable of identifying degradation or potential degradation before there is a loss of intended safety function.
4. Detection of aging effects: This element includes inspection and monitoring details, including method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection, and timing of inspections to ensure timely detection of aging effects. In general, the information in this element describes the “when,” “where,” and “how” of the AMP (i.e., the specific aspects of the activities to collect data as part of the inspection or monitoring activities).

   “Accessible areas” are defined as surfaces of in-scope SSCs and subcomponents that can be visually inspected by direct means without the aid of ladders, scaffolding, removal of protective cover, lifting of cask, or climbing, i.e., visually inspected while inspector is standing on the ground.

   “Normally non-accessible areas” are defined as surfaces (or portions of surfaces) of in-scope SSCs and subcomponents that can be visually inspected directly or by remote means with the aid of ladders, use of scaffolding, removal of protective cover, or lifting of cask.

   “Inaccessible areas” are defined as surfaces of in-scope SSC and subcomponents that cannot be inspected without major excavation or disassembly, e.g., below grade concrete surfaces or embedded reinforcing bars.
5. Monitoring and trending: This element describes how the data collected will be evaluated. This includes an evaluation of the results against the acceptance criteria and an evaluation regarding the rate of degradation to ensure that the timing of the next scheduled inspection will occur before there is a loss of intended safety function.
6. Acceptance criteria: Acceptance criteria, against which the need for corrective action will be evaluated, ensures that the intended safety functions and the approved design bases of the SSC are maintained during the PEO.

7. Corrective actions: Corrective actions are the measures taken when the acceptance criteria are not met. Timely corrective actions, including root cause determination and prevention of recurrence for significant conditions adverse to quality, are critical for maintaining the intended safety functions of the SSCs during the PEO.
8. Confirmation process: This element verifies that preventive actions are adequate and that effective appropriate corrective actions have been completed. The confirmation process is commensurate with the general licensee Quality Assurance (QA) Program approved under 10 CFR Part 50, Appendix B. The QA Program ensures that the confirmation process includes provisions to preclude repetition of significant conditions adverse to quality.
9. Administrative controls: Administrative controls provide a formal review and approval process in accordance with an approved QA program.
10. Operating experience (OE): The OE element of the program supports a determination that the effects of aging will be adequately managed so that the SSC's intended safety functions will be maintained during the PEO. Operating experience provides justification for the effectiveness of each AMP program element and critical feedback for enhancement.

4.3 TN-68 Aging Management Program

4.3.1 TN-68 AMP – Scope of Program

This program visually inspects and monitors the condition of the TN-68 subcomponents listed in Table 4-1. The table also lists the material and environments for each subcomponent along with the aging mechanisms and aging effects to be managed. The following aging effects and mechanisms will be managed via this AMP:

- Steel
  - Loss of material due to general, pitting, crevice, and galvanic corrosion
  - Loss of preload due to stress relaxation of bolts
- Stainless Steel
  - Loss of material due to pitting, crevice, and galvanic corrosion
- Polymers
  - Shrinkage/cracking due to thermal aging and radiation embrittlement
- Aluminum
  - Loss of material due to general, pitting, crevice, and galvanic corrosion

4.3.2 TN-68 AMP – Preventive Actions

The program is a condition-monitoring program that does not include preventive actions.

4.3.3 TN-68 AMP – Parameters Monitored or Inspected

The TN-68 AMP consists of monitoring the interseal pressure, radiation monitoring, and visual inspections.

Interseal Pressure Monitoring

The interseal pressure of the TN-68 dry storage cask seals is monitored to verify the integrity of the TN-68 dry storage cask seals. The interseal region is pressurized via the overpressure system to provide indication of cask seal integrity. A reduction of interseal pressure could indicate leakage due to loss of material of the seals or loss of preload of the bolts and would occur before there is a loss of the confinement intended safety function.

Radiation Monitoring

Periodic radiation monitoring, both gamma and neutron monitoring, will be conducted to ensure that there is no loss of the shielding intended safety function (i.e., met the requirements of 10 CFR Part 20 and 10 CFR Part 72.104) due to loss of material of the steel and stainless steel subcomponents or due to shrinkage/cracking of the polymer subcomponents. Trending the results of the radiation monitoring will enable detection of aging-related degradation before there is a loss of the shielding intended safety function.

Visual Inspections

Periodic visual inspections will be performed on the TN-68 casks looking for loss of material for steel and stainless steel subcomponents, i.e., corrosion. The frequency of these inspections will ensure that the loss of material is detected prior to the loss of an intended safety function.

4.3.4 TN-68 AMP – Detection of Aging Effects

This program manages the TN-68 aging effects by monitoring the interseal pressure, radiation monitoring, and visual inspections.

4.3.4.1 TN-68 AMP – Interseal Pressure Monitoring

The interseal pressure is monitored by measuring the pressure in the overpressure system. The TN-68 AMP utilizes the same equipment, methods, and frequency used to comply with Technical Specification 3.1.5, "Cask Interseal Pressure." Technical Specification Surveillance Requirement SR 3.1.5.1 requires verification that the cask interseal helium pressure is above 3.0 atm absolute every seven days.

4.3.4.2 TN-68 AMP – Radiation Monitoring

Detection of gamma and neutron radiation is accomplished by the placement of thermoluminescent dosimeters (TLDs) at the ISFSI perimeter fence. The TLDs for monitoring neutron radiation shall be capable of detecting low, intermediate and high energy neutrons (e.g., using a CR 39 polycarbonate chip or equivalent dosimetry.) While the TLDs may not be capable of measuring an accurate neutron dose rate (due to calibration difficulties), they are effective in detecting adverse trends in neutron dose rates.

The placement of the TLDs around the ISFSI shall be determined by the licensee considering the following:

- The objective is to monitor for increasing dose rates that could approach the 10 CFR 20.1301 and 10 CFR 72.104 regulatory limits*, and dose to individuals outside the ISFSI*.
- Casks that have been in service the longest would generally be expected to be more susceptible to potential degradation.

All Indicated Changes are in response to RAI 4

- Shadowing or shielding by other casks or structures.

The licensee shall document the basis for the placement of the TLDs within their program document.

Thermoluminescent dosimeter readings are obtained quarterly.

In addition to the TLDs, annual neutron radiation surveys shall be performed around the entire perimeter of the storage pad(s). Surveys points are to be located centrally between every two adjacent casks and approximately one foot outward from the outer edge of the storage pad (or one foot outward from a line connecting the outer edges of individual storage pads).

Results from these monitoring activities provide a means to detect deterioration of the TN-68 dry storage cask gamma and neutron shielding due to loss of material, shrinkage, or cracking.

4.3.4.3 TN-68 AMP – Visual Inspections

Accessible surfaces of all TN-68 casks will be visually inspected on an annual basis (plus 25% allowed by Technical Specification SR 3.0.2). Visual (direct or by remote means) inspections of opportunity (e.g., in the event a TN-68 cask is lifted or a protective cover is removed) will be performed on the surfaces of in-scope subcomponents in the normally non-accessible areas. A scheduled visual inspection of normally non-accessible areas of a lead TN-68 cask will be performed within two years prior to 20 years of the first loaded TN-68 cask being placed in storage, or no later than eighteen months after the effective date of the CoC renewal, whichever is later (i.e., a base line inspection), and on a frequency of every 20 ± 1 years thereafter. The lead cask is defined as the cask that has been in-service the longest. The visual inspections of the normally non-accessible areas shall be VT-3 examinations in accordance with ASME Section XI, IWA-2213.

Note: Eighteen months allows one year for development of the infrastructure for AMP implementation plus six months to complete subsequent baseline inspections.

The visual inspections are looking for loss of material indicated by corrosion or rust stains of in-scope subcomponents. Since the identification of corrosion and rust stains is a simple skill, ASME non-destructive examination (NDE) qualifications are not required to perform these visual inspections of accessible areas; however, the inspectors of accessible areas shall be trained/qualified per the licensee's specific procedures. Personnel performing visual examinations of the normally non-accessible areas shall be qualified and certified in accordance with ASME Section XI, IWA-2300.

All Indicated Changes are in response to RAI 4

Visual inspection of the flange stainless steel weld overlay and carbon steel subcomponents serve as a leading indicator for stainless and carbon steel subcomponents that are not visible during the opportunistic and scheduled inspections, e.g., seals, vent and drain port covers. Similarly, this AMP relies upon these leading indicators to manage aging effects of installed bolts, such that removal of bolts for inspection is not required.

4.3.5 TN-68 AMP – Monitoring and Trending

The inspections and monitoring activities in this AMP are performed periodically in order to identify areas of degradation. Conditions that do not meet the acceptance criteria are entered into the licensee's corrective action program. Visual inspections appropriately consider cumulative OE from previous inspections and assessments, in order to monitor and trend the progression of aging effects over time. Data taken from these inspections and monitoring activities is to be monitored by comparison to past site data taken as well as comparison to industry OE, including data gathered by the Institute of Nuclear Power Operations (INPO) Aging Management INPO Database (AMID) as discussed in Nuclear Energy Institute (NEI) 14-03 [4-2].

As described in Section 4.3.4.3, one lead TN-68 cask is to be selected for the baseline inspection and subsequent inspections of normally non-accessible areas (e.g., the bottom of the cask and under the protective weather cover). If the lead TN-68 cask is not available for subsequent inspections (e.g., has been shipped off-site), another TN-68 cask is to be selected for a new baseline inspection following the considerations/criteria in Section 4.3.4.3.

The quarterly gamma and neutron radiation readings from the TLDs on the ISFSI perimeter fence, *and the annual neutron surveys around the storage pad perimeter,* are *evaluated* to determine if there is an annual increasing trend.

4.3.6 TN-68 AMP – Acceptance Criteria

The TN-68 AMP acceptance criteria ensure that the particular structure and component intended functions are maintained under the existing design basis conditions during the PEO. If any of the acceptance criteria below are not met, further evaluation is required through the licensee's corrective action program.

4.3.6.1 TN-68 AMP – Interseal Pressure

The acceptance criterion for interseal pressure monitoring is the limit specified in Technical Specification 3.1.5, i.e., the cask interseal pressure shall be maintained at a pressure of a least 3.0 atm abs.

4.3.6.2 TN-68 AMP – Radiation Monitoring

The acceptance criterion for radiation monitoring is the absence of an annual increasing trend in neutron or gamma quarterly TLD readings at the ISFSI perimeter fence*, and the annual neutron surveys around the storage pad perimeter*.

All Indicated Changes are in response to RAI 4

4.3.6.3 TN-68 AMP – Visual Inspections

To ensure that an evaluation is performed before there is a loss of intended functions due to a loss of material, the acceptance criteria (for both the accessible and normally non-accessible areas) for the visual inspections are.

- No observed corrosion
- No rust stains on steel or stainless steel surfaces
- No rust stains on the concrete pad

4.3.7 TN-68 AMP – Corrective Actions

Site QA procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 50 Appendix B. The licensee's corrective action program ensures that conditions adverse to quality are promptly identified and corrected, including root cause determination and prevention of recurrence. Deficiencies are either corrected or are evaluated to be acceptable for continued service through engineering analysis, which provides reasonable assurance that the intended safety function is maintained consistent with current licensing basis conditions. Extent of condition investigation per the licensee's corrective action program may cause additional inspections through means of a different method, increased inspection frequency, and/or expanded inspection sample size. Engineering analysis performed to resolve deficiencies shall be based on the same methods used to provide reasonable assurance that the intended safety function is maintained consistent with the current licensing and safety basis.

4.3.8 TN-68 AMP – Confirmation Process

The confirmation process will be commensurate with the general licensee QA program approved under 10 CFR Part 50, Appendix B. The QA program ensures that the confirmation process includes provisions to verify that appropriate corrective actions have been completed and are effective. It also contains provisions to preclude repetition of significant conditions adverse to quality.

4.3.9 TN-68 AMP – Administrative Controls

Administrative controls under the CoC holder or licensee's QA procedures and corrective action program provide a formal review and approval process. Administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, and will continue for the PEO. Licensees and the CoC holder use the 10 CFR Part 72 regulatory requirements to determine if a particular aging-related degradation condition or event identified via OE, research, monitoring, or inspection is reportable to the U.S. Nuclear Regulatory Commission (NRC). Individual events and conditions not rising to the level of NRC reportability based on the criteria in 10 CFR Part 72 are communicated to the CoC holder as outlined in NEI 14-03 [4-2].

4.3.10 TN-68 AMP – Operating Experience

Appendix 3C documents the review of various sources of OE relevant to the TN-68 dry storage cask system. This review included inspections of TN-32 and TN-40 casks that have been in service for several years. While the review identified several conditions that were aging-related, no incidents were identified where aging effects lead to the loss of intended safety functions of a SSC. This OE review supports the conclusion that the effects of aging will be managed adequately so that the SSC's intended safety functions will be maintained during the PEO.

This AMP will be updated, as necessary, to incorporate new information on degradation due to aging effects identified from plant-specific inspection findings, related industry OE, and related industry research. Future plant-specific and industry aging management and aging-related OE are captured through the licensee's OE review process. The ongoing review of both plant-specific and industry OE will continue through the PEO to ensure that this AMP continues to be effective in managing the identified aging effects.

4.4 Storage Pad Aging Management Program

4.4.1 Storage Pad AMP – Scope of Program

This program visually inspects the surfaces of the storage pad subcomponents listed in Table 4-2. The table also lists the material and environments for each subcomponent along with the aging mechanisms and aging effects to be managed. The following aging effects/mechanisms will be managed via this AMP:

- Steel
  - Loss of material due to general, pitting, and crevice corrosion
- Concrete
  - Loss of material due to freeze-thaw, aggressive chemical attack, corrosion of reinforcing steel, delayed ettringite formation [1], salt scaling, and microbiological degradation
  - Cracking due to freeze-thaw, reaction with aggregates, differential settlement, aggressive chemical attack, corrosion of reinforcing steel, and delayed ettringite formation [1]
  - Loss of strength due to reaction with aggregates, aggressive chemical attack, corrosion of reinforcing steel, leaching of calcium hydroxide, delayed ettringite formation [1], and microbiological degradation
  - Reduction of concrete pH due to aggressive chemical attack, leaching of calcium hydroxide, and microbiological degradation
  - Loss of concrete/steel bond due to corrosion of reinforcing steel
  - Increase in porosity and permeability due to leaching of calcium hydroxide and microbiological degradation

4.4.2 Storage Pad AMP – Preventive Actions

The program is a condition-monitoring program that does not include preventive actions.

4.4.3 Storage Pad AMP – Parameters Monitored or Inspected

The Storage Pad AMP consists of visual inspections to monitor for material degradation.

The following accessible areas of the storage pad will undergo direct visual inspection for the aging effects listed in Table 4-2:

- The aboveground exposed surface of the storage pad

[1] Delayed ettringite formation may be ruled out as a credible aging mechanism by the general licensee based on an ISFSI-specific evaluation.

The normally non-accessible areas of the storage pad include:

- External surfaces of the storage pad under the TN-68 casks

The inaccessible areas of the storage pad include:

- Below-grade surfaces of the storage pad
- Components embedded in concrete

4.4.4 Storage Pad AMP – Detection of Aging Effects

Direct visual inspections utilizing American Concrete Institute (ACI) report ACI-349.3R [4-3], Section 3.6.1 are to be conducted of the above-grade portions of the concrete storage pad, allowing for detection of aging effects from Table 4-2. The visual inspector(s) shall meet the qualification requirements of ACI-349.3R [4.3] Chapter 7.

For storage pad concrete, crack maps are developed. Dimensioning is documented in photographic records by inclusion of a tape measure/crack gauge, a comparator, or both.

Potential degradation of the below-grade portion of the concrete pad is managed by assessing the results of the inspections of the accessible surfaces and ensuring that it is in a nonaggressive environment via groundwater sampling at a minimum of three locations in the area of the ISFSI.

The baseline AMP visual inspection and groundwater sampling is to be conducted within two years prior to 20 years of the first loaded TN-68 being placed in storage, or no later than eighteen months after the effective date of the CoC renewal, whichever is later. Subsequent inspections and groundwater sampling are to be conducted every 5 years ± 1 year following the baseline inspection.

Note: Eighteen months allows one year for development of the infrastructure for AMP implementation plus six months to complete subsequent baseline inspections.

4.4.5 Storage Pad AMP – Monitoring and Trending

The inspections and monitoring activities in this AMP are performed periodically in order to identify areas of degradation. Conditions that do not meet the acceptance criteria are entered into the licensee's corrective action program. Other conditions that are noted during the inspection and monitoring activities, such as non-conformances, failures, malfunctions, deficiencies, and deviations are addressed in accordance with the licensee's practices and expectations. Visual inspections appropriately consider cumulative OE from previous inspections and assessments in order to monitor and trend the progression of aging effects over time. Data taken from these inspections and sampling is to be monitored by comparison to past site data taken, as well as comparison to industry OE, including data gathered by the AMID as discussed in NEI 14-03 [4-2].

For storage pad concrete, crack maps are monitored and trended as a means of identifying progressive growth of defects that may indicate degradation due to specific aging effects, such as rebar corrosion. Crack maps and photographic records are compared with those from previous inspections to identify accelerated degradation of the concrete during the PEO.

4.4.6 Storage Pad AMP – Acceptance Criteria

Concrete acceptance criteria from ACI 349.3R [4-3] represent acceptable conditions for observed degradation that has been determined to be inactive. Note that the passive settlement or deflection criteria is not included because settlement would manifest itself as cracks which is already a separate criterion. These criteria are termed second-tier for structures possessing a concrete cover in excess of the minimum requirements of ACI 349. Inactive degradation can be determined by the quantitative comparison of current observed conditions with that of prior inspections. If there is a high potential for progressive degradation or propagation to occur at its present or an accelerated rate, the disposition should consider more frequent evaluations of the specific structure or initiation of repair planning.

The following findings from a visual inspection are considered acceptable:

- Absence of leaching and chemical attack, including microbiological chemical attack
- Absence of signs of corrosion in the steel reinforcement
- Drummy areas that cannot exceed the cover concrete thickness in depth
- Popouts and voids less than 50 mm (2 in.) in diameter or equivalent surface area
- Scaling less than 30 mm (1.125 in.) in depth
- Spalling less than 20 mm (0.75 in.) in depth and 200 mm (8 in.) in any dimension
- Absence of corrosion staining of undefined source on concrete surfaces
- Passive cracks less than 1 mm (0.04 in.) in maximum width
- Absence of visible signs of deterioration from alkali-aggregate reaction such as excessive out-of-plane expansion, delayed ettringite formation, or other cement/aggregate reaction

The acceptance criteria for the groundwater chemistry-sampling program are:

- pH $\geq$ 5.5
- Chlorides $\leq$ 500 ppm
- Sulfates $\leq$ 1500 ppm

If any of the above acceptance criteria are not met, further evaluation is required through the licensee's corrective action program.

4.4.7 Storage Pad AMP – Corrective Actions

Site QA procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 50 Appendix B. The licensee's corrective action program ensures that conditions adverse to quality are promptly identified and corrected, including root cause determination and prevention of recurrence. Deficiencies are either corrected or are evaluated to be acceptable for continued service through engineering analysis, which provides reasonable assurance that the intended safety function is maintained consistent with current licensing basis conditions. Extent of condition investigation per the licensee's corrective action program may cause additional inspections through means of a different method, increased inspection frequency and/or expanded inspection sample size.

4.4.8 Storage Pad AMP – Confirmation Process

The confirmation process will be commensurate with the general licensee QA program approved under 10 CFR Part 50, Appendix B. The QA program ensures that the confirmation process includes provisions to verify that appropriate corrective actions have been completed and are effective. It also contains provisions to preclude repetition of significant conditions adverse to quality.

4.4.9 Storage Pad AMP – Administrative Controls

Administrative controls under the CoC holder or licensee's QA procedures and corrective action program provide a formal review and approval process. Administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, and will continue for the PEO. Licensees and CoC holder use the 10 CFR Part 72 regulatory requirements to determine if a particular aging-related degradation condition or event identified via OE, research, monitoring, or inspection is reportable to the NRC. Individual events and conditions not rising to the level of NRC reportability based on the criteria in 10 CFR Part 72 are communicated to the CoC holder as outlined in NEI 14-03 [4-2].

4.4.10 Storage Pad AMP – Operating Experience

Appendix 3C documents the review of various sources of OE relevant to the TN-68 dry storage casks. This review included inspections of TN-32 and TN-40 casks that have been in service for several years. While the review identified several conditions that were aging-related, no incidents were identified where aging effects lead to the loss of intended safety functions of a TN-68 cask or storage pad. This OE review supports the conclusion that the effects of aging will be managed adequately so that the SSC's intended safety functions will be maintained during the PEO.

This AMP will be updated, as necessary, to incorporate new information on degradation due to aging effects identified from plant-specific inspection findings, related industry OE, and related industry research. Future plant-specific and industry aging management and aging-related OE are captured through the licensee's OE review process. The ongoing review of both plant-specific and industry OE will continue through the PEO to ensure that this AMP continues to be effective in managing the identified aging effects.

4.5 High Burnup Fuel Aging Management Program

4.5.1 HBU Fuel AMP – Scope of Program

Fuel stored in a TN-68 cask is limited to an assembly average burnup of 60 GWd/MTU. The cladding materials for the high burnup (HBU) fuel are zirconium-based, and the fuel is stored in a dry helium environment. The program relies on the joint EPRI and Department of Energy (DOE) HBU Dry Storage Cask Research and Development Project (HDRP) [4-4], conducted in accordance with the guidance in Appendix D of NUREG–1927, Revision 1 [4-1] as a surrogate demonstration program for monitoring the performance of HBU fuel in dry storage. The HDRP is a program designed to collect data from a spent nuclear fuel (SNF) storage system containing HBU fuel in a dry helium environment. The program entails loading and storing an AREVA TN-32B bolted lid cask (the "Research Project Cask") at Dominion Virginia Power's North Anna Power Station with intact HBU fuel (of nominal burnups ranging between 53 GWd/MTU and 58 GWd/MTU). The fuel to be used in the program includes four kinds of cladding (Zircaloy-4, low-tin Zircaloy-4, ZIRLO®, and M5®). The research project cask is licensed to the temperature limits contained in Interim Staff Guidance (ISG) 11, Revision 3 [4-5], and loaded in such a way that the fuel cladding temperature is as close to the limit as practicable.

The parameters of the surrogate demonstration program are applicable to the design-bases HBU fuel, as;

- maximum allowed burnup of the design-bases HBU fuel (i.e., 60.0 GWd/MTU) is on the order of the nominal burnup of the fuel in the surrogate demonstration program (i.e., 58 GWd/MTU),
- the similar cladding texture between Zircaloy-2 and M5 of recrystallized annealed (RXA) and higher hoop stresses in the M5 PWR fuel cladding when compared to Ziraloy-2 BWR fuel cladding, M5 PWR fuel cladding can be considered an enveloping surrogate for Zircalloy-2 BWR fuel cladding, and
- the cladding temperature of the HBU fuel is limited to the values in ISG-11 and the cladding temperature in the surrogate demonstration program is as close to the ISG-11 limits as practicable.

4.5.2 HBU Fuel AMP – Preventive Actions

During the initial loading operations of the cask, the design and CoC 1027 Technical Specifications (TS) require that the fuel be stored in a dry inert environment. TS 3.1.1 demonstrates that the cask cavity is dry by maintaining a cavity absolute pressure at or below 4 mbar for at least 30 minutes with the cask isolated from the vacuum pump. TS 3.1.2 requires that the cask then be backfilled with helium. These two TS requirements ensure that the HBU fuel is stored in an inert environment, thus preventing cladding degradation due to oxidation mechanisms. The cask is loaded in accordance with the criteria of ISG 11, Revision 3 [4-5].

4.5.3 HBU Fuel AMP – Parameters Monitored or Inspected

The parameters monitored or inspected are as described in the HDRP [4-4].

While the research project cask is on the storage pad, these parameters include temperature measurements at various locations within the cask. Temperature is the key driver for hydride-induced embrittlement and thermal creep.

It is anticipated that eventually the research project cask will be transported to an off-site fuel examination facility, where the cask will be reopened and the fuel visually examined for changes that occurred during drying and storage.

4.5.4 HBU Fuel AMP – Detection of Aging Effects

This AMP relies on the HDRP [4-4] as a surrogate demonstration program for monitoring the performance of HBU fuel in dry storage. The program calls for monitoring cask internal temperatures during the drying process and while the cask is in storage on the ISFSI pad. Temperature is the key driver for hydride-induced embrittlement and thermal creep.

After approximately 10 years of storage at the ISFSI site, it is anticipated that the research project cask will be transported to an off-site fuel examination facility. At the fuel examination facility, the cask will be reopened and the fuel visually examined for changes that occurred during drying and storage. Rods will be extracted from the HBU fuel assemblies and nondestructive and destructive examinations will be performed. It is anticipated that these nondestructive and destructive exams will include cladding profilometry (for creep evaluation), rod internal gas pressure, hydride content and orientation, and cladding mechanical testing (i.e., ductility testing). These examinations will be a direct indication of the susceptibility of high burnup fuel to hydride-induced embrittlement and thermal creep.

4.5.5 HBU Fuel AMP – Monitoring and Trending

As information/data from a surrogate demonstration program or from other sources (such as testing or research results and scientific analyses) become available, the licensee will monitor, evaluate, and trend the information via its operating experience program and/or its corrective action program to determine what actions should be taken.

Formal evaluations of the aggregate information from a surrogate demonstration program and other available domestic or international operating experience (including data from monitoring and inspection programs, NRC-generated communications, and other information) will be performed at specific points in time during the PEO, as delineated in Section 4.5.10.

4.5.6 HBU Fuel AMP – Acceptance Criteria

The following acceptance criteria are to be applied to the data obtained from the HDRP [4-4]. If any of the following fuel performance criteria are not met, the condition will be addressed in accordance with licensee's corrective action program:

- Cladding Temperature: The maximum cladding temperature measured is less than or equal to that predicted by the thermal analysis. A benchmarked thermal model against the demonstration test data may be used for the as-loaded configuration to show that calculated maximum cladding temperature is greater than the demonstration's measured maximum cladding temperature.
- Cladding Creep: Total creep strain extrapolated to the total approved storage duration based on the best fit to the data, accounting for initial condition uncertainty, shall be less than 1%.
- Confirmation that hydride reorientation has not compromised the ability to retrieve the spent fuel on a single-assembly basis.

4.5.7 HBU Fuel AMP – Corrective Actions

Site QA procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 50, Appendix B. The licensee's corrective action program ensures that conditions adverse to quality are promptly identified and corrected, including root cause determination and prevention of recurrence. Deficiencies are either corrected or are evaluated to be acceptable for continued service through engineering analysis, which provides reasonable assurance that the intended safety function is maintained consistent with current licensing basis conditions. Extent of condition investigation per the licensee's corrective action program may cause additional inspections through means of a different method, increased inspection frequency, and/or expanded inspection sample size.

4.5.8 HBU Fuel AMP – Confirmation Process

The confirmation process will be commensurate with the general licensee QA program approved under 10 CFR Part 50, Appendix B. The QA program ensures that the confirmation process includes provisions to verify that appropriate corrective actions have been completed and are effective. It also contains provisions to preclude repetition of significant conditions adverse to quality.

4.5.9 HBU Fuel AMP – Administrative Controls

Administrative controls under the CoC holder or licensee's QA procedures and corrective action program provide a formal review and approval process. Administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, and will continue for the PEO. Licensees and CoC holder use the 10 CFR Part 72 regulatory requirements to determine if a particular aging-related degradation condition or event identified via OE, research, monitoring, or inspection is reportable to the NRC. Individual events and conditions not rising to the level of NRC reportability based on the criteria in 10 CFR Part 72 are communicated to the CoC holder as outlined in NEI 14-03 [4-2].

4.5.10 HBU Fuel AMP – Operating Experience

Short-term testing (i.e., laboratory scale testing up to a few months) and scientific analyses examining the performance of HBU fuel have provided a foundation for the technical basis that storage of HBU fuel in the PEO may be performed safely and in compliance with regulations. However, there has been relatively little OE to date with dry storage of HBU fuel. Therefore, the HDRP is used as a surrogate program to monitor and assess data regarding HBU fuel performance to confirm there is no degradation of HBU fuel that would result in an unanalyzed configuration during the PEO.

This AMP will be updated as necessary to incorporate new information on degradation due to aging effects identified from plant-specific inspection findings, related industry OE, and related industry research. Future plant-specific and industry aging management and age-related OE are captured through the licensee's OE review process. The ongoing review of both plant-specific and industry OE will continue through the PEO to ensure that this AMP continues to be effective in managing the identified aging effects.

In addition to the ongoing OE review, this AMP requires periodic written evaluations as described in Table 4-3 of the aggregate impact of aging-related HBU fuel OE, research, monitoring, and inspections on the intended safety functions of the in-scope HBU fuel subcomponents (i.e., tollgates). While licensees and TN Americas LLC assess new information relevant to aging management in accordance with normal corrective action and OE programs, tollgates are an opportunity to seek out other information that may be available and to perform an aggregate assessment. Tollgate assessments are not stopping points. No action, other than performing an assessment and addressing relevant findings in the licensee's corrective action program, is required to continue TN-68 dry storage cask system operation. Tollgate assessment reports are not required to be submitted to the NRC, but are available for inspection. Appendix A of NEI 14-03 [4-2] provides guidance on the performance criteria for the tollgate assessments.

4.6 References

4-1 U.S. Nuclear Regulatory Commission, NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," Revision 1, June 2016 (ADAMS Accession Number ML16179A148).

4-2 Nuclear Energy Institute, NEI 14-03, "Format, Content, and Implementation Guidance for Dry Cask Storage Operations-Based Aging Management," Revision 2, December 2016.

4-3 American Concrete Institute, ACI-349.3R, "Evaluation of Existing Nuclear Safety Related Concrete Structures," 2018.

4-4 Electric Power Research Institute, "High Burnup Dry Storage Cask Research and Development Project: Final Test Plan," Revision 0, DE-NE-0000593, February 27, 2014.

4-5 NRC Spent Fuel Project Office, Interim Staff Guidance 11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3, November 17, 2003.

Table 4-1
Subcomponents Within Scope of TN-68 AMP
(3 Pages)

Subcomponent Parts	UFSAR Drawing (Part #s)	Intended Safety Function(s) (1)	Material Group	Environment (2)	Credible Aging Mechanism	Aging Effect
Gamma Shield	972-70-2 (1)	SH, TH, SR, RT	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
Lid	972-70-2 (2)	CO, SH, TH, SR	Steel	Air-Outdoor(E)	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
Bottom	972-70-2 (4)	SH, TH, SR, RT	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
Upper Trunnion	972-70-2 (6)	SH, SR, RT	Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material
Lower Trunnion	972-70-2 (7)	SH, SR, RT	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
Radial Neutron Shield	972-70-2 (9)	SH, TH	Polymer	Fully Enclosed	Thermal Aging	Shrinkage/Cracking
					Radiation Embrittlement	Shrinkage/Cracking
Outer Shell	972-70-2 (10)	SH, TH, SR	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
Protective Cover	972-70-2 (11)	SH, TH	Steel	Air-Outdoor	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
Top Neutron Shield	972-70-2 (12)	SH, TH	Polymer	Fully Enclosed	Thermal Aging	Shrinkage/Cracking
					Radiation Embrittlement	Shrinkage/Cracking

Table 4-1
Subcomponents Within Scope of TN-68 AMP
(3 Pages)

Subcomponent Parts	UFSAR Drawing (Part #s)	Intended Safety Function(s) (1)	Material Group	Environment (2)	Credible Aging Mechanism	Aging Effect
Lid Bolt	972-70-2 (14)	CO, SH, TH, SR	Steel	Air-Outdoor	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
					Stress Relaxation	Loss of Preload
Lid Seal	972-70-2 (16)	CO	Aluminum	Air-Outdoor	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
Overpressure Port Cover	972-70-2 (18)	CO	Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material
Overpressure Port Cover Seal	972-70-2 (19)	CO	Aluminum	Air-Outdoor	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
Drain Port Cover	972-70-2 (22)	CO, SH, TH, SR	Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material
Vent Port Cover	972-70-2 (23)	CO, SH, TH, SR	Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material
Vent & Drain Port Cover Seal	972-70-2 (24)	CO	Aluminum	Air-Outdoor (E)	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
Vent & Drain Port Cover Bolts (SOC HD Cap)	972-70-2 (25)	CO, SR	Steel	Air-Outdoor	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
					Stress Relaxation	Loss of Preload

Table 4-1
Subcomponents Within Scope of TN-68 AMP
(3 Pages)

Subcomponent Parts	UFSAR Drawing (Part #s)	Intended Safety Function(s) (1)	Material Group	Environment (2)	Credible Aging Mechanism	Aging Effect
Overpressure Port Cover Bolts (SOC HD Cap)	972-70-2 (26)	CO, SR	Steel	Air-Outdoor	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
					Stress Relaxation	Loss of Preload
Flange	972-70-2 (35)	CO, SH, TH, SR	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
			Stainless Steel	Air-Outdoor (E)	Pitting and Crevice Corrosion	Loss of Material
Trunnion Bolt	972-70-2 (37)	SR, RT	Steel	Air-Outdoor (E)	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
					Stress Relaxation	Loss of Preload
Threaded Inserts	972-70-2 (45)	CO, SR	Stainless Steel	Air-Outdoor	Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
SOC Head Cap Screw	972-70-2 (47)	SH, SR	Stainless Steel	Air-Outdoor	Pitting and Crevice Corrosion	Loss of Material
					Galvanic Corrosion	Loss of Material
Shield Ring	972-70-2 (48)	SH	Steel	Air-Outdoor	General Corrosion	Loss of Material
					Pitting and Crevice Corrosion	Loss of Material

1. The intended safety functions are: Confinement (CO), Radiation Shielding (SH), Sub-Criticality Control (CR), Structural Integrity (SR), Heat Removal Capability (TH), Retrievability (RT).
2. If the subcomponent has an internal and external surface exposed to different environments, (I) refers to an internal (or towards the interior of the TN-68) environment and (E) refers to an external (or towards the exterior of the TN-68) environment.

Table 4-2
Subcomponents Within Scope of Storage Pad AMP
(2 Pages)

<table>
<tr><th>Subcomponent Parts</th><th>Intended Safety Function(s) (2)</th><th>Material Group</th><th>Environment</th><th>Credible Aging Mechanism</th><th>Aging Effect</th></tr>
<tr><td rowspan="23">Storage Pad</td><td rowspan="23">TH</td><td rowspan="3">Steel</td><td rowspan="3">Embedded in Concrete</td><td>General Corrosion</td><td>Loss of Material</td></tr>
<tr><td>Pitting and Crevice Corrosion</td><td>Loss of Material</td></tr>
<tr><td>Microbiologically Influenced Corrosion</td><td>Loss of Material</td></tr>
<tr><td rowspan="20">Concrete</td><td rowspan="20">Air-Outdoor</td><td rowspan="2">Freeze-Thaw</td><td>Cracking</td></tr>
<tr><td>Loss of Material</td></tr>
<tr><td rowspan="2">Reaction with Aggregates</td><td>Cracking</td></tr>
<tr><td>Loss of Strength</td></tr>
<tr><td rowspan="4">Aggressive Chemical Attack</td><td>Cracking</td></tr>
<tr><td>Loss of Strength</td></tr>
<tr><td>Loss of Material</td></tr>
<tr><td>Reduction of Concrete pH</td></tr>
<tr><td rowspan="4">Corrosion of Reinforcing Steel</td><td>Loss of Concrete/Steel Bond</td></tr>
<tr><td>Loss of Material</td></tr>
<tr><td>Cracking</td></tr>
<tr><td>Loss of Strength</td></tr>
<tr><td rowspan="3">Leaching of Calcium Hydroxide</td><td>Loss of Strength</td></tr>
<tr><td>Increase in Porosity and Permeability</td></tr>
<tr><td>Reduction of Concrete pH</td></tr>
<tr><td rowspan="3">Delayed Ettringite Formation (1)</td><td>Loss of Material</td></tr>
<tr><td>Loss of Strength</td></tr>
<tr><td>Cracking</td></tr>
<tr><td>Salt Scaling</td><td>Loss of Material</td></tr>
</table>

Table 4-2
Subcomponents Within Scope of Storage Pad AMP
(2 Pages)

Subcomponent Parts	Intended Safety Function(s) [(2)]	Material Group	Environment	Credible Aging Mechanism	Aging Effect
Storage Pad	TH	Concrete	Groundwater/ Soil	Freeze-Thaw	Cracking
					Loss of Material
				Reaction with Aggregates	Cracking
					Loss of Strength
				Differential Settlement	Cracking
				Aggressive Chemical Attack	Cracking
					Loss of Strength
					Loss of Material
					Reduction of Concrete pH
				Corrosion of Reinforcing Steel	Loss of Concrete/Steel Bond
					Loss of Material
					Cracking
					Loss of Strength
				Leaching of Calcium Hydroxide	Loss of Strength
					Increase in Porosity and Permeability
					Reduction of Concrete pH
				Microbiological Degradation	Loss of Strength
					Loss of Material
					Increase in Porosity and Permeability
					Reduction of Concrete pH
				Delayed Ettringite Formation [(1)]	Loss of Material
					Loss of Strength
					Cracking
				Salt Scaling	Loss of Material

(1) Delayed Ettringite Formation may be ruled out as a credible aging mechanism by the general licensee based on an ISFSI-specific evaluation.

(2) The Intended Safety Functions are: Confinement (CO), Radiation Shielding (SH), Sub-Criticality Control (CR), Structural Integrity (SR), Heat Removal Capability (TH), Retrievability (RT).

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Withheld Pursuant to 10 CFR 2.390

ATTACHMENT A
CHANGES TO THE COC 1027 UPDATED FINAL SAFETY ANALYSIS REPORT

CONTENTS

LIST OF TABLES

A.1 Introduction

The proposed changes to the Certificate of Compliance (CoC) No. 1027 Updated Final Safety Analysis Report (UFSAR) for the TN-68 Dry Storage Cask System in support of the CoC 1027 renewal are discussed and described in this attachment.

Table A-1 provides a list of changed UFSAR pages, a description of each change, and the basis for the change. UFSAR Revision 9 is referenced in the table for the location of the changed UFSAR text.

Table A-1
List of UFSAR Changes Associated with CoC 1027 Renewal
(6 Pages)

UFSAR Revision 9 Section #	UFSAR Revision 9 Page	Description of Change (Newly inserted text is shown as bold and underlined; deleted text is shown by a single strike-through.)	Basis for Change
1	1.1-1	Following the third paragraph, add the following new note: **NOTE: CoC 1027 was originally licensed for 20 years. On [mm/dd/yy], the NRC approved renewal of CoC 1027 for an additional 40 years. The aging management activities (AMA) associated with this renewal apply to the previously approved amendments, and future amendments will include an aging management review (AMR) and any resultant, required aging management activities. The current aging management results are detailed in Chapter 15.**	CoC renewal discussion provided in this historical introductory UFSAR section, with pointers to AMR results.
1.2	1.2-5	Add heading "**1.2.4 Aging Management Program Requirements**" with only "Aging Management Program Requirements" underlined. Add text: **Aging management program (AMP) requirements for use of the TN-68 storage system during the period of extended storage operations are contained in Section 15.3.**	Add CoC renewal AMP requirements, with pointers to location of AMP requirements.
2.3.1	2.3-1	Change fist sentence to read: The TN-68 dry storage cask is designed to provide storage of spent fuel for at least ~~40~~ **60** years.	Bounds Extended storage period.
2.3.2.1	2.3-3	Change the second sentence of the third paragraph to read: The neutron flux is 2.34x10^5 n/cm^{2}s (Chapter 14) equivalent to less than ~~1.5~~**4.5**x10^{14} n/cm^2 after ~~20~~**60** years.	Bounds Extended storage period.
2.3.2.1	2.3-4	Change the second paragraph to read: The calculations provided in Chapter 7 define the monitoring system leakage test rate which ensures that no cavity gas can be released to the environment nor air admitted to the casks for the **first 20** ~~40~~ year storage period. **If needed, the monitoring system may be re-pressurized to maintain design conditions.** All seals are considered collectively in the analysis as the monitoring system pressure boundary. This analysis is performed in accordance with ANSI N14.5[(12)].	Clarifies that the analysis in Chapter 7 was performed for a 20-year period and that the monitoring system is capable of being re-pressurized.
2.4	2.4-1	Change the fifth paragraph to read: Cask activation analyses have been performed to quantify specific activity levels of cask materials after years of storage. **This analysis may need to be updated at the time of decommissioning to reflect actual storage conditions, e.g., years of storage.** The following assumptions were made:	Clarify that the activation analysis may need to be updated to reflect more realistic conditions.

Table A-1
List of UFSAR Changes Associated with CoC 1027 Renewal
(6 Pages)

UFSAR Revision 9 Section #	**UFSAR Revision 9 Page**	**Description of Change (Newly inserted text is shown as bold and underlined; deleted text is shown by a single strike-through.)**	**Basis for Change**
2.5	2.5-1	Change the first sentence of the second paragraph to read: Add the following sentences to the end of the second paragraph: The maximum total heat generation rate of the stored fuel is limited to 30 kW in order to keep the maximum fuel cladding temperature below the limit necessary to ensure cladding integrity[(13)] ~~for 40 years storage[(13)]~~.	The temperature limits in ISG-11 ensure that the fuel cladding will continue to perform its intended function through the PEO.
Table 2.5-1	-	Change the minimum design life to read: ~~40~~**<u>60</u>** years	Bounds extended storage period.
3.3.4	3.3-1	Change the first sentence to read: Materials must maintain the ability to perform their safety-related functions over ~~at least~~ the cask's ~~20 year~~ lifetime under the cask's thermal, radiological, corrosion, and stress environment.	Materials must maintain their safety-related functions regardless of the duration of the cask's lifetime.
3.3.4	3.3-1	The effect of fast neutron irradiation of metals is a function of the integrated fast neutron flux, which is on the order of 10^{14} n/cm^2 inside the TN-68 ~~after 40 years~~.	Bounds extended storage period.
7.1.5	7.1-6	Change the fourth paragraph to read: The overpressure system pressure is also corrected for the corresponding drop in temperature over the first year. At the end of the first year, the overpressure system pressure is 6.34 atm abs (78.5 psig). These calculations are repeated every year for ~~the~~ 20 year**<u>s</u>**~~ life of the cask~~. **<u>If needed, the monitoring system may be re-pressurized to maintain design conditions.</u>** Figure 7.1-1 illustrates the pressure drop from the overpressure system to the atmosphere. Figure 7.1-1 also illustrates the pressure drop in the cask cavity due to fuel cooling.	Clarifies that the analysis was performed for a 20-year period and that the monitoring system is capable of being re-pressurized.
7.2.1	7.2-1	Change the first sentence to read: The TN-68 dry storage cask is designed to provide storage of spent· fuel for at least ~~40~~**<u>60</u>** years.	Bounds extended storage period.

Table A-1
List of UFSAR Changes Associated with CoC 1027 Renewal
(6 Pages)

UFSAR Revision 9 Section #	UFSAR Revision 9 Page	Description of Change (Newly inserted text is shown as bold and underlined; deleted text is shown by a single strike-through.)	Basis for Change
7.2.2	7.2-1	Change the first paragraph and equation to read: ~~The TN-68 cask cavity's equilibrium pressure during normal storage conditions with no fuel rod rupture is 2.2 atm abs (17.6 psig). The internal pressure is determined on the basis that a minimum of 1 atm pressure must exist on the coldest day at the end of life. Pressure variations due to daily and seasonal changes in ambient temperature conditions will be small due to the large thermal capacity of the cask. The initial pressure of 2.2 atm abs assures that at the end of 40 years, on the coldest day (–20°F ambient), the internal pressure of the cask is:~~ ~~Pcavity = 2.20 atm abs × (596°R / 862°R) = 1. 5 atm abs (7.7 psig)‡~~ ~~‡ 596° R is the average gas cavity temperature after 40 years of storage assuming an external ambient temperature of -20° F and 21.2 kW initial heat load.~~ **<u>Initially, the cavity is pressurized with helium such that the cavity pressure remains above 1 atm abs pressure on the coldest day at the end of life. The Technical Specification minimum allowed TN-68 cask cavity pressure at the end of the helium backfill is 1.8 atm abs. A steady state run of the full cask model described in Section 4.5.1 determines the average cavity gas temperature after completion of the helium backfilling. An ambient temperature of 70°F is considered for this run. The average gas cavity temperature is 350°F (810°R), and is retrieved from the model using the methodology described in Section 4.7.1. The average gas cavity temperature after 60 years of storage assuming -20°F and a heat load of 8.42 kW was determined to be 47.5°F (507°R). After 60 years of storage, the internal pressure of the cask is:</u>** **<u>$P_{60\ yrs}$ = 1.8 atm abs X (507°R/810°R) = 1.127 atm abs</u>**	Bounds extended storage period.

Table A-1
List of UFSAR Changes Associated with CoC 1027 Renewal
(6 Pages)

UFSAR Revision 9 Section #	UFSAR Revision 9 Page	Description of Change (Newly inserted text is shown as bold and underlined; deleted text is shown by a single strike-through.)	Basis for Change
11.1.2.3	11.1-4	Change the third paragraph to read: In this latter case, leakage out of the interspace to the atmosphere and the cask cavity would occur. This would not result in release of radioactive material from the cask cavity until the pressure fell to the cask cavity pressure. At the test leak rate of 1 x 10^{-5} ref cm^3/sec, this would not occur during ~~the~~ **a** 20 year ~~storage~~ period. **If needed, the monitoring system may be re-pressurized to maintain design conditions.**	Clarifies that the analysis was performed for a 20-year period and that the monitoring system is capable of being re-pressurized.
14.1	14.1-1	Change the sixth paragraph to read: Cask activation analyses have been performed to quantify specific activities of cask materials after years of storage. **This analysis may need to be updated at the time of decommissioning to reflect actual storage conditions, e.g., years of storage.** The following assumptions were made:	Clarify that the activation analysis may need to be updated to reflect more realistic conditions.
15 (new)	new	Add centered heading "**15. AGING MANAGEMENT**"	New UFSAR chapter.
15.1 (new)	new	Add heading "**15.1 Aging Management Review**" with only "Aging Management Review" underlined. Add text: "**The aging management review (AMR) of the TN-68 dry storage cask system contained in the application for initial Certificate of Compliance (CoC) renewal [15.1] provides an assessment of aging effects that could adversely affect the ability of in-scope structures, systems, and components (SSCs) to perform their intended functions during the period of extended operation. Aging effects, and the mechanisms that cause them, were evaluated for the combinations of materials and environments identified for the subcomponent of the in-scope SSCs based on a review of the Managing Aging Processes in Storage (MAPS) Report [15.2]. Aging effects that could adversely affect the ability of the in-scope SSCs to perform their safety function(s) require an aging management activity (AMA) to address potential degradation that may occur during the extended storage period. The AMA may consist of a time-limited-aging-analysis (TLAA) or an aging management program (AMP). TLAAs and AMPs that are credited with managing aging effects during the extended storage period are discussed in Sections 15.2 and 15.3, respectively.**"	New section describing AMR.

Table A-1
List of UFSAR Changes Associated with CoC 1027 Renewal
(6 Pages)

UFSAR Revision 9 Section #	UFSAR Revision 9 Page	Description of Change (Newly inserted text is shown as bold and underlined; deleted text is shown by a single strike-through.)	Basis for Change
15.2 (new)	new	Add heading "**<u>15.2 Time-Limited Aging Analyses</u>**" with only "Time-Limited Aging Analyses" underlined. Add text: "**<u>A comprehensive review to identify the TLAAs for the in-scope SSCs of the TN-68 Dry Storage System was performed to determine the analyses that could be credited with managing aging effects over the period of extended operation. The TLAAs identified involved the in-scope SSCs, considered the effects of aging, involved explicit time-limited assumptions, provided conclusions regarding the capability of an SSC to perform its intended function through the operating term, and were contained or incorporated in the design basis. The identified TLAAs were dispositioned by demonstrating the existing analysis remains valid for the PEO or the analysis was updated. The identified TLAA was:</u>** – **<u>Ensuring cavity pressure remains above one atmosphere on the coldest day at the end of the storage period.</u>**	New section discusses TLAAs that are credited with managing aging effects during the extended storage period.
15.3 (new)	new	Add heading "**<u>15.3 Aging Management Program</u>**" with only "Aging Management Program" underlined. Add text: "**<u>Aging effects that could result in the loss of in-scope SSCs' intended function(s) are managed during the period of extended operation. Many aging effects are adequately managed for the extended storage period using TLAA, as discussed in Section 15.2. An AMP is used to manage those aging effects that are not managed by TLAA. The AMPs that manage each of the identified aging effects for all in-scope SSCs include the following:</u>** – **<u>TN-68 Aging Management Program</u>** – **<u>Storage Pad Aging Management Program</u>** – **<u>High Burnup Fuel Aging Management Program</u>**	New section describing the aging management programs credited with managing aging during the extended storage period.
15.3.1 (new)	new	Add heading "**<u>15.3.1 TN-68 Aging Management Program</u>**" with only "TN-68 Aging Management Program" underlined. Add the information from Section 4.3, including subsections and tables.	New section for this AMP.
15.3.2 (new)	new	Add table heading "**<u>15.3.2 Storage Pad Aging Management Program</u>**" with only "Storage Pad Aging Management Program" underlined. Add the information from Section 4.4, including subsections and tables..	New section for this AMP.

Table A-1
List of UFSAR Changes Associated with CoC 1027 Renewal
(6 Pages)

UFSAR Revision 9 Section #	**UFSAR Revision 9 Page**	**Description of Change (Newly inserted text is shown as bold and underlined; deleted text is shown by a single strike-through.)**	**Basis for Change**
15.3.3 (new)	new	Add table heading "**<u>15.3.3 High Burnup Fuel Aging Management Program</u>**" with only "High Burnup Fuel Aging Management Program" underlined. Add the information from Section 4.5, including subsections and tables..	New section for this AMP.
15.4 (new)	new	Add heading "**<u>15.4 Supplemental Information</u>**" with only "Supplemental Information" underlined.	New Section for Chapter 15 References.
15.4.1 (new)	New	Add heading "**<u>15.4.1 References</u>**" with only "References" underlined.	New Section for Chapter 15 References.

Table A-1
List of UFSAR Changes Associated with CoC 1027 Renewal
(6 Pages)

UFSAR Revision 9 Section #	**UFSAR Revision 9 Page**	**Description of Change (Newly inserted text is shown as bold and underlined; deleted text is shown by a single strike-through.)**	**Basis for Change**
15.4.1 (new)	New	Add the following references: **[1] (This renewal application).** **[2] U.S. Nuclear Regulatory Commission, NUREG-2214, "Managing Aging Process in Storage (MAPS) Report" July 2019.** **[3] Electric Power Research Institute, "High Burnup Dry Storage Cask Research and Development Project: Final Test Plan," Revision 0, DE-NE-0000593, February 27, 2014.** **[4] NRC Spent Fuel Project Office, Interim Staff Guidance 11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3, November 17, 2003.** **[5] U.S. Nuclear Regulatory Commission, NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," Revision 1, June 2016 (ADAMS Accession Number ML16179A148).** **[6] Nuclear Energy Institute, NEI 14-03, "Format, Content, and Implementation Guidance for Dry Cask Storage Operations-Based Aging Management," Revision 2, December 2016.** **[7] American Concrete Institute, ACI-349.3R, "Evaluation of Existing Nuclear Safety Related Concrete Structures," 2018.** **[8] Electric Power Research Institute, "High Burnup Dry Storage Cask Research and Development Project: Final Test Plan," Revision 0, DE-NE-0000593, February 27, 2014.** **[9] NRC Spent Fuel Project Office, Interim Staff Guidance 11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3, November 17, 2003.**	New Section for Chapter 15 References.

ATTACHMENT B
CHANGES TO COC 1027 AND TECHNICAL SPECIFICATIONS

CONTENTS

B.1 Mark-up of Proposed CoC Changes

The following 12 pages provide mark-ups of proposed changes to the Certificate of Compliance (CoC) No. 1027, initial Amendment 0 and Amendment 1 to support the CoC 1027 renewal.

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket Number	Amendment No.	Amendment Date	Package Identification No.
1027	05/30/00	05/28/20	72-1027	0	---	USA/72-1027

Issued To: (Name/Address)

CoC Insert A

~~Transnuclear, Inc.~~
~~4 Skyline Drive~~
~~Hawthorne, NY 10532~~

TN Americas LLC
7160 Riverwood Drive, Suite 200
Columbia, MD 21046

Safety Analysis Report Title

~~Transnuclear, Inc.~~ TN Americas LLC
Final Safety Analysis Report for the TN-68 Dry Storage Cask
Docket No. 72-1027

CONDITIONS:

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications), and the conditions specified below:

1. CASK

a. Model No.: TN-68

The TN-68 dry storage cask consists of a cask and basket assembly. The TN-68 is designed to contain up to 68 intact, unconsolidated General Electric boiling water reactor (BWR) fuel assemblies.

b. Description

The TN-68 cask being approved is described in the SAR and in NRC's Safety Evaluation Report (SER) accompanying the Certificate of Compliance (CoC). The TN-68 dry storage cask was designed by Transnuclear to store irradiated BWR spent fuel assemblies at an independent spent fuel storage installation (ISFSI).

The TN-68 cask body is a right circular cylinder composed of the following components: confinement vessel with bolted lid closure, basket for fuel assemblies, gamma shield, trunnions, neutron shield, pressure monitoring system, and weather cover.

The confinement vessel consists of an inner shell which is a welded, carbon steel cylinder with an integrally-welded, carbon steel bottom closure; a welded flange forging; a flanged and bolted carbon steel lid with an inner metallic seal; and vent and drain covers with closure bolts and inner metallic seals.

CoC Insert A

	Renewed Effective Date	Renewed Expiration Date		Revision No	Revision Effective Date	
	TBD	05/28/2060		0	NA	

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

1. b. Description (continued)

The basket consists of an assembly of stainless steel cells that are welded to stainless steel plates. Above and below the stainless steel plates are slotted neutron absorber plates which form an egg-crate structure. The neutron absorber plates provide heat conduction paths from the fuel assemblies to the cask cavity, and the neutron absorber plates provide criticality control.

The gamma shield encloses the confinement vessel and consists of an independent shell and bottom plate of carbon steel which is welded to the closure flange. An optional carbon steel gamma shield ring may be used and is installed above the neutron shield. Gamma shielding is also provided by the confinement lid.

There are four trunnions attached to the cask body. The top trunnions are used for lifting and the bottom trunnions may be used for rotating the unloaded cask.

The radial neutron shield consists of a borated polyester resin compound which surrounds the gamma shield. The resin compound is cast into long, slender aluminum containers which are enclosed in a smooth outer steel shell. The aluminum containers provide a conduction path for heat transfer from the cask body to the outer shell. Axial neutron shielding is provided by a polypropylene disk placed on the cask lid.

The overpressure monitoring system provides continuous monitoring of the pressure in the interspace between the inner and outer seals on the lid, vent, and drain port covers. The overpressure monitoring system consists of a tank filled with helium, pressure transducers or switches, and associated tubing, fittings, and valves.

The torispherical weather cover with an elastomeric seal provides weather protection for the closure lid and seal components, the top neutron shield, and the overpressure system.

The auxiliary equipment necessary for ISFSI site operation is not included as part of the TN-68 cask system reviewed for a Certificate of Compliance under 10 CFR Part 72, Subpart L. Such equipment may include, but is not limited to, special lifting devices, transfer trailers or equipment, and vacuum drying/helium leak test equipment.

2. OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, unloading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 8 of the SAR.

3. ACCEPTANCE TEST AND MAINTENANCE PROGRAM

Written cask acceptance tests and a maintenance program shall be prepared consistent with the technical basis described in Chapter 9 of the SAR.

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

4. QUALITY ASSURANCE

Activities in the areas of design, procurement, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important to safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the cask system.

5. HEAVY LOADS REQUIREMENTS

Each licensed facility must ensure that cask lifting is evaluated in accordance with the existing heavy loads requirements and procedures of the licensed facility in which the lift is made. An additional safety review (under 10 CFR 50.59 or 10 CFR 72.48, if applicable) is required to show operational compliance with existing facility/site-specific heavy loads requirements.

6. APPROVED CONTENTS

Contents of the TN-68 system must meet the specifications given in Appendix A to this certificate.

7. DESIGN FEATURES

Features or characteristics for the site, cask, or ancillary equipment must be in accordance with Appendix A to this certificate.

8. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A dry run training exercise of the loading, closure, handling, unloading and transfer of the TN-68 cask shall be conducted by the cask user prior to the first use of the system to load spent fuel assemblies. The dry run may be performed in an alternate step sequence from the actual procedures. The dry run shall include but is not limited to the following:

Preparation of the TN-68 cask for loading and moving the TN-68 cask into the spent fuel pool.

Selection and verification of specific fuel assemblies to ensure type conformance.

Loading a dummy fuel assembly into the TN-68 and performing appropriate independent verification.

Installation of the TN-68 lid and removal of the TN-68 cask from the spent fuel pool.

Cask draining, vacuum drying, helium backfilling, and leakage testing.

Loading the TN-68 cask onto the cask transporter.

Transferring the cask to the ISFSI.

Placement of the TN-68 cask at the ISFSI.

Unloading operations including reflooding.

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

9. SPECIAL REQUIREMENTS FOR CASK

Two different plate materials are approved for use as the neutron absorber in the TN-68 cask design. These neutron absorber materials are used to ensure subcriticality during loading and unloading operations that use deionized water inside the vessel. One of the approved materials is a borated wrought aluminum alloy. The other is a specific composition of a metal matrix composite material called Boralyn (TM). This composition is designated 1100/B4C/15p. Composite materials outside of this designation are envisioned as alternative materials for this application that TN could approve only after development of appropriate qualification test data that would ensure the alternative plate material meets or exceeds the service requirements for the TN-68 cask design. Criteria and methods used to establish the acceptability of such a material shall be equivalent to those used for the approved Boralyn (TM) composition. The major characteristics are described in Section 9.1.5.2 of the SER for the TN-68 cask design.

10. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to this certificate, which includes Appendix A (Technical Specifications), shall submit an application for amendment of the certificate.

11. AUTHORIZATION

The TN-68 system, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, and the attached Appendix A.

FOR THE NUCLEAR REGULATORY COMMISSION

Susan J Shankman for

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

UNITED STATES NUCLEAR REGULATORY COMMISSION

Date: May 10, 2000

CoC Insert B

Attachment:
1. Appendix A

CoC Insert B

12. UFSAR UPDATE FOR RENEWED COC

    The CoC holder shall submit an updated final safety analysis report (UFSAR) to the Commission, in accordance with 10 CFR 72.4, within 90 days of the effective date of the renewal. The UFSAR shall reflect the changes and CoC holder commitments resulting from the review and approval of the renewal of the CoC. The CoC holder shall continue to update the UFSAR pursuant to the requirements of 10 CFR 72.248.

13. 72.212 EVALUATIONS FOR RENEWED COC USE

    Any general licensee that initiates spent fuel dry storage operations with the TN-68 dry storage cask system after the effective date of the CoC renewal and any general licensee operating a TN-68 dry storage cask system as of the effective date of the CoC renewal, including those that put additional storage systems into service after that date, shall:

    a. as part of the evaluations required by 10 CFR 72.212(b)(5), include evaluations related to the terms, conditions, and specifications of this CoC amendment as modified (i.e., changed or added) as a result of the renewal of the CoC;

    b. as part of the document review required by 10 CFR 72.212(b)(6), include a review of the UFSAR changes resulting from the renewal of the CoC and the NRC Safety Evaluation Report related to the renewal of the CoC; and

    c. ensure that the evaluations required by 10 CFR 72.212(b)(7) and (8) capture the evaluations and review described in (a.) and (b.) of this CoC condition.

    The general licensee shall complete Condition 13 prior to entering the period of extended operation or no later than one year after the effective date of the CoC renewal, whichever is later.

14. AMENDMENTS AND REVISIONS FOR RENEWED COC

    All future amendments and revisions to this CoC shall include evaluations of the impacts to aging management activities (i.e., time-limited aging analyses and aging management programs) to ensure that they remain adequate for any changes to SSCs within the scope of renewal.

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket No.	Amendment No.	Amendment Effective Date	Package Identification No.
1027	05/30/00	05/28/20	72-1027	1	10/30/07	USA/72-1027

Issued To: (Name/Address)

CoC Insert A

~~Transnuclear, Inc.~~
~~7135 Minstrel Way, Suite 300~~
~~Columbia, MD 21045~~

TN Americas LLC
7160 Riverwood Drive, Suite 200
Columbia, MD 21046

Safety Analysis Report Title

~~Transnuclear, Inc.~~ TN Americas LLC
Final Safety Analysis Report for the TN-68 Dry Storage Cask
Docket No. 72-1027

CONDITIONS:

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications), and the conditions specified below:

1. CASK

a. Model No.: TN-68

The TN-68 dry storage cask consists of a cask and basket assembly. The TN-68 is designed to contain up to 68 intact, unconsolidated General Electric boiling water reactor (BWR) fuel assemblies; and up to eight (8) damaged fuel assemblies subject to the limits specified in Section 2.1.1 of Appendix A to this certificate.

b. Description

The cask being certified is described in the Safety Analysis Report (SAR) and in NRC's Safety Evaluation Report (SER) accompanying the Certificate of Compliance. The TN-68 dry storage cask was designed by Transnuclear to store irradiated BWR spent fuel assemblies at an independent spent fuel storage installation (ISFSI).

The TN-68 cask body is a right circular cylinder composed of the following components: confinement vessel with bolted lid closure, basket for fuel assemblies, gamma shield, trunnions, neutron shield, pressure monitoring system, and weather cover.

The confinement vessel consists of an inner shell which is a welded, carbon steel cylinder with an integrally-welded, carbon steel bottom closure; a welded flange forging; a flanged and bolted carbon steel lid with an inner metallic seal; and vent and drain covers with closure bolts and inner metallic seals.

CoC Insert A

	Renewed Effective Date	Renewed Expiration Date		Revision No	Revision Effective Date	
	TBD	05/28/2060		0	NA	

1. b. Description (continued)

The basket consists of an assembly of stainless steel cells that are welded to stainless steel plates. Above and below the stainless steel plates are slotted neutron absorber plates which form an egg-crate structure. The neutron absorber plates provides heat conduction paths from the fuel assemblies to the cask cavity, and the neutron absorber plates provide criticality control.

The gamma shield encloses the confinement vessel and consists of an independent shell and bottom plate of carbon steel which is welded to the closure flange. An optional carbon steel gamma shield ring may be used and is installed above the neutron shield. Gamma shielding is also provided by the confinement lid.

There are four trunnions attached to the cask body. The top trunnions are used for lifting and the bottom trunnions may be used for rotating the unloaded cask.

The radial neutron shield consists of a borated polyester resin compound which surrounds the gamma shield. The resin compound is cast into long, slender aluminum containers which are enclosed in a smooth outer steel shell. The aluminum containers provide a conduction path for heat transfer from the cask body to the outer shell. Axial neutron shielding is provided by a polypropylene disk placed on the cask lid.

The overpressure monitoring system provides continuous monitoring of the pressure in the interspace between the inner and outer seals on the lid, vent, and drain port covers. The overpressure monitoring system consists of a tank filled with helium, pressure transducers or switches, and associated tubing, fittings, and valves.

The torispherical weather cover with an elastomeric seal provides weather protection for the closure lid and seal components, the top neutron shield, and the overpressure system.

The auxiliary equipment necessary for ISFSI operation is not included as part of the TN-68 cask system reviewed for a Certificate of Compliance under 10 CFR Part 72, Subpart L. Such equipment may include, but is not limited to, special lifting devices, transfer trailers or equipment, and vacuum drying/helium leak test equipment.

2. OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, unloading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 8 of the SAR.

3. ACCEPTANCE TEST AND MAINTENANCE PROGRAM

Written cask acceptance tests and a maintenance program shall be prepared consistent with the technical basis described in Chapter 9 of the SAR.

4. QUALITY ASSURANCE

Activities in the areas of design, procurement, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important to safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the cask system.

5. HEAVY LOADS REQUIREMENTS

Each licensed facility must ensure that cask lifting is evaluated in accordance with the existing heavy loads requirements and procedures of the licensed facility in which the lift is made. An additional safety review (under 10 CFR 50.59 or 10 CFR 72.48, if applicable) is required to show operational compliance with existing facility/site-specific heavy loads requirements.

6. APPROVED CONTENTS

Contents of the TN-68 system must meet the specifications given in Appendix A to this certificate.

7. DESIGN FEATURES

Features or characteristics for the site, cask, or ancillary equipment must be in accordance with Appendix A to this certificate.

8. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A dry run training exercise of the loading, closure, handling, unloading and transfer of the TN-68 cask shall be conducted by the cask user prior to the first use of the system to load spent fuel assemblies. The dry run may be performed in an alternate step sequence from the actual procedures. The dry run shall include but is not limited to the following:

Preparation of the TN-68 cask for loading and moving the TN-68 cask into the spent fuel pool.

Selection and verification of specific fuel assemblies to ensure type conformance.

Loading a dummy fuel assembly into the TN-68 and performing appropriate independent verification.

Installation of the TN-68 lid and removal of the TN-68 cask from the spent fuel pool.

Cask draining, vacuum drying, helium backfilling, and leakage testing.

Loading the TN-68 cask onto the cask transporter.

Transferring the cask to the ISFSI.

Placement of the TN-68 cask at the ISFSI.

Unloading operations including reflooding.

9. (Deleted.)

10. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to this certificate, which includes Appendix A (Technical Specifications), shall submit an application for amendment of the certificate. This amendment does not prevent the use by the general licensee of either the original issue of the certificate or of previously approved amendments of this certificate for storage under the provisions of 10 CFR 72.210.

11. AUTHORIZATION

The TN-68 system, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, and the attached Appendix A.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: October 30, 2007

Attachment:

1. Appendix A - Technical Specifications

CoC Insert B

CoC Insert B

12. UFSAR UPDATE FOR RENEWED COC

    The CoC holder shall submit an updated final safety analysis report (UFSAR) to the Commission, in accordance with 10 CFR 72.4, within 90 days of the effective date of the renewal. The UFSAR shall reflect the changes and CoC holder commitments resulting from the review and approval of the renewal of the CoC. The CoC holder shall continue to update the UFSAR pursuant to the requirements of 10 CFR 72.248.

13. 72.212 EVALUATIONS FOR RENEWED COC USE

    Any general licensee that initiates spent fuel dry storage operations with the TN-68 dry storage cask system after the effective date of the CoC renewal and any general licensee operating a TN-68 dry storage cask system as of the effective date of the CoC renewal, including those that put additional storage systems into service after that date, shall:

    a. as part of the evaluations required by 10 CFR 72.212(b)(5), include evaluations related to the terms, conditions, and specifications of this CoC amendment as modified (i.e., changed or added) as a result of the renewal of the CoC;

    b. as part of the document review required by 10 CFR 72.212(b)(6), include a review of the UFSAR changes resulting from the renewal of the CoC and the NRC Safety Evaluation Report related to the renewal of the CoC; and

    c. ensure that the evaluations required by 10 CFR 72.212(b)(7) and (8) capture the evaluations and review described in (a.) and (b.) of this CoC condition.

    The general licensee shall complete Condition 13 prior to entering the period of extended operation or no later than one year after the effective date of the CoC renewal, whichever is later.

14. AMENDMENTS AND REVISIONS FOR RENEWED COC

    All future amendments and revisions to this CoC shall include evaluations of the impacts to aging management activities (i.e., time-limited aging analyses and aging management programs) to ensure that they remain adequate for any changes to SSCs within the scope of renewal.

B.2 Mark-up of Proposed Technical Specifications Changes

The following 4 pages provide mark-ups of proposed changes to the Technical Specifications (TS) associated with CoC 1027, initial Amendment 0 and Amendment 1 to support the CoC 1027 renewal.

5. If the measured average surface dose rates do not meet the limits of TS 5.2.3.2 or TS 5.2.3.3, whichever are lower, the licensee shall take the following actions:

    a. Notify the U.S. Nuclear Regulatory Commission (Director of the Office of Nuclear Material Safety and Safeguards) within 30 days.
    b. Administratively verify that the correct fuel was loaded, and
    c. Perform an analysis to determine that placement of the as-loaded cask at the ISFSI will not cause the ISFSI to exceed the radiation exposure limits of 10 CFR Part 20 and 72.

6. If the analysis in 5.2.3.5.c shows that placement of the as-loaded cask at the ISFSI will cause the ISFSI to exceed the radiation exposure limits of 10 CFR Part 20 and 72, the licensee shall remove all fuel assemblies from the cask within 30 days of the time of cask loading.

7. Surface dose rates shall be measured approximately at the following points (see also Figure 5.2.3-1).

    a. Above the Radial Neutron Shield (A): Midway between the top of the cask body flange and the top of the radial neutron shield. At least six measurements equally spaced circumferentially.

    b. Sides of Radial Neutron Shield (B,C,D): one sixth, one half, and five sixths of the distance from the top of the radial neutron shield. At least six measurements equally spaced circumferentially at each elevation, two of which shall be at the circumferential location of the cask trunnions. However, no measurement shall be taken directly over the trunnion.

    c. Below Radial Neutron Shield (E): Midway between the bottom of the radial neutron shield and the bottom of the cask. At least six measurements equally spaced circumferentially.

    d. Top of Cask (F, G, and H): At the center of the protective cover, one measurement (F). Halfway between the center and the knuckle at least four measurements equally spaced circumferentially (G). At the knuckle at least four measurements equally spaced circumferentially (H).

8. The average dose rates shall be determined as follows.

    In each of the four measurement zones in TS 5.2.3.7, the sum of the dose rate measurements is divided by the number of measurements to determine the average for that zone. The neutron and gamma-ray dose rates are averaged separately. Uniformly spaced dose rate measurement locations are chosen such that each point in a given zone represents approximately the same surface area.

TS Insert A

TS Insert A

5.2.4 Aging Management Program

Each general licensee shall have a program to establish, implement, and maintain written procedures for each aging management program (AMP) described in the updated final safety analysis report (UFSAR). The program shall include provisions for changing AMP elements, as necessary, and within the limitations of the approved licensing bases to address new information on aging effects based on inspection findings and/or industry operating experience provided to the general licensee during the renewal period. The program document shall contain a reference to the specific aspect of the AMP element implemented by that program document, and that reference shall be maintained even if the program document is modified.

The general licensee shall establish and implement this program document prior to entering the period of extended operation or no later than one year after the effective date of the CoC renewal, whichever is later. The general licensee shall maintain the program document for as long as the general licensee continues to operate a TN-68 dry storage cask system in service for longer than 20 years.

The general licensee shall obtain CoC holder review and approval of any changes to the Aging Management Program prior to implementation in order to verify that underlying methodologies, calculations or analyses used in determining that no aging affects result in a loss of intended safety functions remain valid.

5. If the measured average surface dose rates do not meet the limits of TS 5.2.3.2 or TS 5.2.3.3, whichever are lower, the licensee shall take the following actions:

    a. Notify the U.S. Nuclear Regulatory Commission (Director of the Office of Nuclear Material Safety and Safeguards) within 30 days.
    b. Administratively verify that the correct fuel was loaded, and
    c. Perform an analysis to determine that placement of the as-loaded cask at the ISFSI will not cause the ISFSI to exceed the radiation exposure limits of 10 CFR Part 20 and 72.

6. If the analysis in 5.2.3.5.c shows that placement of the as-loaded cask at the ISFSI will cause the ISFSI to exceed the radiation exposure limits of 10 CFR Part 20 and 72, the licensee shall remove all fuel assemblies from the cask within 30 days of the time of cask loading.

7. Surface dose rates shall be measured approximately at the following points (see also Figure 5.2.3-1).

    a. Above the Radial Neutron Shield (A): Midway between the top of the cask body flange and the top of the radial neutron shield. At least six measurements equally spaced circumferentially.

    b. Sides of Radial Neutron Shield (B,C,D): one sixth, one half, and five sixths of the distance from the top of the radial neutron shield. At least six measurements equally spaced circumferentially at each elevation, two of which shall be at the circumferential location of the cask trunnions. However, no measurement shall be taken directly over the trunnion.

    c. Below Radial Neutron Shield (E): Midway between the bottom of the radial neutron shield and the bottom of the cask. At least six measurements equally spaced circumferentially.

    d. Top of Cask (F, G, and H): At the center of the protective cover, one measurement (F). Halfway between the center and the knuckle at least four measurements equally spaced circumferentially (G). At the knuckle at least four measurements equally spaced circumferentially (H).

8. The average dose rates shall be determined as follows.

   In each of the four measurement zones in TS 5.2.3.7, the sum of the dose rate measurements is divided by the number of measurements to determine the average for that zone. The neutron and gamma-ray dose rates are averaged separately. Uniformly spaced dose rate measurement locations are chosen such that each point in a given zone represents approximately the same surface area.

TS Insert A

TS Insert A

5.2.4 Aging Management Program

Each general licensee shall have a program to establish, implement, and maintain written procedures for each aging management program (AMP) described in the updated final safety analysis report (UFSAR). The program shall include provisions for changing AMP elements, as necessary, and within the limitations of the approved licensing bases to address new information on aging effects based on inspection findings and/or industry operating experience provided to the general licensee during the renewal period. The program document shall contain a reference to the specific aspect of the AMP element implemented by that program document, and that reference shall be maintained even if the program document is modified.

The general licensee shall establish and implement this program document prior to entering the period of extended operation or no later than one year after the effective date of the CoC renewal, whichever is later. The general licensee shall maintain the program document for as long as the general licensee continues to operate a TN-68 dry storage cask system in service for longer than 20 years.

The general licensee shall obtain CoC holder review and approval of any changes to the Aging Management Program prior to implementation in order to verify that underlying methodologies, calculations or analyses used in determining that no aging affects result in a loss of intended safety functions remain valid.